

## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.1 Containment

#### BASES

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#### BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low-leakage steel vessel designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA) such that offsite radiation exposures are maintained within limits. The containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with elliptical upper and lower heads, completely enclosed by a seismic Category I reinforced concrete shield building. A 4.5 foot wide annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to permit inservice inspection and air flow over the steel dome for containment cooling. The containment utilizes the outer concrete building for shielding and a missile barrier, and the inner steel containment for leak tightness and passive containment cooling.

Containment piping penetration assemblies provide for the passage of process, service and sampling pipelines into the containment vessel while maintaining containment integrity. The shield building provides biological shielding and environmental missile protection for the containment vessel and the Nuclear Steam Supply System.

The inner steel containment and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate Surveillance Requirements conform with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

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BASES

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

- a. All penetrations required to be closed during accident conditions are either:
  1. capable of being closed by an OPERABLE automatic containment 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 3.6.3, "Containment Isolation Valves";
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and
- c. All equipment hatches are closed.

APPLICABLE  
SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. The DBA analyses assume that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of 0.10% of containment air weight of the original content of containment air after a DBA per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as  $L_a$ : the maximum allowable containment leakage rate at the calculated peak containment internal pressure ( $P_a$ ) resulting from the limiting DBA. The allowable leakage rate represented by  $L_a$  forms the basis for the acceptance criteria imposed on containment leakage rate testing.  $L_a$  is assumed to be 0.10% per day in the safety analysis.

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

Satisfactory leakage rate test results is a requirement for the establishment of containment OPERABILITY.

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

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LCO

Containment OPERABILITY is maintained by limiting leakage to  $\leq 1.0 L_a$ , except prior to the first startup after performing a required Containment Leakage Rate Testing Program Leakage Test. At this time, the applicable leakage limits must be met.

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of  $1.0 L_a$ .

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APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. The MODES 5 and 6 requirements are specified in LCO 3.6.8, "Containment Penetrations".

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ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining

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BASES

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ACTIONS

A.1 (continued)

containment OPERABLE during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required leakage test is required to be  $< 0.6 L_a$  for combined Type B and C leakage, and  $< 0.75 L_a$  for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of  $\leq 1.0 L_a$ . At  $\leq 1.0 L_a$  the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

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BASES (continued)

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- REFERENCES
1. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."
  2. Chapter 15, "Accident Analysis."
  3. Section 6.2, "Containment Systems."
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

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BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder, 10 feet in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analyses.

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APPLICABLE  
SAFETY ANALYSES

The DBA that results in the largest release of radioactive material within containment is a loss of coolant accident (LOCA) (Ref. 3). In the analyses of DBAs, it is assumed that containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of containment leakage. The containment is designed with an allowable leakage rate of 0.10% of containment air weight of the original content of containment air per day after a DBA (Ref. 2). This leakage rate is defined in 10 CFR 50,

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

Appendix J (Ref. 1), as  $L_a$ , the maximum allowable containment leakage rate at the calculated peak containment internal pressure  $P_a$  following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of offsite radiation exposures resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is necessary to support containment OPERABILITY following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry and exit from containment.

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APPLICABILITY

In MODES 1, 2, 3, and 4 a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES and large inventory of coolant. Therefore, containment air locks are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required within MODES 5 and 6 as specified in LCO 3.6.8.

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BASES (continued)

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**ACTIONS**

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair without interrupting containment integrity. If containment entry is required, it is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event that air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

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

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires containment be restored to OPERABLE status within 1 hour.

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BASES

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ACTIONS

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

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is reasonable based on engineering judgement and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified to be locked closed by administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions are modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the airlock are inoperable. With both doors in the same airlock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of an airlock for entry and exit for 7 days, under administrative controls if both airlocks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specification (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note

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(continued)

BASES

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ACTIONS

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

is not intended to preclude performing other activities (non-TS-related activities) if the containment is entered, using the inoperable airlock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open.

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

With an air lock door interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with Condition A.

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same airlock are inoperable. With both doors in the same airlock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the airlock to ensure that only one door is opened at a time (the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to airlock doors located in high radiation areas that allows these doors to be verified locked closed by administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position is small.

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

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be initiated immediately to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within

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BASES

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ACTIONS

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

limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status, assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with respect to air

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.2.1 (continued)

lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is as required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock door interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the containment air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of containment OPERABILITY if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at 24 month Frequency. The 24 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not challenged during the use of the airlock.

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BASES (continued)

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REFERENCES

1. 10 CFR 50, Appendix J, Option B "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, Performance-Based Requirements."
  2. Section 6.2, "Containment Systems."
  3. Chapter 15, "Accident Analysis."
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

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BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Section 6.2 (Ref. 1) identifies parameters which initiate isolation signal generation for containment isolation valves. The containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that containment function assumed in the safety analysis will be maintained.

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BASES

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BACKGROUND  
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Containment Air Filtration System [16-inch] purge valves

The Containment Air Filtration System operates to:

- a. Supply outside air into the containment for ventilation and cooling or heating,
- b. Reduce the concentration of noble gases within containment prior to and during personnel access, and
- c. Equalize internal and external pressures.

Since the valves used in the Containment Air Filtration System are designed to meet the requirements for automatic containment isolation valves, these valves may be opened as needed in MODES 1, 2, 3 and 4.

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APPLICABLE  
SAFETY ANALYSES

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

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 2). In the analyses for each of the accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized.

The DBA dose analysis assumes that, following containment isolation signal generation, the containment purge isolation valves are closed within 10 seconds. The remainder of the automatic isolation valves are assumed closed and the containment leakage is terminated except for the design leakage rate,  $L_s$ . Since the containment isolation valves

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**BASES**

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**APPLICABLE  
SAFETY ANALYSES  
(continued)**

are powered from the 1E division batteries no diesel generator startup time is applied.

The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the design of the containment purge isolation valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are pneumatically operated, spring closed valves that fail in the closed position and are provided with power via independent sources.

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

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**LCO**

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The valves covered by this LCO are listed along with their associated stroke times in the Section 6.2 (Ref. 1).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, or blind flanges are in place and closed systems are intact. These passive isolation valves/devices are those listed in Reference 1.

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

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BASES (continued)

**APPLICABILITY** In MODES 1, 2, 3, and 4 a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment isolation valves are not required to be OPERABLE in MODES 5 and 6 to prevent leakage of radioactive material from containment. However, containment closure capability is required in MODES 5 and 6. The requirements for containment isolation valves during MODES 5 and 6 are addressed in LCO 3.6.8, "Containment Penetrations."

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

A second Note has been added to provide clarification that, for 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 containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event that the containment isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1.

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BASES

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ACTIONS  
(continued)

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable the affected penetration flow path must be isolated. 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 and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, or a check valve with flow through the valve secured. For a penetration flow path isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within 4 hours. The 4 hour Completion Time is reasonable considering the time required to isolate the penetration, the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4, and the availability of a second barrier.

For affected penetrations that cannot be restored to OPERABLE status within the 4 hour Completion Time and have been isolated in accordance with Required Action A.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that containment penetrations that are required to be isolated following an accident and that are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of potentially being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5, if not performed within the previous 92 days," is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

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BASES

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ACTIONS

A.1 and A.2 (continued)

Condition A has been modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths with one containment isolation valve and a closed system, Condition C provides the appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high-radiation areas, and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves once they have been verified to be in the proper position, is small.

B.1

With two containment isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. 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 and deactivated automatic valve, a closed manual valve and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2 which remains in effect. This periodic verification is necessary to ensure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative control and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

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BASES

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ACTIONS  
(continued)

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration must be isolated. 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 and deactivated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event that the affected penetration is isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Ref. 4. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas, and allows these devices to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking,

sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

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(continued)

BASES

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ACTIONS  
(continued)

D.1 and D.2

If the Required Actions and associated Completion Times are 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 at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.3.1

This SR ensures that the [16 inch] purge 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 leakage outside of limits. The SR is not required to be met when the [16 inch] purge valves are open for the reasons stated. The valves may be opened for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The [16 inch] purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed, or otherwise secured 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 of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for valves outside

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BASESSURVEILLANCE  
REQUIREMENTSSR 3.6.3.2 (continued)

containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed, or otherwise secured 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 of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative control and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed 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, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

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(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. 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 times are specified in Section 6.2.3 (Ref. 1) and Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.3.5

Automatic containment isolation valves close on isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The Frequency of this SR is in accordance with the Inservice Testing Program.

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REFERENCES

1. Section 6.2, "Containment Systems."
  2. Chapter 15, "Accident Analysis."
  3. NUREG-1449, "Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States."
  4. Standard Review Plan 6.2.4.
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of transients which result in a negative pressure.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the operating band of conditions used in the containment pressure analyses for the Design Basis Events which result in internal or external pressure loads on the containment vessel. Should operation occur outside these limits, the initial containment pressure would be outside the range used for containment pressure analyses.

APPLICABLE  
SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients (Ref. 1).

The initial pressure containment used in the containment analysis was 15.7 psia (1.0 psig). This resulted in a maximum peak pressure from a DBA as indicated in reference 1. The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure,  $P_a$ , results from the limiting DBA. The maximum containment pressure resulting from the worst case DBA does not exceed the containment design pressure, 59 psig.

(continued)

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

The containment was also designed for an external pressure load equivalent to 3.0 psig. The limiting negative pressure transient is a loss of all AC power sources coincident with extreme cold weather conditions which cool the external surface of the containment vessel. The initial pressure condition used in this analysis was -0.2 psig. This resulted in a minimum pressure inside containment, as illustrated in reference 1, which is less than the design load. Other external pressure load events evaluated include:

- Failed fan cooler control
- Malfunction of containment purge system
- Inadvertent Incontainment Refueling Water Storage Tank (IRWST) drain
- Inadvertent Passive Containment Cooling System (PCS) actuation

Since the containment external pressure design limits can be met by ensuring compliance with the initial pressure condition, NUREG-1431 LCO 3.6.12, Vacuum Relief System is not applicable to the AP1000 containment.

Containment pressure satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following negative pressure transients.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure initial conditions assumed in the accident analyses are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.

(continued)

BASES

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APPLICABILITY  
(continued)

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

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ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits assumed in the containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending of both containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the main control room, including alarms, to alert the operator to an abnormal containment pressure condition.

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REFERENCES

1. Section 6.2, "Containment Analysis."
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during plant operations. The total amount of energy to be removed from containment by the passive containment cooling system during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE  
SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure transients. No two DBAs are assumed to occur simultaneously or consecutively. The

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

postulated DBAs are analyzed with regard to containment Engineered Safety Feature (ESF) systems, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure, resulting in one Passive Containment Cooling System flow path being rendered inoperable.

The limiting DBA for the maximum peak containment air temperature is a LOCA or SLB. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F.

The DBA temperature transients are used to establish the environmental qualification operating envelope for containment. The basis of the containment environmental qualification temperature envelope is to ensure the performance of safety related equipment inside containment (Ref. 2). The containment vessel design temperature is 300°F. The containment vessel temperature remains below 300°F for DBAs. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBAs.

The temperature limit is also used in the depressurization analyses to ensure that the minimum pressure limit is maintained following an inadvertent actuation of the Passive Containment Cooling System (Ref. 1).

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is an SLB or LOCA. The temperature limit is used in the DBA analyses to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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(continued)

BASES

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LCO                      During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is computed to remain within acceptable limits. As a result, the ability of containment to perform its design function is ensured.

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APPLICABILITY            In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

---

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored to within its limit within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the conservative analysis to variations in this parameter, and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit 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 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.5.1

Verifying that the containment average air temperature is within the LCO limit ensures that containment operation remains within the limits assumed for the containment analyses. In order to determine the containment average air temperature, a weighted average is calculated using measurements taken at locations within the containment selected to provide a representative sample of the associated containment atmosphere. The 24 hour Frequency of this Surveillance Requirement is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the main control room, including alarms, to alert the operator to an abnormal containment temperature condition.

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REFERENCES

1. Section 6.2, "Containment Systems."
  2. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.6 Passive Containment Cooling System (PCS) - Operating

#### BASES

#### BACKGROUND

The PCS provides containment cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA). The Passive Containment Cooling System is designed to meet the requirements of 10 CFR 50 Appendix A GDC 38 "Containment Heat Removal" and GDC 40 "Testing of Containment Heat Removal Systems" (Ref. 1).

The PCS consists of a 800,000 gal (nominal) cooling water tank, four headered tank discharge lines with flow restricting orifices, and two separate full capacity discharge flow paths to the containment vessel with 3 sets of isolation valves, each capable of meeting the design bases. Algae growth is not expected within the PCCWST; however, to assure water clarity is maintained, a prevailing concentration of hydrogen peroxide is maintained at 50 ppm. The recirculation pumps and heater provide freeze protection for the passive containment cooling water storage tank. However, OPERABILITY of the tank is assured by compliance with the temperature limits specified in SR 3.6.6.1 and not by the recirculation pumps and heater. In addition to the recirculation pumps and heater, the PCS water storage tank temperature can be maintained within limits by the ambient temperature, the large thermal inertia of the tank, or heat from other sources. The PCS valve room temperature must not be below freezing for an extended period to assure the water flow path to the containment shell is available. The isolation valves on each flow path are powered from a separate Division.

Upon actuation of the isolation valves, gravity flow of water from the cooling water tank (contained in the shield building structure above the containment) onto the upper portion of the containment shell reduces the containment pressure and temperature following a DBA. The flow of water to the containment shell surface is initially established to assure that the required short term containment cooling requirements following the postulated worst case LOCA are achieved. As the decay heat from the core becomes less with

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BASES

BACKGROUND  
(continued)

time, the water flow to the containment shell is reduced in three steps. The change in flow rate is attained without active components in the system and is dependent only on the decreasing water level in the elevated storage tank. In order to ensure the containment surface is adequately and effectively wetted, the water is introduced at the center of the containment dome and flows outward. Weirs are placed on the dome surface to distribute the water and ensure effective wetting of the dome and vertical sides of the containment shell. The monitoring of the containment surface through the Reliability Assurance Program (RAP) and the Inservice Testing Program assures containment surface does not unacceptably degrade containment heat removal performance. During the initial test program, the containment coverage will be measured at the base of the upper annulus in addition to the coverage at the spring line for the full flow case and a lower flow case with PCS recirculation pumps delivering to the containment shell. These benchmark values at the base of the upper annulus will be used to develop acceptance criteria for technical specifications. Contamination can be removed by PCS actuation and by using coating vendor cleaning procedures.

The path for the natural circulation of air is from the air intakes in the shield building, down the outside of the baffle, up along the containment shell to the top, center exit in the shield building and is always open. The drains in the upper annulus region must be clear to prevent water from blocking the air flow path. Heat is removed from within the containment utilizing the steel containment shell as the heat transfer surface combining conductive heat transfer to the water film, convective heat transfer from the water film to the air, radiative heat transfer from the film to the air baffle, and mass transfer (evaporation) of the water film into the air. As the air heats up and water evaporates into the air, it becomes less dense than the cooler air in the air inlet annulus. This differential causes an increase in the natural circulation of the air upward along the containment surface, with heated air/water vapor exiting the top/center of the shield building. Additional system design details are provided in reference 3.

The PCS is actuated either automatically, by a containment High-2 pressure signal, or manually. Automatic actuation opens the cooling water tank discharge valves, allowing gravity flow of the cooling water onto the containment shell. The manual containment cooling actuation consists of

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BASES

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BACKGROUND  
(continued)

four momentary controls, if two associated controls are operated simultaneously actuation will occur in all divisions. The discharge continues for at least three days.

The PCS is designed to limit post-accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a DBA.

The PCS is an ESF system and is designed to ensure that the heat removal capability required during the post accident period can be attained.

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APPLICABLE  
SAFETY ANALYSES

The Passive Containment Cooling System limits the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF system, assuming the loss of one Class 1E Engineered Safety Features Actuation Cabinet (ESFAC) Division, which is the worst case single active failure and results in one PCS flow path being inoperable.

The analyses and evaluations assume a unit specific power level of 3400 MWt, one passive containment cooling train operating, and initial (pre-accident) containment conditions of 120°F and 1.0 psig. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the effectiveness of the Passive Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment

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(continued)

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

backpressure. For these calculations, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the calculated transient containment pressures in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment cooling system performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 100% of the required peak cooling capacity during the post accident condition.

The modeled Passive Containment Cooling System actuation response time from the containment analysis is based upon a response time associated with exceeding the containment High-2 pressure setpoint to opening of isolation valves.

The Passive Containment Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

During a DBA, one passive containment cooling water flow path is required to maintain the containment peak pressure and temperature below the design limits (Ref. 3). To ensure that this requirement is met, two passive containment cooling water flow paths are provided. Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE.

The PCS includes a cooling water tank, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the cooling water tank upon an actuation signal. An OPERABLE flow path consists of a normally closed valve capable of automatically opening in series with a normally open valve. For the two flow paths containing air-operated valves, it is preferred because of PRA insights that these valves be normally closed.

The PCS cooling water storage tank ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA). To be considered OPERABLE, the PCS cooling water storage tank must meet the water volume and temperature limits

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BASES

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LCO  
(continued) established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

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APPLICABILITY In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the PCS.

During shutdown the PCS may be required to remove heat from containment. The requirements in MODES 5 and 6 are specified in LCO 3.6.7, Passive Containment Cooling System (PCS) - Shutdown.

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ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of DBA occurring during this period.

C.1

If the cooling water tank is inoperable, it must be restored to OPERABLE status within 8 hours. The tank may be declared

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BASES

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ACTIONS

C.1 (continued)

inoperable due to low water level or temperature out of limits. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1 and D.2

If any of the Required Actions and associated Completion Times for Condition A or B are not met, or if the LCO is not met for reasons other than Condition A or B, 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 6 hours and to MODE 5 within 84 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 extended interval to reach MODE 5 allows additional time and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.1

This surveillance requires verification that the cooling water temperature is within the limits assumed in the accident analyses. The 24 hour Frequency is adequate to identify a temperature change that would approach the temperature limit and has been shown to be acceptable in similar applications.

The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are between 32°F and 100°F, which ensures that the PCS tank water temperature remains within the limits specified in SR 3.6.6.1.

SR 3.6.6.2

Verification that the cooling water volume is above the required minimum ensures that a sufficient supply is available for containment cooling. Since the cooling water

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.6.2 (continued)

volume is normally stable and low level is indicated by a main control room alarm, a 7 day Frequency is appropriate and has been shown to be acceptable in similar applications.

SR 3.6.6.3

Verifying the correct alignment of power operated, and automatic valves, excluding check valves, in the Passive Containment Cooling System provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct positions prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification, through control room instrumentation or a system walkdown, that valves capable of potentially being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single flow path. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.6.4

This SR requires verification that each automatic isolation valve actuates to its correct position upon receipt of an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform these Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability (and confirmed by operating experience) of the equipment. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.6.6.5

Periodic inspections of the PCS air flow path from the shield building annulus inlet to the exit ensure that it is unobstructed, the baffle plates are properly installed, and the upper annulus safety-related drains are unobstructed. Although there are no anticipated mechanisms which would cause air flow path or annulus drain obstruction and the effect of a missing air baffle section is small, it is considered prudent to verify this capability every 24 months. Additionally, the 24 month Frequency is based on the desire to perform this Surveillance under conditions that apply during a plant outage, on the need to have access to the locations, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation in similar situations.

SR 3.6.6.6

This SR requires performance of a Passive Containment Cooling System test to verify system flow and water coverage capabilities. The system performance test demonstrates that the containment cooling capability assumed in accident analyses is maintained by verifying the flow rates via each standpipe and measurement of containment wetting coverage. The System Level Operability Testing Program provides specific test requirements and acceptance criteria. Although the likelihood that system performance would degrade with time is low, it is considered prudent to periodically verify system performance. The first refueling and 10 year Frequency is based on the ability of the more frequent surveillances to verify the OPERABILITY of the active components and features which could degrade with time.

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REFERENCES

1. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
  2. 10 CFR 50, Appendix K, "ECCS Evaluation Models."
  3. Chapter 6.2, "Containment Systems."
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B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 Passive Containment Cooling System (PCS) – Shutdown

BASES

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BACKGROUND            A description of the PCS is provided in the Bases for LCO 3.6.6, "Passive Containment Cooling System – Operating."

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APPLICABLE SAFETY ANALYSES        The PCS limits the temperature and pressure that could be experienced during shutdown following a loss of decay heat removal.

For shutdown events, the Reactor Coolant System (RCS) sensible and decay heat removal requirements are reduced as compared to heat removal requirements for MODE 1, 2, 3, or 4 events. Therefore, the shutdown containment heat removal requirements are bounded by analyses of MODES 1, 2, 3, and 4 events. A discussion of MODES 1, 2, 3, and 4 DBAs is provided in the Bases for LCO 3.6.6, "Passive Containment Cooling System (PCS) – Operating."

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

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LCO                      For postulated shutdown events, one passive containment cooling water flow path is required to provide the required containment heat removal capability (Ref. 1). To ensure that this requirement is met, two passive containment cooling water flow paths are provided. Therefore, in the event of an accident, at least one flow path operates, assuming the worst case single active failure occurs. A third PCS flow path is provided for protection against multiple failure scenarios modeled in the PRA. To ensure that these requirements are met, three PCS water flow paths must be OPERABLE.

The PCS includes a cooling water tank, valves, piping, instruments and controls to ensure an OPERABLE flow path capable of delivering water from the cooling water tank upon an actuation signal.

The PCS cooling water storage tank ensures that an adequate supply of water is available to cool and depressurize the containment in the event of a loss of decay heat removal. To be considered OPERABLE, the PCS cooling water storage

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BASES

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LCO  
(continued) tank must meet the water volume and temperature limits established in the SRs. To be considered OPERABLE, the air flow path from the shield building annulus inlet to the exit must be unobstructed, with unobstructed upper annulus safety-related drains providing a path for containment cooling water runoff to preclude blockage of the air flow path.

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APPLICABILITY OPERABILITY of the PCS is required in either MODE 5 or 6 with the calculated reactor decay heat greater than 9 MWt for heat removal in the event of a loss of nonsafety decay heat removal capabilities.

With the decay heat less than 9 MWt, the decay heat can be easily removed from containment with air cooling alone. Confirmation of decay heat levels may be determined consistent with the assumptions and analysis basis of ANS 1979 plus 2 sigma or via an energy balance of the reactor coolant system.

The PCS requirements in MODES 1, 2, 3, and 4 are specified in LCO 3.6.6, Passive Containment Cooling System (PCS) – Operating.

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ACTIONS

A.1

With one passive containment cooling water flow path inoperable, the affected flow path must be restored within 7 days. In this degraded condition, the remaining flow paths are capable of providing greater than 100% of the heat removal needs after an accident, even considering the worst single failure. The 7 day Completion Time was chosen in light of the remaining heat removal capability and the low probability of a DBA occurring during this period.

B.1

With two passive containment cooling water flow paths inoperable, at least one affected flow path must be restored to OPERABLE status within 72 hours. In this degraded condition, the remaining flow path is capable of providing greater than 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen in light of the remaining heat removal capability and the low probability of an event occurring during this period.

(continued)

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BASES

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ACTIONS  
(continued)

C.1

If the cooling water tank is inoperable, it must be restored to OPERABLE status within 8 hours. The tank may be declared inoperable due to low water volume or temperature out of limits. The 8 hour Completion Time is reasonable based on the remaining heat removal capability of the system and the availability of cooling water from alternate sources.

D.1.1, D.1.2, and D.2

Action must be initiated if any of the Required Actions and associated Completion Times for Condition A or B are not met, or if the LCO is not met for reasons other than Condition A or B. If in MODE 5 with the RCS pressure boundary open and/or pressurizer level < 20%, action must be initiated, immediately, to increase the RCS level to a pressurizer level  $\geq$  20% and to close the RCS so that the PRHR HX operation is available. If in MODE 6, action must be initiated, immediately, to increase the refueling cavity water level  $\geq$  23 feet above the top of the reactor vessel flange. In both cases, the time to RCS boiling is maximized by maximizing the RCS inventory and maintaining RCS temperature as low as practical. Additionally, action to suspend positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

These Actions place the plant in a condition which maximize the time to actuation of the Passive Containment Cooling System, thus providing time for repairs or application of alternative cooling capabilities.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.7.1

The LCO 3.6.6 Surveillance Requirements (SR 3.6.6.1 through 3.6.6.6) are applicable. The Frequencies associated with each specified SR are applicable. Refer to the corresponding Bases for LCO 3.6.6 for a discussion of each SR.

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REFERENCES

1. Section 6.2, "Containment Systems."
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## B 3.6 CONTAINMENT SYSTEMS

### B 3.6.8 Containment Penetrations

#### BASES

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#### BACKGROUND

Containment closure capability is required during shutdown operations when there is fuel inside containment. Containment closure is required to maintain within containment the cooling water inventory. Due to the large volume of the IRWST and the reduced sensible heat during shutdown, the loss of some of the water inventory can be accepted. Further, accident analyses have shown that containment closure capability is not required to meet offsite dose requirements. Therefore, containment does not need to be leak tight as required for MODES 1 through 4.

In MODES 5 and 6, the LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no requirement for containment leak tightness, compliance with the Appendix J leakage criteria and tests are not required.

In Modes 5 and 6, there is no potential for steam release into the containment immediately following an accident. Pressurization of the containment could only occur after heatup of the IRWST due to PRHR HX operation (MODE 5 with RCS intact) or after heatup of the RCS with direct venting to the containment (MODE 5 with reduced RCS inventory or MODE 6 with the refueling cavity not fully flooded) or after heatup of the RCS and refueling cavity (MODE 6 with refueling cavity fully flooded). The time from loss of normal cooling until steam release to the containment for four representative sets of plant conditions is shown in Figure B 3.6.8-1 as a function of time after shutdown. Because local manual action may be required to achieve containment closure it is assumed that the containment hatches, air locks and penetrations must be closed prior to steaming into containment.

Figure B 3.6.8-1 provides allowable closure times for four representative sets of plant conditions. The time to steaming is dependent on various plant parameters (RCS temperature, IRWST temperature, etc.) and plant configuration (RCS Pressure Boundary Intact, RCS Open, etc.). Therefore, the actual representation of the time to

(continued)

## BASES

BACKGROUND  
(continued)

steaming may be different than that provided in Figure B 3.6.8-1. In determining the minimum time to steaming, conservative assumptions regarding core decay heat, RCS configuration, and initial RCS inventory are used to minimize the calculated time to steaming.

As presented in Tables 54-1 and 54-4 of Reference 2, the most risk significant events during shutdown are events that lead to a loss of RNS cooling. Of these, the limiting events that lead to steaming to containment are the loss of shutdown cooling events, specifically:

- Loss of decay heat removal during drained conditions due to a failure of component cooling water or service water system;
- Loss-of-offsite power during drained conditions; and
- Loss of decay heat removal during drained conditions due to failure of the normal residual heat removal system.

These events are further discussed in Section 19.59.5 of Reference 1. Time to steaming is dependent on the postulated RCS configuration (intact versus open), and is based on the response of the plant considering features such as the operation of the 4th stage ADS valves if necessary, status of the upper internals, status of refueling cavity, etc. Conservative assumptions regarding these features are made in the determination of the minimum time to steaming.

The risk of overdraining the RCS has been significantly reduced in the AP1000 due to the automatic protection features associated with the hot leg level instruments which isolate letdown on low hot leg water level. Since a loss of RNS cooling does not occur for this event it is no longer a significant contributor to core damage, as shown in Table 54-4 of Reference 2.

The assumptions used in determining the required closure time for the various containment openings should be conservative, and should be consistent with the plant operating procedures, staffing levels, and status of the containment openings. The evaluation should consider the ability to close the containment for the limiting loss of shutdown cooling event, and considering the possibility of a station blackout.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. If closed, the equipment hatch must be held in place by at least [four] bolts. Good engineering practice dictates that bolts required by this LCO be approximately equally spaced. Alternatively, if open, each equipment hatch can be installed using a dedicated set of hardware, tools and equipment. A self-contained power source is provided to drive each hoist while lowering the hatch into position. Large equipment and components may be moved through the hatches as long as they can be removed and the hatch closed prior to steaming into the containment.

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BASES

BACKGROUND  
(continued)

Reviewers Note: The design of the equipment hatch is such that the [four] bolts would only be needed to support the hatch in place and provide adequate strength to support the hatch dead weight and associated loads. The hatch is installed on the inside containment and is held in place against a matching flange surface with mating bolt pattern by the bolts. Once the dead weight is supported, any pressure (greater than atmospheric) within containment will serve to exert closure force on the hatch toward the mating flange surface serving to reduce stresses on bolts. Therefore the determination of the number of bolts is limited to the quantity required to support the hatch itself and not related to any potential containment pressure.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Temporary equipment connections (e.g., power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment.

Containment spare penetrations which also provide a part of the containment boundary provide for temporary support services (electrical, I&C, air, and water supplies) during MODES 5 and 6. Each penetration is flanged and normally closed. During periods of plant shutdown, temporary support systems may be routed through the penetrations; temporary equipment connections (e.g., power or communications cables) are permitted as long as they can be removed to allow containment closure prior to steaming into the containment. The spare penetrations must be closed or, if open, capable of closure prior to steaming to containment.

Containment penetrations, including purge system flow paths, that provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual

(continued)

BASES

BACKGROUND  
(continued)

isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary barrier for the containment penetrations. The equivalent isolation barrier must be capable of maintaining containment isolation at the containment design pressure of 59 psig (Ref. 1).

APPLICABLE  
SAFETY ANALYSES

For postulated shutdown events in MODES 5 and 6, RCS heat removal is provided by either passive residual heat removal (PRHR) or IRWST injection and containment sump recirculation. To support RCS heat removal, containment closure is required to limit the loss of the cooling water inventory from containment (Ref. 1).

Containment penetrations satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO limits the loss of cooling water inventory in containment to assure continued coolant inventory by limiting the potential escape paths for water released within containment. Penetrations closed in accordance with these requirements are not required to be leak tight.

The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed or capable of being closed prior to steaming into the containment. The equipment hatches may be open; however, the hatches shall be clear of obstructions such that capability to close the hatch within the indicated time period is maintained. The hardware, tools, equipment and power sources necessary to install the hatches shall be available when the hatch is open. Both doors in each containment air lock may be open; however, the air locks shall be clear of obstructions such that the capability to close at least one door within the indicated time period is maintained. Alternatively, one door in an air lock may be closed. Containment spare penetrations may be open; however, the penetrations shall be capable of being closed within the indicated time period. Direct access penetrations shall be closed by at least one manual or automatic isolation valve, blind flange or equivalent, or capable of being closed by at least one valve actuated by a containment isolation signal. If direct access penetrations

(continued)

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**BASES**

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LCO  
(continued) are open, OPERABILITY of the containment isolation instrumentation is required for the open penetrations by LCO 3.3.2, Function 3.a, Containment Isolation, Manual Initiation. An OPERABLE Containment Isolation Function includes LCO 3.3.2, Function 19.b, Containment Air Filtration System Isolation, Containment Isolation. Figure B 3.6.8-1 provides the acceptable required closure times for various representative modes and conditions.

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APPLICABILITY The containment penetration requirements are applicable during conditions for which the primary safety related core cooling and boration capabilities are provided by IRWST or injection or PRHR – MODES 5 and 6. The capability to close containment is required to ensure that the cooling water inventory is not lost in the event of an accident.

In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1.

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**ACTIONS**A.1

If the containment equipment hatches, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the containment isolation function not capable of actuation when automatic isolation valves are open, the penetration(s) must be restored to the required status within 1 hour.

B.1.1, B.1.2, and B.2

If Required Action A.1 is not completed within 1 hour or the LCO is not met for reasons other than Condition A, action must be taken to minimize the probability and consequences of an accident.

In MODE 5, action must be initiated, immediately, to be in MODE 5 with a pressurizer level  $\geq 20\%$  and to close the RCS so that the PRHR HX operation is available. In MODE 6, action must be initiated, immediately, to be in MODE 6 with the refueling cavity water level  $\geq 23$  feet above the top of the reactor vessel flange. The time to RCS steaming to containment is maximized by maximizing RCS inventory, and allowing PRHR HX operation. Additionally, action to suspend

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### Time Permitted for Containment Closure

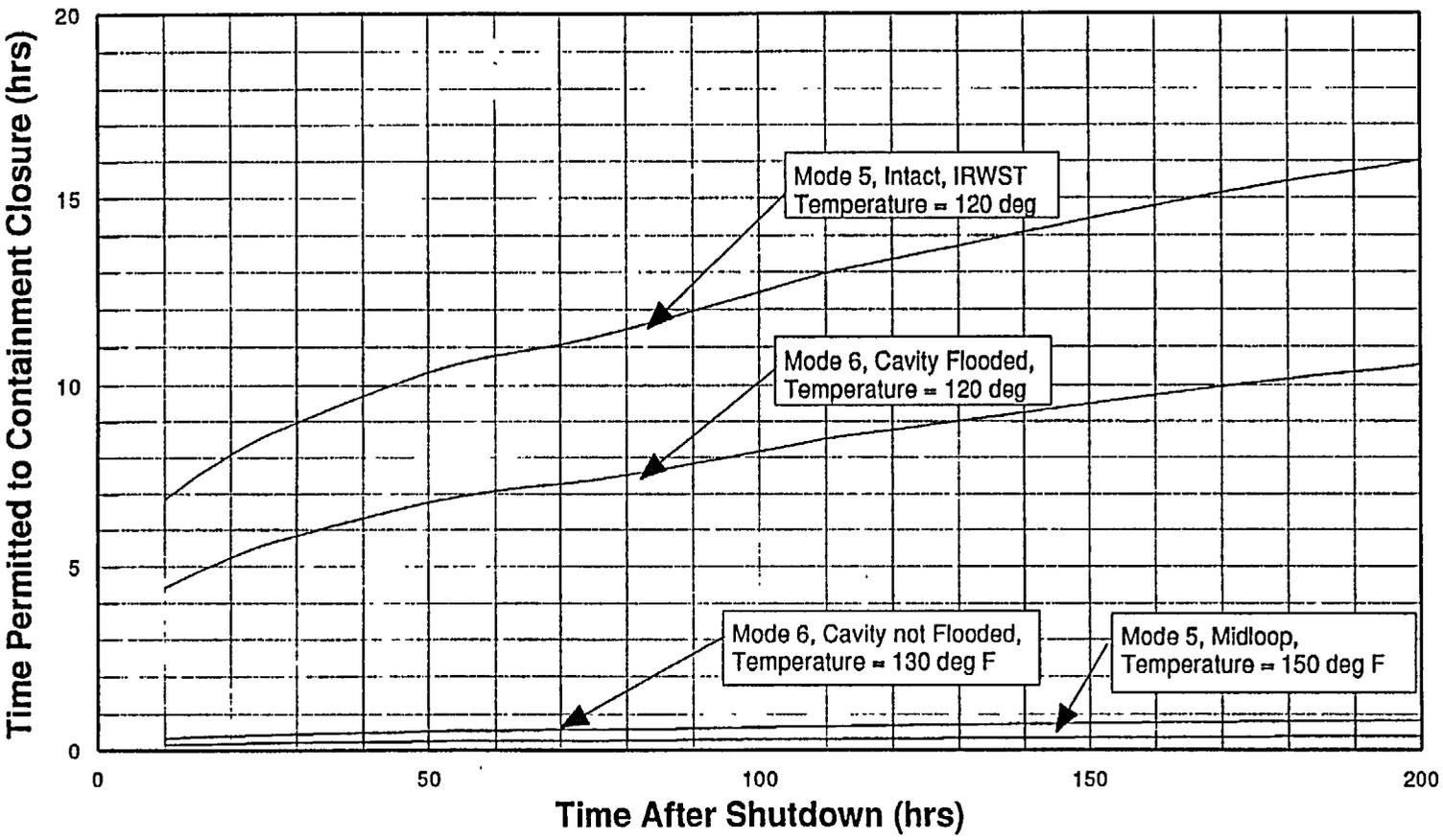


Figure B 3.6.8-1 (page 1 of 1)  
Time Prior to Coolant Inventory Boiling

BASES

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## ACTIONS

B.1 and B.2 (continued)

positive reactivity additions is required to ensure that the shutdown margin is maintained. Sources of positive reactivity addition include boron dilution, withdrawal of reactivity control assemblies, and excessive cooling of the RCS.

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SURVEILLANCE  
REQUIREMENTSSR 3.6.8.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal. Open containment spare penetrations shall be verified capable of being closed prior to steaming to containment by removal of obstructions and installation of the flange or by other closure means which will limit loss of the cooling water inventory from containment.

The Surveillance is performed every 7 days. The Surveillance interval is selected to ensure that the required penetration status is maintained during shutdown inspections, testing, and maintenance.

SR 3.6.8.2

Each of the two equipment hatches is provided with a set of hardware, tools, equipment, and self-contained power source for moving the hatch from its storage location and installing it in the opening. The required set of hardware and tools shall be visually inspected to ensure that they can perform the required functions. The equipment and power source shall be inspected and/or operated as necessary to verify that the hatch can be installed. The power source shall be verified as containing sufficient energy to install the hatch from the storage location.

The 7 day Frequency is adequate considering that the hardware, tools, equipment, and power sources are dedicated to the associated equipment hatch and not used for any other functions.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTSSR 3.6.8.2 (continued)

The SR is modified by a Note which only requires that the surveillance be met for an open equipment hatch. If the equipment hatch is installed in position, then the availability of the means to install the hatch is not required.

SR 3.6.8.3

This Surveillance demonstrates that at least one valve in each open penetration actuates to its isolation position on manual initiation or on an actual or simulated containment isolation signal. The 24 month Frequency maintains consistency with other similar valve testing requirements. The OPERABILITY requirements for the Containment Isolation function are specified in LCO 3.3.2.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

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REFERENCES

1. DCD Chapter 19.
  2. AP1000 Probabilistic Risk Assessment.
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B 3.6 CONTAINMENT

B 3.6.9 pH Adjustment

BASES

BACKGROUND

The Passive Core Cooling System (PXS) includes two pH adjustment baskets which provide adjustment of the pH of the water in the containment following an accident where the containment floods.

Following an accident with a large release of radioactivity, the containment pH is automatically adjusted to greater than or equal to 7.0, to enhance iodine retention in the containment water. Chemical addition is necessary to counter the affects of the boric acid contained in the safety injection supplies and acids produced in the post-LOCA environment (nitric acid from the irradiation of water and air and hydrochloric acid from irradiation and pyrolysis of electric cable insulation). The desired pH values significantly reduce formation of elemental iodine in the containment water, which reduces the production of organic iodine and the total airborne iodine in the containment. This pH adjustment is also provided to prevent stress corrosion cracking of safety related containment components during long-term cooling.

Dodecahydrate trisodium phosphate (TSP) contained in baskets provides a passive means of pH control for such accidents. The baskets are made of stainless steel with a mesh front that readily permits contact with water. These baskets are located inside containment at an elevation that is below the minimum floodup level. The baskets are placed at least a foot above the floor to reduce the chance that water spills will dissolve the TSP. Natural recirculation of water inside the containment, following a LOCA, is driven by the core decay heat and provides mixing to achieve a uniform pH. The dodecahydrate form of TSP ( $\text{Na}_3\text{PO}_4 \cdot 12\text{H}_2\text{O}$ ) is initially loaded into the baskets because it is hydrated and will undergo less physical and chemical change than would anhydrous TSP as a result of the humidity inside containment. (Refs. 1 and 2)

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(continued)

BASES (continued)

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APPLICABLE SAFETY ANALYSES In the event of a Design Basis Accident (DBA), iodine may be released from the fuel to containment. To limit this iodine release from containment, the pH of the water in the containment sump is adjusted by the addition of TSP. Adjusting the sump water to neutral or alkaline pH ( $\text{pH} \geq 7.0$ ) will augment the retention of the iodine, and thus reduce the iodine available to leak to the environment.

pH adjustment satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO The requirement to maintain the pH adjustment baskets with  $\geq [560 \text{ ft}^3]$  of TSP assures that for DBA releases of iodine into containment, the pH of the containment sump will be adjusted to enhance the retention of the iodine.

A required volume is specified instead of mass because it is not feasible to weigh the TSP in the containment. The minimum required volume is based on the manufactured density of TSP. This is conservative because the density of TSP may increase after installation due to compaction.

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APPLICABILITY In MODES 1, 2, 3, and 4 a DBA could cause release of radioactive iodine to containment requiring pH adjustment. The pH adjustment baskets assist in reducing the airborne iodine fission product inventory available for release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, pH adjustment is not required to be OPERABLE in MODES 5 and 6.

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ACTIONS A.1

If the TSP volume in the baskets is not within limits, the iodine retention may be less than that assumed in the accident analysis for the limiting DBA. Due to the very low probability that the volume of TSP may change, the variations are expected to be minor such that the required

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(continued)

BASES

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ACTIONS

A.1 (continued)

capability is substantially available. The 72 hour Completion Time for restoration to within limits is consistent with times applied to minor degradations of ECCS parameters.

B.1 and B.2

If the Required Actions and associated Completion Times are 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 at least MODE 3 within 6 hours and to MODE 5 within 84 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.9.1

The minimum amount of TSP is [560] ft<sup>3</sup>. A volume is specified since it is not feasible to weigh the TSP contained in the pH adjustment baskets. This volume is based on providing sufficient TSP to buffer the post accident containment water to a minimum pH of 7.0. Additionally, the TSP volume is based on treating the maximum volume of post accident water ([918,600] gallons) containing the maximum amount of boron ([3009] ppm) as well as other sources of acid. The minimum required mass of TSP is [27,540] pounds.

The minimum required volume of TSP is based on this minimum required mass of TSP, the minimum density of TSP plus margin to account for degradation of TSP during plant operation. The minimum TSP density is based on the manufactured density, since the density may increase and the volume decrease, during plant operation, due to agglomeration from humidity inside the containment. The minimum required TSP volume also has about 10% margin to account for degradation of TSP during plant operation.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.9.1 (continued)

The periodic verification is required every 24 months, since access to the TSP baskets is only feasible during outages, and normal fuel cycles are scheduled for 24 months. Operating experience has shown this Surveillance Frequency acceptable due to the margin in the volume of TSP placed in the containment building.

SR 3.6.9.2

Testing must be performed to ensure the solubility and buffering ability of the TSP after exposure to the containment environment. A representative sample of [2.36] grams of TSP from one of the baskets in containment is submerged in  $\geq 1$  liter of water at a boron concentration of [3009] ppm and at the standard temperature of  $25 \pm 5^\circ\text{C}$ . Without agitation, the solution pH should be raised to  $\geq 7.0$  within 4 hours.

The minimum required amount of TSP is sufficient to buffer the maximum amount of boron [3009] ppm, the maximum amount of other acids, and the maximum amount of water [918,600] gallons that can exist in the containment following an accident and achieve a minimum pH of 7.0.

Agitation of the test solution is prohibited, since an adequate standard for the agitation intensity cannot be specified. The test time of 4 hours is necessary to allow time for the dissolved TSP to naturally diffuse through the sample solution. In the post LOCA sump area, rapid mixing would occur due to liquid flow, significantly decreasing the actual amount of time before the required pH is achieved. This would ensure compliance with the Standard Review Plan requirement of a pH  $\geq 7.0$  by the onset of recirculation after a LOCA.

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REFERENCES

1. Section 6.3.2.1.4, "Containment pH Control."
  2. Section 6.3.2.2.4, "pH Adjustment Baskets."
  3. Section 15.6.5.3.1, "Identification of Cause and Accident Description."
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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Six MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in Reference 1. The MSSV design includes staggered setpoints, as shown in Table 3.7.1-2 of the specification, so that only the needed valves actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open the valves following a turbine-reactor trip.

APPLICABLE  
SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to  $\leq 110\%$  of design pressure for any anticipated operating occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, are those characterized as decreased heat removal events, which are presented in Section 15.2 (Ref. 3). Of these, the full power turbine trip without turbine bypass is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS

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**BASES**

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**APPLICABLE  
SAFETY ANALYSES  
(continued)**

or the Main Steam System. The reactor is tripped on high pressurizer pressure. Pressurizer safety valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems. The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

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

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**LCO**

The accident analysis requires six MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% of RTP. A MSSV will be considered inoperable if it fails to open in the event of a pressure excursion in excess of the setpoint. The LCO requires that six MSSVs be OPERABLE in compliance with Reference 2. Operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 of the specification and Required Action A.1.

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

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

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

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(continued)

BASES (continued)

APPLICABILITY In MODE 1 at or above 67% RTP, six MSSVs per steam generator are required to be OPERABLE. Below 67% RTP in MODE 1, 2, 3, or 4 (without the normal residual heat removal system in service), only four MSSVs per steam generator are required to be OPERABLE.

In MODES 4 (with the normal residual heat removal system in service) and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. There is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

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

A.1 and A.2

With one or more MSSVs inoperable, reduce power so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all six MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if two MSSVs are inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 33%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 33%. This is accomplished by reducing THERMAL POWER by at least 33%, which conservatively limits the energy transfer to both steam generators to approximately 67% of total capacity, consistent with the relief capacity of the most limiting steam generator.

For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:

$$FRC = \frac{A}{B}$$

(continued)

BASES

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ACTIONS            A.1 and A.2 (continued)

where:

A = the relief capacity of the MSSV; and

B = the total relief capacity of all the MSSVs of the steam generator

The FRC is the relief capacity necessary to address operation with reduced THERMAL POWER.

The reduced THERMAL POWER levels in the LCO, prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER is determined as follows:

$$RP = \left[ 1 - \frac{(N_1 \times FRC_1 + N_2 \times FRC_2 + N_3 \times FRC_3 + N_4 \times FRC_4 + N_5 \times FRC_5 + N_6 \times FRC_6)}{B} \right] \times 100\%$$

where:

RP = Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP;

$N_1, N_2, N_3, N_4, N_5, N_6$  represent the status of MSSV 1, 2, 3, 4, 5, and 6, respectively,

= 0 if the MSSV is OPERABLE,  
= 1 if the MSSV is inoperable;

$FRC_1, FRC_2, FRC_3, FRC_4, FRC_5, FRC_6$  = the relief capacity of MSSV 1, 2, 3, 4, 5, and 6 respectively, as defined above.

B.1 and B.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have less than two MSSVs OPERABLE, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4, with RCS cooling provided by the RNS, within 24 hours. The allowed Completion Times are reasonable,

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(continued)

BASES

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ACTIONS

B.1 and B.2 (continued)

based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ASME OM Code (Ref. 5). According to Reference 5, the following tests are required:

- a. Visual examination;
- b. Seat tightness determination;
- c. Set pressure determination (lift setting);
- d. Compliance with owner's seat tightness criteria;  
and
- e. Verification of the balancing device integrity on  
balanced valves.

The ANSI/ASME standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a  $\pm 3\%$  setpoint tolerance for OPERABILITY; however, the valves are reset to  $\pm 1\%$  during the Surveillance to allow for drift. The lift settings, according to Table 3.7.1-2, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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

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BASES (continued)

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REFERENCES

1. Chapter 10, "Steam and Power Conversion Systems Description."
  2. ASME Boiler and Pressure Vessel Code, Section III, Article NC-7000, "Overpressure Protection," Class 2 Components.
  3. Section 15.2, "Decreased Heat Removal by Secondary System."
  4. ASME Boiler and Pressure Vessel Code, Section XI, Article IV-3500, "Inservice Test: Category C Valves."
  5. ASME OM Code-1995 and Addenda through the 1996 Addenda, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices in Light Water Reactor Power Plants."
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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

Each main steamline has one safety related MSIV to isolate steam flow from the secondary side of the steam generators following a high energy line break. MSIV closure terminates flow from the unaffected (intact) steam generator.

One MSIV is located in each main steam line outside containment. The MSIVs are downstream from the main steam safety valves (MSSVs). Downstream from the MSIVs, main steam enters the high pressure turbine through four stop valves and four governing control valves. Closing the MSIVs isolates each steam generator from the other and isolates the turbine bypass system, and other steam supplies from the steam generator.

The MSIVs, turbine stop and control valves, turbine bypass valves, and moisture separator reheat supply steam control valves close on a main steam isolation signal generated by either low steam line pressure, high containment pressure, Low T<sub>cold</sub>, or high negative steam pressure rate. The MSIVs fail closed on loss of control air or actuation signal from either of two 1E power divisions.

Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.

A description of the MSIVs is found in the Section 10.3 (Ref. 1). Descriptions for the turbine bypass valves, and moisture separator reheat supply steam control valve are found in the Section 10.4 (Ref. 6).

APPLICABLE  
SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for the large steam line break (SLB) inside containment, discussed in the Section 6.2 (Ref. 2). It is also affected by the accident analysis of the SLB events presented in the Section 15.1 (Ref. 3). The design precludes the blowdown of more than one steam generator, assuming a single active component failure (e.g., the failure of one MSIV to close on demand).

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

Design basis events of concern for containment analysis are SLB inside containment with the failure of the associated MSIV to close, or a main feedline break with the associated failure of a feedline isolation or control valve to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers, downstream from the other MSIV, contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Core Makeup Tanks (CMTs).

The accident analysis compares several different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes consideration of scenarios with offsite power available, and with a loss of offsite power. With offsite power available, the reactor coolant pumps continue to circulate coolant for a longer period through the steam generators, maximizing the Reactor Coolant System cooldown. The reactor protection system includes a safety related signal that initiates the coastdown of the reactor coolant pumps early in the large SLB transient. Therefore, there is very little difference in the predicted departure from nucleate boiling ratio between cases with and without offsite power. Significant single failures considered include failure of an MSIV to close.

The non-safety related turbine stop or control valves, in combination with the turbine bypass, and moisture separator reheat supply steam control valves, are assumed as a backup to isolate the steam flow path given a single failure of an MSIV. The safety analyses do not differentiate between the availability of the turbine stop valve or its series control valve. Either the turbine stop valves or its associated turbine control valve are required by this LCO to be OPERABLE. These valves, along with the turbine bypass, and

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(continued)

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

moisture separator reheat supply steam control valves are considered as alternate downstream valves.

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

- a. High energy line break inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from both steam generators until the unaffected loop MSIV closes. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIV in the unaffected loop. Closure of the MSIV isolates the break from the unaffected steam generator.
- b. A break outside of containment, and upstream or downstream from the MSIVs, is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs or alternate downstream valves isolates the break, and limits the blowdown to a single steam generator.
- c. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator to minimize radiological releases.
- d. The MSIVs are also utilized during other events such as a feedwater line break; however, these events are less limiting so far as MSIV OPERABILITY is concerned.

The MSIVs and the alternate downstream valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Following an SLB and main steam isolation signal, the analyses assume continued steam loss through the steamline condensate drain lines, turbine gland seal system, and the main steam to auxiliary steam header which supplies the auxiliary steam line to the deaerator. Since these valves are not assumed for steam isolation, they do not satisfy the 10 CFR 50.36(c)(2)(ii) criteria.

(continued)

BASES (continued)

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## LCO

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

This LCO requires that four turbine stop valves or their associated turbine control valve, six turbine bypass valves, and four moisture separator reheat supply steam control valve be OPERABLE. A valve is considered OPERABLE when its isolation time is within the safety analysis isolation time limit of 5 seconds and it closes on an MSIV actuation signal. The turbine bypass valves are alternatively considered OPERABLE when closed and administratively maintained closed with automatic actuation blocked as appropriate.

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits or the NRC staff approved licensing basis.

This LCO provides assurance that the design and performance of the alternate downstream valves are compatible with the accident conditions for which they are called upon to function (Ref. 5).

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APPLICABILITY

The MSIVs, turbine stop or associated turbine control valves, turbine bypass valves, and moisture separator reheat supply steam control valves must be OPERABLE in MODE 1 and MODES 2, 3, and 4, except when steam flow is isolated when there is significant mass and energy in the RCS and steam generators. Therefore, these valves must be OPERABLE or closed. When these valves are closed, they are already performing their required function.

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

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(continued)

BASES (continued)

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ACTIONS

A.1

With one MSIV inoperable in MODE 1, action must be taken to restore OPERABLE status within 8 hours. Some repairs to the valves can be made with the plant hot. The 8 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of these valves. With a single MSIV inoperable, the safety function, isolation of the steam flow path, is provided by the OPERABLE alternate downstream valves, but cannot accommodate a single failure. The assumptions and criteria of the accident analyses are preserved by the ability to automatically isolate the steam flow path.

The 8 hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides a positive means for containment isolation.

B.1

With any number of the turbine stop valves and the associated turbine control valve, turbine bypass, or moisture separator reheat supply steam control valves inoperable in MODE 1, action must be taken to restore OPERABLE status within 72 hours. Some repairs to the valves can be made with the plant hot. The 72 hour Completion Time is reasonable considering the low probability of an accident occurring during this time period that would require a closure of these valves. With the backup isolation valves inoperable, the safety function, isolation of the steam flow path, is provided by the remaining OPERABLE valves, but cannot accommodate a single failure. The assumptions and criteria of the accident analyses are preserved by the ability to automatically isolate the steam flow path.

C.1

With two MSIVs inoperable in MODE 1 or one MSIV and an alternate downstream valve inoperable or if the valves cannot be restored to OPERABLE status in accordance with Required Action A.1 or B.1, the unit must be placed in a MODE in which the LCO does not apply. To achieve this

(continued)

BASES

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ACTIONS

C.1 (continued)

status, the unit must be placed in MODE 2 within 6 hours and Condition D would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner and without challenging unit systems.

D.1 and D.2

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

Since the MSIVs are required to be OPERABLE in MODES 2, 3, and 4, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

The 8 hour Completion Time is consistent with that allowed in Condition A, and conservative considering the reduced energy in the steam generators in MODES 2, 3, and 4.

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

E.1 and E.2

If the MSIVs cannot be restored to OPERABLE status or closed within the associated Completion Times of Condition D, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 with normal residual heat removal system in service within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is  $\leq 5.0$  seconds, on an actual or simulated actuation signal. The MSIV isolation time is assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 7), requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

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

SR 3.7.2.2

This SR verifies that the turbine stop, turbine control, turbine bypass, and moisture separator reheat supply steam control valves' closure time is  $\leq 5.0$  seconds, on an actual or simulated actuation signal. These alternate downstream isolation valves must meet the MSIV isolation time assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The alternate downstream valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. As the alternate downstream valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 7), requirements during operation in MODE 1 or 2.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.2.2 (continued)

The Frequency is in accordance with the Inservice Testing Program.

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

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REFERENCES

1. Section 10.3, "Main Steam System."
  2. Section 6.2.1, "Containment Functional Design."
  3. Section 15.1, "Increase in Heat Removal by Secondary System."
  4. Section 10.2, "Turbine Generator."
  5. NUREG-138, Issue 1, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director NRR to NRR Staff."
  6. Section 10.4, "Other Features of Steam and Power Conversion Systems."
  7. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Isolation and Control Valves (MFIVs and MFCVs)

BASES

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BACKGROUND

The MFIVs isolate main feedwater (MFW) flow to the secondary side of the steam generators following a high energy line break. The safety related function of the MFCVs is to provide the second isolation of MFW flow to the secondary side of the steam generators following a high energy line break. Closure of the MFIVs or MFCVs terminates flow to the steam generators, terminating the event for feedwater line breaks occurring upstream of the MFIVs or MFCVs. The consequences of events occurring in the main steam lines or in the MFW lines downstream from the MFIVs will be mitigated by their closure. Closure of the MFIVs or MFCVs, effectively terminates the addition of main feedwater to an affected steam generator, limiting the mass and energy release for steam or feedwater line breaks inside containment, and reducing the cooldown effects for steam line breaks (SLBs).

The MFIVs or MFCVs isolate the nonsafety related portions from the safety related portions of the system. In the event of a secondary side pipe rupture inside containment, the valves limit the quantity of high energy fluid that enters containment through the break, and provide a pressure boundary for the controlled addition of startup feedwater (SFW) to the intact loops of the steam generator.

One MFIV and one MFCV are located on each MFW line, outside but close to containment. The MFIVs and MFCVs are located in the MFW line and are independent of the delivery of the MFW or SFW via the SFW line which is separately connected and isolated from the steam generator. This configuration permits MFW or SFW to be supplied to the steam generators following MFIV or MFCV closure. The piping volume from these valves to the steam generators must be accounted for in calculating mass and energy releases following either an SLB or FWLB.

The MFIVs and MFCVs close on receipt of engineered safeguards feedwater isolation signal generated from any of the following conditions:

(continued)

BASES

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BACKGROUND  
(continued)

- Automatic or manual safeguards actuation "S" signal
- High steam generator level
- Low-2 T<sub>avg</sub> signal coincident with reactor trip (P-4)
- Manual actuation

Additionally, the MFIVs close automatically on a Low-1 T<sub>avg</sub> coincident with reactor trip (P-4). Each valve may be actuated manually. In addition to the MFIVs and the MFCVs, a check valve is available outside containment to isolate the feedwater line penetrating containment. In the event of feedwater line depressurization due to pump trip on line break, the check valve provides rapid backup isolation of the steam generators limiting the inventory loss. A description of the MFIVs and MFCVs is found in Reference 1.

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APPLICABLE  
SAFETY ANALYSES

The design basis of the MFIVs and MFCVs is established by the analyses for the large SLB. It is also influenced by the accident analysis for the large Feedwater Line Break (FWLB). Closure of the MFIVs (or MFCVs) may also be relied on to mitigate an SLB for core response analysis and excess feedwater event upon the receipt of a steam generator water level - High 2 signal.

Failure of an MFIV (or MFCV), to close following an SLB or FWLB, can result in additional mass and energy being delivered to the steam generators, contributing to cooldown. This failure also results in additional mass and energy releases following an SLB or FWLB event.

The MFIVs and MFCVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO ensures that the MFIVs and the MFCVs will isolate the main feedwater system.

This LCO requires that the one isolation valve and one control valve on each feedwater line be OPERABLE. These valves are considered OPERABLE when their isolation times are within limits and they close on an isolation actuation signal.

(continued)

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BASES

LCO  
(continued) Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. A main feedwater isolation signal on high steam generator level is relied on to terminate an excess feedwater flow event, and therefore failure to meet the LCO may result in the introduction of water into the main steam lines.

APPLICABILITY The MFIVs and MFCVs must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and the steam generators. This ensures that, in the event of a high energy line break, a single failure cannot result in the blowdown of more than one steam generator. In MODE 1, 2, 3, or 4, these valves are required to be OPERABLE to limit the amount of available fluid that could be added to the containment in the case of a secondary system pipe break inside containment. When the valves are closed and deactivated or isolated by a closed manual valve, they are already performing their safety function.

In MODES 5 and 6 steam generator energy is low. Therefore, the MFIVs and the MFCVs are normally closed since MFW is not required.

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

A.1, A.2, B.1, and B.2

With one or two MFIVs, or one or two MFCVs inoperable, close or isolate inoperable affected flow path in 72 hours. When these flow paths are isolated, they are performing their required safety function.

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

For inoperable MFIVs and MFCVs valves that cannot be restored to OPERABLE status within the specified Completion Time but are closed or isolated, the flow paths must be

(continued)

BASES

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## ACTIONS

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

verified on a periodic basis to be closed or isolated. This is necessary to ensure that the assumptions in the safety analyses remain valid. The 7 day Completion Time is reasonable based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1

With two inoperable valves in the same flow path there may be no redundant system to operate automatically and perform the required safety function. Under these conditions, one valve in the affected flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. This action returns the system to the situation in which at least one valve in the affected flow path is performing the required safety function. The 8 hour Completion Time is a reasonable amount of time to complete the actions required to close the MFIV, or MFCV, which includes performing a controlled plant shutdown. The Completion Time is reasonable based on operating experience to reach MODE 2 with the MFIV or MFCV closed, from full-power conditions in an orderly manner and without challenging plant systems.

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

If the MFIVs and MFCVs cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time, the unit must be placed in a mode in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, in MODE 4 with the normal residual heat removal system in service within 24 hours, and the affected flow path isolated within 36 hours or in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFIV and MFRV is  $\leq 5.0$  seconds, on an actual or simulated actuation signal. The MFIV and MFRV isolation times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. These valves should not be tested at power, since even a part stroke exercise increases the risk of a valve closure when the unit is generating power. This is consistent with the ASME Code, Section XI (Ref. 2), quarterly stroke requirements during operation in MODE 1 or 2.

The Frequency is in accordance with the Inservice Testing Program.

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

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REFERENCES

1. Section 10.4.7, "Condensate and Feedwater System."
  2. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.7 PLANT SYSTEMS

B 3.7.4 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube leakage from the Reactor Coolant System (RCS). Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant. While fission products present in the primary coolant, as well as activated corrosion products, enter the secondary coolant system due to the primary to secondary leakage, only the iodines are of a significant concern relative to airborne release of activity in the event of an accident or abnormal occurrence (radioactive noble gases that enter the secondary side are not retained in the coolant but are released to the environment via the condenser air removal system throughout normal operation).

The limit on secondary coolant radioactive iodines minimizes releases to the environment due to anticipated operational occurrences or postulated accidents.

APPLICABLE  
SAFETY ANALYSES

The accident analysis of the main steam line break (SLB) as discussed in Chapter 15 (Ref. 1) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1  $\mu\text{Ci/gm}$  DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of a postulated SLB are within the acceptance criteria in SRP Section 15.0.1, and within the exposure guideline values of 10 CFR Part 50.34.

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

(continued)

BASES (continued)

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LCO As indicated in the Applicable Safety Analyses, the specific activity limit of the secondary coolant is required to be  $\leq 0.1 \mu\text{Ci/gm}$  DOSE EQUIVALENT I-131 to maintain the validity of the analyses reported in Chapter 15 (Ref. 1).

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

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

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

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ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.4.1

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

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REFERENCES

1. Chapter 15, "Accident Analyses."
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B 3.7 PLANT SYSTEMS

B 3.7.5 Spent Fuel Pool Water Level

BASES

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BACKGROUND

The minimum water level in the spent fuel pool meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are at their maximum capacity. The water also provides shielding during the movement of spent fuel, and a large capacity heat sink in the event the spent fuel pool cooling system is inoperable.

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

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APPLICABLE  
SAFETY ANALYSES

The minimum water level in the spent fuel pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.183 (Ref. 4). The design basis radiological consequences resulting from a postulated fuel handling accident are within the dose values provided in Section 15.7.4 (Ref. 3).

According to Reference 3 there is 23 ft of water between the damaged fuel bundle and the fuel pool surface during a fuel handling accident. In the case of a single bundle dropped and lying horizontally on top of the spent fuel racks, however, there may be < 23 ft of water above the top of the fuel bundle and the surface, indicated by the width of the bundle. This slight reduction in water depth does not adversely affect the margin of conservatism associated with the assumed pool scrubbing factor of 500 for elemental iodine.

In addition to mitigation of the effects of a fuel handling accident, the required minimum water level in the spent fuel pool provides a large capacity heat sink for spent fuel pool cooling in the event the spent fuel pool cooling system is inoperable.

The Spent Fuel Pool Water Level satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2)(ii).

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(continued)

BASES (continued)

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

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APPLICABILITY This LCO applies at all times since the loss of spent fuel pool cooling is not MODE dependent.

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ACTIONS LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. Spent fuel pool cooling requirements are independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. Spent fuel pool cooling requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

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

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

A.2

If the water level in the spent fuel pool is  $< 23$  ft, the heat capacity of the spent fuel pool will be less than that assumed in the event of a loss of spent fuel pool cooling. In this case, action must be initiated within 1 hour to restore the water level in the spent fuel pool to  $\geq 23$  ft above the top of the irradiated fuel assemblies. Initiation of this action requires that the action be continued until a water level of  $\geq 23$  ft is attained.

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BASES

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ACTIONS

A.2 (continued)

The Completion Time of 1 hour assures prompt action to compensate for a degraded condition.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.5.1

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

During refueling operations, the level in the spent fuel pool is in equilibrium with the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.4.1.

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REFERENCES

1. Section 9.1.2, "Spent Fuel Storage."
  2. Section 9.1.3, "Spent Fuel Pool Cooling System."
  3. Section 15.7.4, "Fuel Handling Accident."
  4. Regulatory Guide 1.183 Rev. 0, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."
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B 3.7 PLANT SYSTEMS

B 3.7.6 Main Control Room Emergency Habitability System

BASES

BACKGROUND

The Main Control Room Habitability System (VES) provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. The system is designed to operate following a Design Basis Accident (DBA) which requires protection from the release of radioactivity. In these events, the Nuclear Island Non-Radioactive Ventilation System (VBS) would continue to function if AC power is available. If AC power is lost or a High-2 main control room (MCR) radiation signal is received, the VES is actuated. The major functions of the VES are: 1) to provide forced ventilation to deliver an adequate supply of breathable air for the MCR occupants; 2) to provide forced ventilation to maintain the MCR at a 1/8 inch water gauge positive pressure with respect to the surrounding areas; and 3) to limit the temperature increase of the MCR equipment and facilities that must remain functional during an accident, via the heat absorption of passive heat sinks.

The VES consists of compressed air storage tanks, two air delivery flow paths, associated valves, piping, and instrumentation. The tanks contain enough breathable air to supply the required air flow to the MCR for at least 72 hours. The VES system is designed to maintain CO<sub>2</sub> concentration less than 0.5% for up to 11 MCR occupants.

Sufficient thermal mass exists in the surrounding concrete structure (including walls, ceiling and floors) to absorb the heat generated inside the MCR, which is initially at or below 75°F. Heat sources inside the MCR include operator workstations, emergency lighting and occupants. Sufficient insulation is provided surrounding the MCR pressure boundary to preserve the minimum required thermal capacity of the heat sink. The insulation also limits the heat gain from the adjoining areas following the loss of VBS cooling.

In the unlikely event that power to the VBS is unavailable for more than 72 hours, MCR envelope habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR envelope.

(continued)

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**BASES**

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**BACKGROUND**  
(continued)

The compressed air storage tanks are initially pressurized to 3400 psig. During operation of the VES, a self contained pressure regulating valve maintains a constant downstream pressure regardless of the upstream pressure. An orifice downstream of the regulating valve is used to control the air flow rate into the MCR. The MCR is maintained at a 1/8 inch water gauge positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas.

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**APPLICABLE**  
**SAFETY ANALYSES**

The compressed air storage tanks are sized such that the set of tanks has a combined capacity that provides at least 72 hours of VES operation.

Operation of the VES is automatically initiated by either of two safety related signals: 1) undervoltage to Class 1E battery charger, or 2) high-2 particulate or iodine radioactivity.

In the event of a loss of all AC power, the VES functions to provide ventilation, pressurization, and cooling of the MCR pressure boundary.

In the event of a high level of gaseous radioactivity outside of the MCR, the VBS continues to operate to provide pressurization and filtration functions. The MCR air supply downstream of the filtration units is monitored by a safety related radiation detector. Upon an undervoltage to Class 1E battery charger or high-2 particulate or iodine radioactivity setpoint, a safety related signal is generated to isolate the MCR from the VBS and to initiate air flow from the VES storage tanks. Isolation of the VBS consists of closing safety related valves in the supply and exhaust ducts that penetrate the MCR pressure boundary. VES air flow is initiated by a safety related signal which opens the isolation valves in the VES supply lines.

The VES functions to mitigate a DBA or transient that either assumes the failure of or challenges the integrity of the fission product barrier.

The VES satisfies the requirements of Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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(continued)

BASES (continued)

LCO

The VES limits the MCR temperature rise and maintains the MCR at a positive pressure relative to the surrounding environment.

Two air delivery flow paths are required to be OPERABLE to ensure that at least one is available, assuming a single failure.

The VES is considered OPERABLE when the individual components necessary to deliver a supply of breathable air to the MCR are OPERABLE. This includes components listed in SR 3.7.6.2 through 3.7.6.8. In addition, the MCR pressure boundary must be maintained, including the integrity of the walls, floors, ceilings, electrical and mechanical penetrations, and access doors.

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

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

APPLICABILITY

The VES is required to be OPERABLE in MODES 1, 2, 3, and 4 and during movement of irradiated fuel because of the potential for a fission product release following a DBA.

The VES is not required to be OPERABLE in MODES 5 and 6 when irradiated fuel is not being moved because accidents resulting in fission product release are not postulated.

ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering

LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

When a VES valve or damper is inoperable, action is required to restore the component to OPERABLE status. A Completion Time of 7 days is permitted to restore the valve or damper to OPERABLE status before action must be taken to reduce power. The Completion Time of 7 days is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining components can provide the required capability.

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(continued)

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BASES

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ACTIONS  
(continued)

B.1

When the main control room air temperature is outside the acceptable range during VBS operation, action is required to restore it to an acceptable range. A Completion Time of 24 hours is permitted based upon the availability of temperature indication in the MCR. It is judged to be a sufficient amount of time allotted to correct the deficiency in the nonsafety ventilation system before shutting down.

C.1

If the MCR pressure boundary is damaged or otherwise degraded, action is required to restore the integrity of the pressure boundary and restore it to OPERABLE status within 24 hours. A Completion Time of 24 hours is permitted based upon operating experience. It is judged to be a sufficient amount of time allotted to correct the deficiency in the pressure boundary.

D.1 and D.2

In MODE 1, 2, 3, or 4 if Conditions A, B, or C cannot be restored to OPERABLE status within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. This is done by entering MODE 3 within 6 hours and MODE 5 within 36 hours.

E.1

During movement of irradiated fuel assemblies, if the Required Action A.1, B.1, or C.1 cannot be completed within the required Completion Time, the movement of fuel must be suspended. Performance of Required Action E.1 shall not preclude completion of actions to establish a safe condition.

F.1, F.2, and F.3

If the VES is inoperable in MODE 1, 2, 3, or 4, the VES may not be capable of performing the intended function, and the plant must be brought to MODE 4, where the probability and consequences of an event are minimized, and the VES must be

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(continued)

BASES

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ACTIONS

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

restored to OPERABLE status within 36 hours. This is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 within 12 hours.

G.1

During movement of irradiated fuel assemblies with the VES inoperable, the Required Action is to immediately suspend activities that present a potential for releasing radioactivity that might enter the MCR. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.6.1

The MCR air temperature is checked at a frequency of 24 hours to verify that the VBS is performing as required to maintain the initial condition temperature assumed in the safety analysis, and to ensure that the MCR temperature will not exceed the required conditions after loss of VBS cooling. The surveillance limit of 75°F is the initial heat sink temperature assumed in the VES thermal analysis. The 24 hour Frequency is acceptable based on the availability of temperature indication in the MCR.

SR 3.7.6.2

Verification every 24 hours that compressed air storage tanks are pressurized to [ $\geq$  3400 psig] is sufficient to ensure that there will be an adequate supply of breathable air to maintain MCR habitability for a period of 72 hours. The Frequency of 24 hours is based on the availability of pressure indication in the MCR.

SR 3.7.6.3

VES air delivery isolation valves are required to be verified as OPERABLE. The Frequency required is in accordance with the Inservice Testing Program.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.7.6.4

VES air header isolation valves are required to be verified open at 31 day intervals. This SR is designed to ensure that the pathways for supplying breathable air to the MCR are available should loss of VBS occur. These valves should be closed only during required testing or maintenance of downstream components, or to preclude complete depressurization of the system should the VES isolation valves in the air delivery line open inadvertently or begin to leak.

SR 3.7.6.5

Verification that the air quality of the air storage tanks meets the requirements of Appendix C, Table C-1 of ASHRAE Standard 62 is required every 92 days. If air has not been added to the air storage tanks since the previous verification, verification may be accomplished by confirmation of the acceptability of the previous surveillance results along with examination of the documented record of air makeup. The purpose of ASHRAE Standard 62 states: "This standard specifies minimum ventilation rates and indoor air quality that will be acceptable to human occupants and are intended to minimize the potential for adverse health effects." Verification of the initial air quality (in combination with the other surveillances) ensures that breathable air is available for 11 MCR occupants for at least 72 hours.

SR 3.7.6.6

Verification that all VBS isolation valves are OPERABLE and will actuate upon demand is required every 24 months to ensure that the MCR can be isolated upon loss of VBS operation.

SR 3.7.6.7

Verification that each VES pressure relief isolation valve within the MCR pressure boundary is OPERABLE is required in accordance with the Inservice Testing Program. The SR is used in combination with SR 3.7.6.7 to ensure that adequate vent area is available to mitigate MCR overpressurization.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.7.6.8

Verification that the VES pressure relief damper is OPERABLE is required at 24 month intervals. The SR is used in combination with SR 3.7.6.6 to ensure that adequate vent area is available to mitigate MCR overpressurization.

SR 3.7.6.9

Verification of the OPERABILITY of the self-contained pressure regulating valve in each VES air delivery flow path is required in accordance with the Inservice Testing Program. This is done to ensure that a sufficient supply of air is provided as required, and that uncontrolled air flow into the MCR will not occur.

SR 3.7.6.10

Per Reference 1, a functional test is required to establish that one VES air delivery flow path, using the safety related compressed air storage tanks, pressurizes the MCR envelope to at least a positive 1/8 inch water gauge pressure relative to the surrounding spaces at the required air addition flow rate of  $65 \pm 5$  scfm (Ref. 3). The test need not last 72 hours, only long enough to demonstrate the ability to achieve the required differential pressure. The MCR envelope leakage rate must be within the design capacity of the VES to pressurize the MCR for 72 hours. One air delivery flow path is tested on an alternating basis. The system performance test demonstrates that the MCR pressurization assumed in dose analysis is maintained.

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REFERENCES

1. Section 6.4, "Main Control Room Habitability Systems."
2. Section 9.4.1, "Nuclear Island Non-Radioactive Ventilation System."
3. SECY-95-132, "Policy and Technical Issues Associated With The Regulatory Treatment of Non-Safety Systems (RTNSS) In Passive Plant Designs (SECY-94-084)," May 22, 1995.

## B 3.7 PLANT SYSTEMS

## B 3.7.7 Startup Feedwater Isolation and Control Valves

BASES

## BACKGROUND

The startup feedwater system supplies feedwater to the steam generators during plant startup, hot standby and cooldown, and in the event of main feedwater unavailability.

The startup feedwater system serves no safety related function and has no safety related design basis, except to isolate feedwater in the event of a feedwater, steam line break, a steam generator tube rupture or other secondary side event.

The startup feedwater system consists of a flow path to each of the steam generators. Each flow path consists of two series startup feedwater valves to provide feedwater control for low feedwater demand conditions. Feedwater can be supplied to the startup feedwater line via either the main or startup feedwater pumps. The feedwater is delivered directly to the SG independent of the main feedwater line. Each startup feedwater line contains one control valve and one isolation valve (Ref. 1).

APPLICABLE  
SAFETY ANALYSES

The basis for the requirement to isolate the startup feedwater system is established by the analysis for large Steam Line Break (SLB) inside containment. It is also based on the analysis for a large Feedline Break (FLB) and a steam generator tube rupture.

Failure to isolate the startup feedwater system following a SLB or FLB can lead to additional mass and energy being delivered to the steam generators, resulting in excessive cooldown and additional mass and energy release in containment. Failure to isolate the startup feedwater following a steam generator tube rupture may result in overfilling the steam generator.

Low  $T_{cold}$  or high steam generator level signals close the startup feedwater control and isolation valves and trips the startup feedwater pumps.

The startup feedwater isolation and control valves are components which actuate to mitigate a Design Basis Accident, and as such meet Criterion 3 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

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LCO This LCO ensures that the startup feedwater isolation and control valves will actuate on command, following a SLB, FLB or SGTR, and isolate startup feedwater flow to the steam generators.

The startup feedwater isolation and control valves are considered OPERABLE when they automatically close on an isolation actuation signal, and their isolation times are within the required limits.

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APPLICABILITY The startup feedwater isolation and control valves must be OPERABLE whenever there is significant mass and energy in the Reactor Coolant System and the steam generators. In MODES 1, 2, 3 and 4, the startup feedwater isolation and control valves are required to be OPERABLE in order to limit the amount of mass and energy that could be added to containment in the event of a SLB or FLB and prevent steam generator overfill in the event of an SGTR. When the valves are closed, they are already performing their safety function.

In MODES 5 and 6, the energy in the steam generators is low, and isolation of the startup feedwater system is not required.

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ACTIONS The ACTIONS are modified by a Note allowing flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the flow paths can be rapidly isolated.

The second Note allows separate Condition entry for each flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable flow path.

A.1 and A.2

With only one isolation or control valve OPERABLE in one or more flow paths, there is no redundant capability to isolate the flow paths. In this case, both an isolation and a control valve in each flow path must be restored to OPERABLE status with 72 hours, or the flow path must be isolated. A Completion Time of 72 hours is acceptable since, with one valve in a flow path inoperable, there is a second valve available in the flow path to isolate the line.

(continued)

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BASES

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ACTIONS

A.1 and A.2 (continued)

If the inoperable valve in the flow path can not be restored to OPERABLE status, then the flow path must be isolated within a Completion Time of 72 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.

For flow paths isolated in accordance with Required Action A.2.1, the affected flow paths must be verified to be isolated on a periodic basis. This is necessary to ensure that flow paths required to be isolated following an accident will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that the isolation devices are in the correct position. The Completion Time of "once per 7 days" is appropriate considering the fact that the devices are operated under administrative controls, valve status indications in the main control room and the probability of their misalignment is low.

B.1

With both the isolation and control valves inoperable in one flow path, the affected flow path must be restored to OPERABLE status or isolated within a Completion Time of 8 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.

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

If the isolation and control valves cannot be restored to OPERABLE status, closed, or isolated within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in least MODE 3 within 6 hours, and in MODE 4 with RCS cooling provided by the normal residual heat removal system within 24 hours, and the affected flow path isolated within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.7.1

This surveillance requires verification in accordance with the Inservice Testing Program to assure that both startup feedwater isolation and control valves are OPERABLE. The Surveillance Frequency is provided in the Inservice Testing Program.

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REFERENCES

1. Section 10.4.9, "Startup Feedwater System."
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B 3.7 PLANT SYSTEMS

B 3.7.8 Main Steam Line Leakage

BASES

BACKGROUND

A limit on leakage from the main steam line inside containment is required to limit system operation in the presence of excessive leakage. Leakage is limited to an amount which would not compromise safety consistent with the Leak-Before-Break (LBB) analysis discussed in Chapter 3 (Ref. 1). This leakage limit ensures appropriate action can be taken before the integrity of the lines is impaired.

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

As described in Section 3.6 (Ref. 1), LBB has been applied to the main steam line pipe runs inside containment. Hence, the potential safety significance of secondary side leaks inside containment requires detection and monitoring of leakage inside containment. This LCO protects the main steam lines inside containment against degradation, and helps assure that serious leaks will not develop. The consequences of violating this LCO include the possibility of further degradation of the main steam lines, which may lead to pipe break.

APPLICABLE  
SAFETY ANALYSES

The safety significance of plant leakage inside containment varies depending on its source, rate, and duration. Therefore, detection and monitoring of plant leakage inside containment are necessary. This is accomplished via the instrumentation required by LCO 3.4.10, "RCS Leakage Detection Instrumentation," and the RCS water inventory

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

balance (SR 3.4.8.1). Subtracting RCS leakage as well as any other identified non-RCS leakage into the containment area from the total plant leakage inside containment provides qualitative information to the operators regarding possible main steam line leakage. This allows the operators to take corrective action should leakage occur which is detrimental to the safety of the facility and/or the public.

Although the main steam line leakage limit is not required by the 10 CFR 50.36(c)(2)(ii) criteria, this specification has been included in Technical Specifications in accordance with NRC direction (Ref. 2).

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LCO

Main steam line leakage is defined as leakage inside containment in any portion of the two (2) main steam line pipe walls. Up to 0.5 gpm of leakage is allowable because it is below the leak rate for LBB analyzed cases of a main steam line crack twice as long as a crack leaking at ten (10) times the detectable leak rate under normal operating load conditions. Violation of this LCO could result in continued degradation of the main steam line.

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APPLICABILITY

Because of elevated main steam system temperatures and pressures, the potential for main steam line leakage is greatest in MODES 1, 2, 3, and 4.

In MODES 5 and 6, a main steam line leakage limit is not provided because the main steam system pressure is far lower, resulting in lower stresses and a reduced potential for leakage. In addition, the steam generators are not the primary method of RCS heat removal in MODES 5 and 6.

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(continued)

BASES (continued)

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ACTIONS

A.1 and A.2

With main steam line leakage in excess of the LCO limit, the unit must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be placed in MODE 3 with 8 hours and MODE 5 within 48 hours. This action reduces the main steam line pressure and leakage, and also reduces the factors which tend to degrade the main steam lines. The Completion Time of 6 hours to reach MODE 3 from full power without challenging plant systems is reasonable based on operating experience. Similarly, the Completion Time of 36 hours to reach MODE 5 without challenging plant systems is also reasonable based on operating experience. In MODE 5, the pressure stresses acting on the main steam line are much lower, and further deterioration of the main steam line is less likely.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.8.1

Verifying that main steam line leakage is within the LCO limit assures the integrity of those lines inside containment is maintained. An early warning of main steam line leakage is provided by the automatic system which monitor the containment sump level. Main steam line leakage would appear as unidentified leakage inside containment via this system, and can only be positively identified by inspection. However, by performance of an RCS water inventory balance (SR 3.4.8.1) and evaluation of the cooling and chilled water systems inside containment, determination of whether the main steam line is a potential source of unidentified leakage inside containment is possible.

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REFERENCES

1. Section 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping."
  2. NRC letter, Diane T. Jackson to Westinghouse (Nicholas J. Liparulo), dated September 5, 1996, "Staff Update to Draft Safety Evaluation Report (DSER) Open Items (OIs) Regarding the Westinghouse AP600 Advanced Reactor Design," Open Item #365.
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B 3.7 PLANT SYSTEMS

B 3.7.9 Fuel Storage Pool Makeup Water Sources

BASES

BACKGROUND

The spent fuel storage pool is normally cooled by the nonsafety spent fuel pool cooling system. In the event the normal cooling system is unavailable, the spent fuel storage pool can be cooled by the normal residual heat removal system. Alternatively, the spent fuel storage pool contains sufficient water inventory for decay heat removal by boiling. To support extended periods of loss of normal pool cooling, makeup water is required to provide additional cooling by boiling. Both safety and non-safety makeup water sources are available on-site.

Two safety-related, gravity fed sources of makeup water are provided to the spent fuel storage pool. These makeup water sources contain sufficient water to maintain spent fuel storage pool cooling for 72 hours. The containment cooling system water storage tank provides makeup water when pool decay heat is  $> 5.4$  MWt and the decay heat in the reactor is less than 9.0 MWt. The cask washdown pit provides makeup water when decay heat in the pool is  $\geq 4.6$  MWt and  $\leq 5.4$  MWt. Additional on-site makeup water sources are available to provide fuel pool cooling between 3 and 7 days.

The containment cooling system water storage tank is isolated by two normally closed valves. The normally closed valves will be opened only to provide emergency makeup to the spent fuel storage pool. A third downstream valve permits the operator to regulate addition of water to the spent fuel storage pool as required to maintain the cooling water inventory.

Once decay heat in the fuel pool is reduced to below 4.6 MWt, the spent fuel storage pool water inventory is sufficient, without makeup, to maintain spent fuel storage pool for 72 hours. When the spent fuel storage pool decay heat load is reduced below 4.6 MWt, the cask washdown pit may be drained and returned to use for shipping cask cleaning operations.

(continued)

BASES

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BACKGROUND  
(continued)

A general description of the fuel storage pool design is given in Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in Section 9.1.3 (Ref. 2).

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APPLICABLE  
SAFETY ANALYSES

In the event the normal spent fuel storage pool cooling system is unavailable, the spent fuel cooling is provided by the heat capacity of the water in the pool. The worst case decay heat load (decay heat > 5.4 MWt) is produced by an emergency full core off-load following a refueling plus ten years of spent fuel. For this case the spent fuel storage pool inventory provided by the water over the stored fuel and below the pump suction connection is capable of cooling the spent fuel storage pool without boiling for at least 2.5 hours, following a loss of normal spent fuel storage pool cooling. After boiling starts, makeup water may be required to replace water lost by boiling and is available, without offsite support, via the passive containment cooling water storage tank.

The requirements of LCO 3.6.6, "Passive Containment Cooling System – Operating," are applicable in MODES 1, 2, 3, and 4 and LCO 3.6.7, "Passive Containment Cooling System – Shutdown," are applicable in MODES 5 and 6 with decay heat > 9.0 MWt. LCOs 3.6.6 and 3.6.7 require availability of the containment cooling water tank for containment heat removal. Below 9.0 MWt decay heat, containment air cooling is adequate. Since there are no design conditions which result in both reactor decay heat > 9.0 MWt and spent fuel storage pool decay heat > 5.4 MWt, the applicability for LCOs 3.6.6/3.6.7 and for LCO 3.7.9 are mutually exclusive.

Since none of the Chapter 15 Design Basis Accident analyses assume availability of the containment cooling water tank or the cask washdown pit for spent fuel storage pool makeup, the fuel storage pool makeup water sources specification does not satisfy any of the 10 CFR 50.36(c)(2)(ii) criteria. This LCO is included in accordance with NRC guidance provided in an NRC letter (Reference 3).

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(continued)

BASES (continued)

LCO The fuel storage pool makeup water sources, the cask washdown pit, and the containment cooling water tank are required to contain 13.75 ft. and 400,000 gallons of water, respectively. An OPERABLE flow path from the required makeup source assures spent fuel cooling for at least 72 hours. Several additional makeup sources are available, including the ground level containment cooling ancillary water storage tank. These makeup sources assure spent fuel cooling for at least 7 days.

Note 1 specifies that either the cask washdown pit or the passive containment cooling water storage tank is required to be OPERABLE when the spent fuel storage pool decay heat  $\geq 4.6$  Mwt and  $\leq 5.4$  Mwt. Note 2 specifies that the passive containment cooling water storage tank source is required to be OPERABLE when the spent fuel storage pool decay heat is  $> 5.4$  Mwt, which is normal following a full core off load. The larger makeup source is necessary for the higher decay heat load. When a portion of the fuel is returned to the reactor vessel in preparation for startup, the pool decay heat is reduced to  $\leq 5.4$  Mwt and makeup from the cask washdown pit is sufficient.

APPLICABILITY This LCO applies during storage of fuel in the fuel storage pool with a calculated decay heat  $\geq 4.6$  Mwt. With decay heat  $< 4.6$  Mwt, the assumed spent fuel storage pool water inventory (i.e., level below the pump suction connection to the pool) provides for 3 days of cooling without makeup.

ACTIONS LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. Spent fuel pool cooling requirements are independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since spent fuel pool cooling requirements apply at all times, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. Spent fuel pool cooling requirements are independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

If the passive containment cooling water storage tank (with decay heat  $> 5.4$  Mwt) and/or the cask washdown pit (with decay heat  $\geq 4.6$  and  $\leq 5.4$  Mwt) is inoperable, Action must be initiated immediately to restore the makeup source or its associated flow path to OPERABLE status.

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BASES

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ACTIONS

A.1 (continued)

Additionally, in order to provide the maximum cooling capability, the spent fuel pool should be filled to its maximum level. Nonsafety related makeup sources can be used to fill the pool. This action is not specified in the specification, since the benefit of adding approximately 6 inches of water to the pool is less than a 5% improvement in cooling capability.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.9.1

This SR verifies sufficient passive containment cooling system water storage tank volume is available in the event of a loss of spent fuel cooling.

The 7 day Frequency is appropriate because the volume in the passive containment cooling system water storage tank is normally stable and water level changes are controlled by plant procedures.

SR 3.7.9.2

This SR verifies sufficient cask washdown pit water volume is available in the event of a loss of spent fuel cooling. The 13.75 ft. level specified provides makeup water for stored fuel with decay heat  $\geq 4.6$  and  $\leq 5.4$  MWt.

The 30 day Frequency is appropriate because the cask washdown pit has only one drain line which is isolated by series manual valves which are only operated in accordance with plant procedures, thus providing assurance that inadvertent level reduction is not likely.

SR 3.7.9.3

This SR requires verification of the OPERABILITY of the manual makeup water source isolation valves in accordance with the requirements and Frequency specified in the

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.9.3 (continued)

Inservice Testing Program. Manual valves PCS-PL-V009, PCS-PL-V045, PCS-PL-V051, isolate the makeup flow path from the passive containment cooling system water storage tank. Manual valves SFS-PL-V066 and SFS-PL-V068 isolate the makeup flow path from the cask washdown pit.

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REFERENCES

1. Section 9.1.2, "Spent Fuel Storage."
  2. Section 9.1.3, "Spent Fuel Pool Cooling System."
  3. NRC letter, William C. Huffman to Westinghouse Electric Corporation, "Summary of Telephone Conference with Westinghouse to Discuss Proposed Design Changes to the AP600 Main Control Room Habitability System," dated September 11, 1997.
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B 3.7 PLANT SYSTEMS

B 3.7.10 Steam Generator Isolation Valves

BASES

BACKGROUND

The steam generator isolation valves consist of the power operated relief valve (PORV) block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B), and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B). The PORV flow paths must be isolated following a Steam Generator Tube Rupture (SGTR) to minimize radiological releases. The blowdown flow path must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for Reactor Coolant System (RCS) heat removal.

A PORV is installed in a 6 inch branch line off of the main steam line piping from each steam generator, to provide for controlled removal of reactor decay heat during normal reactor cooldown when the main steam isolation valves are closed or the turbine bypass system is not available. A normally-open block valve is provided in each PORV line to provide backup isolation capability. Both the PORV and the block valve receive a Protection and Safety Monitoring System (PMS) isolation signal on low steam line pressure. The block valve is also a containment isolation valve.

The blowdown line from each steam generator is provided with two series isolation valves, both located outside, but close to, containment. The blowdown valves receive a PMS isolation signal on low SG level and on PRHR actuation. The first blowdown isolation valve outside of containment is also a containment isolation valve.

The steam generator PORVs and the blowdown isolation valves fail closed on loss of control or actuation power. The steam generator PORV block valves fail as-is on loss of control or actuation power. The steam generator isolation valves may also be actuated manually.

Descriptions of the PORVs and SG blowdown isolation are found in Section 10.3.2.2.3 and Section 10.4.8 (Refs. 1 & 2).

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(continued)

BASES (continued)

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APPLICABLE  
SAFETY ANALYSES

The PORV flow paths must be isolated following an SGTR to minimize radiological releases from the ruptured steam generator into the atmosphere. The PORV flow path is assumed to open due to high secondary side pressure, during the SGTR. Dose analyses take credit for subsequent isolation of the PORV flow path by the PORV and/or the block valve which receive a close signal on low steam line pressure.

The blowdown flow path on each SG must be isolated following Loss of Feedwater and Feedwater Line Break events to retain the steam generator water inventory for use in Reactor Coolant System (RCS) heat removal via the SGs. RCS heat removal for these events is, primarily, provided by the Passive Residual Heat Removal Heat Exchanger (PRHR HX); however, the SG heat removal is assumed. The SG blowdown isolation valves receive an isolation signal on low SG level or PRHR actuation. These events take credit for steam generator heat removal using the water inventory retained after blowdown isolation. If the blowdown line were not isolated, much of the inventory would drain from the SG rather than cool the RCS.

The steam generator isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

This LCO requires that the steam generator isolation valves consisting of the PORV, PORV block valve, and blowdown isolation valves on each steam generator to be OPERABLE. These isolation valves are considered OPERABLE when the valves are capable of closing on a PMS actuation signal.

This LCO provides assurance that the PORV and PORV block valve will perform their design safety function to mitigate the consequences of an SGTR that could result in offsite exposures.

Additionally, this LCO provides assurance that the steam generator blowdown isolation valves will perform their design safety function to mitigate the consequences of Loss of Feedwater and Feedwater Line Break events by retaining the steam generator water inventory for Reactor Coolant System (RCS) heat removal.

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(continued)

BASES (continued)

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**APPLICABILITY** The steam generator isolation valves must be OPERABLE in MODES 1, 2, and 3, and in MODE 4 with the RCS cooling not being provided by the Normal Residual Heat Removal System (RNS).

In MODE 4 with the RCS cooling being provided by the RNS and in MODES 5 and 6, the steam generators are not needed for RCS cooling and the potential for an SGTR or Loss of Feedwater and Feedwater Line Break events is minimized due to the reduced mass and energy in the RCS and steam generators.

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**ACTIONS** The ACTIONS are modified by a Note allowing the blowdown isolation flow paths to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the flow path can be rapidly isolated when a need for blowdown isolation is indicated.

The second Note allows separate Condition entry for each steam generator isolation flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable flow path.

A.1

With one valve in one or more PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours is based on the availability of one OPERABLE PORV flow path isolation valve which is fully capable of performing the required isolation function.

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(continued)

BASES

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ACTIONS  
(continued)

B.1 and B.2

With one valve in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. A Completion Time of 72 hours to isolate the flow path is based on the availability of one OPERABLE blowdown flow path isolation valve which is fully capable of performing the required isolation function.

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

C.1

With both valves in one or more PORV flow paths inoperable, action must be taken to isolate the flow path with a closed and deactivated valve. The valve must be deactivated to assure that the flow path will not be opened by a high pressure signal during the course of an SGTR event. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 6.

D.1 and D.2

With two valves in one or more blowdown flow paths inoperable, action must be taken to isolate the flow path with a closed valve. This action places the flow path in a condition which assures the safety function is performed. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the SG isolation valves. The incremental conditional core damage probability with this AOT is more than an order of magnitude less than the value indicated to have a small impact on plant risk in Reference 3.

(continued)

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BASES

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ACTIONS

D.1 and D.2 (continued)

Since the blowdown isolation valve is not deactivated, periodic verification is required to assure that the flow path remains isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of status indications available in the control room, and other administrative controls, to ensure that the valve remains in the closed position.

E.1 and E.2

If the SG isolation valves cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 with the RCS cooling provided by the RNS within 24 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.7.10.1

The function of the SG isolation valves (PORV block valves (SGS-PL-V027A & B), PORVs (SGS-PL-V233A & B) and blowdown isolation valves (SGS-PL-V074A & B and SGS-PL-V075A & B)) is to isolate the steam generators in the event of SGTR, Loss of Feedwater or Feedwater Line Break. Stroking the valves closed demonstrates their capability to perform the isolation function. The Frequency for this SR is in accordance with the Inservice Testing Program.

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REFERENCES

1. Section 10.3.2.2.3, "Power-Operated Atmospheric Relief Valves."
2. Section 10.4.8, "Steam Generator Blowdown System."
3. Regulatory Guide 1.177, 8/98, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 DC Sources – Operating

BASES

BACKGROUND

The Class 1E DC and UPS System (IDS) provides electrical power for safety related and vital control instrumentation loads, including monitoring and main control room emergency lighting. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost and cannot be recovered for 72 hours. As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the Class 1E DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The Class 1E DC electrical power system also conforms to the requirements of Regulatory Guide 1.6 (Ref. 2) and IEEE-308 (Ref. 3).

The 125 VDC electrical power system consists of four independent safety related Class 1E DC electrical power subsystems (Division A, B, C, and D). Divisions A and D each consist of one 24 hour battery bank, one battery charger, and the associated control equipment and interconnecting cable. Divisions B and C each consist of two battery banks (one 24 hour and one 72 hour), two battery chargers, and the associated control equipment and interconnecting cabling. The loads on the battery banks (including those on the associated inverters) are grouped according to their role in response to a Design Basis Accident (DBA). Loads which are a one time or limited duration load (engineered safeguards features (ESF) actuation cabinets and reactor trip function) required within the first 24 hours following an accident are connected to the "24 hour" battery bank. Loads which are continuous or required beyond the first 24 hours following an accident (emergency lighting, post accident monitoring, and Qualified Data Processing System) are connected to the "72 hour" battery bank. There are a total of six battery banks. A battery bank consists of two batteries connected in parallel. Each battery consist of 60 cells connected in series. Divisions A and D each have one 4800 ampere hour battery bank and Divisions B and C each have two 4800 ampere hour battery banks.

(continued)

BASES

BACKGROUND  
(continued)

Additionally, there is one installed spare battery bank and one installed spare battery charger, which provide backup service in the event that one of the battery banks and/or one of the preferred battery chargers is out of service. The spare battery bank and charger are Class 1E and have the same rating as the primary components. If the spare battery bank with the charger is substituted for one of the preferred battery banks or chargers, then the requirements of independence and redundancy between subsystems are maintained and the division is OPERABLE.

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

Each battery bank provides power to an inverter, which in turn powers an AC instrumentation and control bus. The AC instrumentation and control bus loads are connected to inverters according to the battery bank type, 24 hour or 72 hour.

The Class 1E DC power distribution system is described in more detail in Bases for LCO 3.8.5, "Distribution System – Operating," and LCO 3.8.6, "Distribution System – Shutdown."

Each battery has adequate storage capacity to carry the required load for the required duration as discussed in Reference 4.

Each 125 VDC battery bank, including the spare battery bank, is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystems to ensure that a single failure in one subsystem does not cause a failure in a separate subsystem. There is no sharing between separate Class 1E subsystems such as batteries, battery chargers, or distribution panels.

The batteries for each Class 1E electrical power subsystem are based on 125% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery discussed in Reference 4. The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each electrical power subsystem has ample power output capacity for the steady state operation of connected loads

(continued)

BASES

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BACKGROUND (continued) required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger has sufficient capacity to restore the battery bank from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads (Ref. 4).

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APPLICABLE SAFETY ANALYSES The initial conditions of DBA and transient analyses in the Chapter 6 (Ref. 6) and Chapter 15, (Ref. 7), assume that engineered safety features are OPERABLE. The Class 1E DC electrical power system provides 125 volts power for safety related and vital control instrumentation loads including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

The OPERABILITY of the Class 1E DC sources is consistent with the initial assumptions of the accident analyses. This includes maintaining at least three of the four divisions of DC sources OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite and onsite AC power sources; and
- b. A worst case single failure.

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

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LCO Class 1E DC electrical power subsystems are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of Class 1E DC electrical power from one division does not prevent the minimum safety function from being performed (Ref. 4).

An OPERABLE Class 1E DC electrical power subsystem requires all required batteries and respective chargers to be operating and connected to the associated DC bus(es). The spare battery and/or charger may be used by one subsystem for OPERABILITY.

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(continued)

BASES (continued)

APPLICABILITY The Class 1E DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment integrity and other vital functions are maintained in the event of a postulated DBA.

Class 1E DC electrical power requirements for MODES 5 and 6 are addressed in the Bases for LCO 3.8.2, "DC Sources – Shutdown."

ACTIONS

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

Condition A represents one division with one or two battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action A.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 6 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action A.2) from any discharge that might have occurred due to the charger inoperability.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary

shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If established battery terminal float voltage cannot be restored to greater than or equal to the minimum established float voltage within 6 hours, and the charger is not operating in the current-limiting mode, a faulty charger is indicated. A faulty charger that is incapable of maintaining established battery terminal float voltage does not provide assurance that it can revert to and operate properly in the current limit mode that is necessary during the recovery period following a battery discharge event that the DC system is designed for.

If the charger is operating in the current limit mode after 6 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action A.2).

Required Action A.2 requires that the battery float current be verified as less than or equal to [5] amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or equal to [5] amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action A.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The 7 day Completion

Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

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

Condition B represents two divisions with one or more battery chargers inoperable (e.g., the voltage limit of SR 3.8.1.1 is not maintained). The ACTIONS provide a tiered response that focuses on returning the battery to the fully charged state and restoring a fully qualified charger to OPERABLE status in a reasonable time period. Required Action B.1 requires that the battery terminal voltage be restored to greater than or equal to the minimum established float voltage within 2 hours. This time provides for returning the inoperable charger to OPERABLE status or providing an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage. Restoring the battery terminal voltage to greater than or equal to the minimum established float voltage provides good assurance that, within 24 hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to the charger inoperability.

A discharged battery having terminal voltage of at least the minimum established float voltage indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within 24 hours, avoiding a premature shutdown with its own attendant risk.

If the charger is operating in the current limit mode after 2 hours that is an indication that the battery is partially discharged and its capacity margins will be reduced. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within 24 hours (Required Action B.2).

Required Action B.2 requires that the battery float current be verified as less than or equal to [5] amps. This indicates that, if the battery had been discharged as the result of the inoperable battery charger, it has now been fully recharged. If at the expiration of the initial 24 hour period the battery float current is not less than or

equal to [5] amps this indicates there may be additional battery problems and the battery must be declared inoperable.

Required Action B.3 limits the restoration time for the inoperable battery charger to 7 days. This action is applicable if an alternate means of restoring battery terminal voltage to greater than or equal to the minimum established float voltage has been used (e.g., balance of plant non-Class 1E battery charger). The 7 day Completion Time reflects a reasonable time to effect restoration of the qualified battery charger to OPERABLE status.

#### C.1

Condition C represents one division with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery chargers. Any event that results in a loss of the AC bus supporting the battery chargers will also result in loss of DC to that train.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

#### D.1

Condition D represents two divisions with one or more batteries inoperable. With one or more batteries inoperable, the DC bus is being supplied by the OPERABLE battery charger. Any event that results in a loss of the AC bus supporting the battery charger will also result in loss of DC to that train. The [2] hour limit allows sufficient time to effect restoration of an inoperable battery given that the majority of the conditions that lead to battery

inoperability (e.g., loss of battery charger, battery cell voltage less than [2.07] V, etc.) are identified in Specifications 3.8.1, 3.8.2, and 3.8.7 together with additional specific completion times.

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC electrical power subsystem; however, all applicable Surveillances must be met by the spare equipment used, prior to declaring the subsystem OPERABLE.

#### E.1

If one of the Class 1E DC electrical power subsystems is inoperable, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis.

Because of the passive system design and the use of fail-safe components, the remaining Class 1E DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate most DBAs following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one DC subsystem against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System (PMS) (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the DC electrical power subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS Division is similar to loss of one DC

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(continued)

BASES

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ACTIONS

E.1 (continued)

electrical power subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

F.1

Condition F represents two subsystems with a loss of ability to completely respond to an event, and a potential loss of ability to remain energized during normal operation. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for complete loss of DC power to the affected distribution subsystems. The 2 hour limit is consistent with the allowed time for two inoperable DC distribution subsystems.

If two of the required DC electrical power subsystems are inoperable (e.g., inoperable battery, inoperable battery charger(s), or inoperable battery charger and associated inoperable battery), the two remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate all but the very worst case events. Since a subsequent worst case single failure would, however, result in the loss of the third subsystem, leaving only one subsystem with limited capacity to mitigate events, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 11) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

G.1 and G.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required 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

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(continued)

BASES

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ACTIONS

G.1 and G.2 (continued)

brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the battery chargers which support 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 and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. 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. This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 8).

SR 3.8.1.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. 9), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying [400] amps at the minimum established float voltage for [24] hours. The ampere requirements are based on the output rating of the

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.2 (continued)

chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least [2] hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is  $\leq$  [2] amps.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these 24 month intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

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BASES

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| SURVEILLANCE  
| REQUIREMENTS  
| (continued)

SR 3.8.1.3

A battery service test is a special test of battery capability, as found, to satisfy the design requirements (battery duty cycle) of the Class 1E DC electrical power system. The discharge rate and test length corresponds to the design duty cycle requirements as specified in Reference 4.

The Surveillance Frequency of 24 months is consistent with the recommendations of Regulatory Guide 1.32 (Ref. 9) and Regulatory Guide 1.129 (Ref. 10), which state that the battery service test should be performed with intervals between tests not to exceed 24 months. This Surveillance may be performed during any plant condition with the spare battery and charger providing power to the bus.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.1.3 (continued)

duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

This SR is modified by two Notes. Note 1 allows the performance of a modified performance discharge test in lieu of a service test.

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

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems if the spare battery is not connected. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment.

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| BASES (continued)

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| REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," U.S. Nuclear Regulatory Commission, March 10, 1971.
3. IEEE-308 1991, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers.
4. Section 8.3.2, "Class 1E DC Power Systems."
5. IEEE-485 1997, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronic Engineers, June 1983.
6. Chapter 6, "Engineered Safety Features."
7. Chapter 15, "Accident Analyses."

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BASES

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REFERENCES  
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8. IEEE-450 1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications," Institute of Electrical and Electronic Engineers, June 1986.
  9. Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1977.
  10. Regulatory Guide 1.129 Revision 1, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1978.
  11. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 DC Sources – Shutdown

BASES

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BACKGROUND            A description of the Class 1E DC power sources is provided in the Bases for LCO 3.8.1, "DC Sources – Operating."

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APPLICABLE SAFETY ANALYSES    The initial conditions of Design Basis Accident (DBA) and transient analyses in the Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the emergency auxiliaries and control and switching during all MODES of operation.

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

The OPERABILITY of the minimum Class 1E DC power sources during MODES 5 and 6 and during movement of irradiated fuel assemblies 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 Class 1E DC power sources are provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications 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 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, 3, and 4] have no specific analyses in MODES [5 and 6] because the energy contained within the reactor pressure boundary, reactor coolant temperature and

(continued)

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case Design Basis Accidents which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The Class 1E DC Sources satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Class 1E DC electrical power subsystems are required to be OPERABLE to support required trains of Class 1E Distribution System divisions required to be OPERABLE by LCO 3.8.6. This ensures the availability of sufficient Class 1E DC 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, inadvertent reactor vessel draindown).

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

APPLICABILITY

The Class 1E DC power sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies 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;
- b. Required features needed to mitigate a fuel-handling accident are available;

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(continued)

BASES

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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 Class 1E DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1, "DC Sources – Operating."

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ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1 and A.2

With one or more of the required (per LCO 3.8.6, "Distribution Systems – Shutdown") Class 1E DC power subsystems inoperable, the remaining subsystems may be capable of supporting sufficient systems to allow continuation of CORE ALTERATIONS, fuel movement, and/or operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with the associated DC power source(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCO ACTIONS. In many instances this option would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit) to assure continued safe operation. The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize 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 Class 1E DC electrical power to the unit safety systems.

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(continued)

BASES

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ACTIONS

A.1 and A.2 (continued)

The installed spare battery bank and charger may be used to restore an inoperable Class 1E DC power subsystem; however, all applicable surveillances must be met by the spare equipment used, prior to declaring the subsystem operable.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required Class 1E DC electrical power subsystems should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires performance of all Surveillances required by SR 3.8.1.1 through SR 3.8.1.8. Therefore, see the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

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REFERENCES

1. Chapter 6, "Engineered Safety Features."
  2. Chapter 15, "Accident Analysis."
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Inverters – Operating

BASES

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BACKGROUND

The inverters are the preferred source of power for the Class 1E AC instrument and control buses because of the stability and reliability they achieve. Divisions A and D, each consist of one Class 1E inverter. Divisions B and C, each consist of two inverters. The function of the inverter is to convert Class 1E DC electrical power to AC electrical power, thus providing an uninterruptible power source for the instrumentation and controls for the Protection and Safety Monitoring System (PMS). The inverters are powered from the Class 1E 125 V battery sources (Ref. 1).

Under normal operation, a Class 1E inverter supplies power to the Class 1E AC instrument and control bus. If the inverter is inoperable or the Class 1E 125 VDC input to the inverter is unavailable, the Class 1E AC instrument and control bus is powered from the backup source associated with the same division via a static transfer switch featuring a make-before-break contact arrangement. In addition, a manual mechanical bypass switch is used to provide a backup power source to the Class 1E AC instrument and control bus when the inverter is removed from service. The backup source is a Class 1E regulating 480-208/120 volt transformer providing a regulated output to the Class 1E AC instrument and control bus through a static transfer switch and a manual bypass switch.

In addition to powering safety loads, the Class 1E AC power sources are used for emergency lighting in the main control room and remote shutdown workstation. When a normal AC power source for emergency lighting is lost, the loads are automatically transferred to a Class 1E AC power source. Specific details on inverters and their operating characteristics are found in Chapter 8 (Ref. 1).

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APPLICABLE  
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) transient analyses in Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume engineered safety features are OPERABLE. The inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the PMS instrumentation and controls so that the fuel, Reactor Coolant System (RCS),

(continued)

BASES

APPLICABLE  
SAFETY ANALYSES  
(continued)

and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of AC instrument and control buses OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite and onsite AC power source; and
- b. A worst case single failure.

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

LCO

The inverters ensure the availability of AC electrical power for the systems instrumentation required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Maintaining the required inverters OPERABLE ensures that the redundancy incorporated into the design of the PMS instrumentation and controls is maintained. The six inverters ensure an uninterruptible supply of AC electrical power to the six Class 1E AC instrument and control buses even if all AC power sources are de-energized.

OPERABLE inverters require that the Class 1E AC instrument and control bus be powered by the inverter with output voltage and frequency within tolerances, and the power input to the inverter from a 125 VDC station battery.

This LCO is modified by a Note that allows one inverter to be disconnected from its associated Class 1E DC bus for  $\leq 72$  hours, if the associated Class 1E AC instrument and control bus is powered from its Class 1E regulating transformer during the period and all other inverters are

(continued)

BASES

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LCO  
(continued)

OPERABLE. This allows an equalizing charge to be placed on one battery bank. If the inverter was not disconnected, the resulting voltage condition might damage the inverter. These provisions minimize the loss of equipment that would occur in the event of a loss of offsite power. The 72 hour time period for the allowance minimizes the time during which a loss of offsite power could result in the loss of equipment energized from the affected Class 1E AC instrument and control bus while taking into consideration the time required to perform an equalizing charge on the battery bank.

The intent of this Note is to limit the number of inverters that may be disconnected. Only the inverter associated with the single battery bank undergoing an equalizing charge may be disconnected. All other inverters must be aligned to their associated batteries.

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APPLICABILITY

The inverters are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

Inverter requirements for MODES 5 and 6 are covered in the Bases for LCO 3.8.4, "Inverters - Shutdown."

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ACTIONS

A.1

With a required inverter inoperable, its associated Class 1E AC instrument and control bus is automatically energized from its regulating transformer. A manual switch is also provided which can be used if the static transfer switch does not properly function.

(continued)

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BASES

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ACTIONS

A.1 (continued)

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.5, "Distribution System – Operating." This ensures that the vital bus is re-energized within 12 hours.

Required Action A.1 allows 24 hours to fix the inoperable inverter and return it to service. The 24 hour time limit is based upon engineering judgment, taking into consideration the time required to repair an inverter and the additional risk to which the unit is exposed because of the inverter inoperability. This has to be balanced against the risk of an immediate shutdown, along with the potential challenges to safety systems such a shutdown might entail. When the AC instrument and control bus is powered from its regulating transformer, it is relying upon interruptible AC electrical power sources (offsite and onsite). The uninterruptible inverter source to the AC instrument and control buses is the preferred source for powering instrumentation trip setpoint devices.

B.1 and B.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to MODE 5 where the probability and consequences on an event are minimized. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.1

This Surveillance verifies that the inverters are functioning properly with all required switches and circuit breakers closed and Class 1E AC instrument and control buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the PMS instrumentation connected

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.3.1 (continued)

to the Class 1E AC instrument and control buses. The 7 day Frequency takes into account the effectiveness of the voltage and frequency instruments, the redundant capability of the inverters, and other indications available in the control room that alert the operator to inverter malfunctions.

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REFERENCES

1. Section 8.3.2.1.1.2, "Class 1E Uninterruptible Power Supplies."
  2. Chapter 6, "Engineered Safety Features."
  3. Chapter 15, "Accident Analyses."
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.4 Inverters – Shutdown

BASES

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BACKGROUND            A description of the inverters is provided in the Bases for Specification 3.8.3, "Inverters – Operating."

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APPLICABLE SAFETY ANALYSES    The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Protection and Monitoring System Engineered Safety Feature Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

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

The OPERABILITY of the minimum inverters to each Class 1E AC instrument and control bus during MODES 5 and 6, ensures that (Refs. 1 and 2):

- 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 power is available to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

In general, when the unit is shut down, the Technical Specifications 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 all on-site power is not required. The rationale for this is based on the fact that

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

many Design Basis Accidents (DBAs) that are analyzed in MODES 1, 2, 3, and 4 have no specific analyses in MODES 5 and 6 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence being significantly reduced or eliminated, and in minimal consequences. These deviations from DBA analysis assumptions and design requirements during shutdown conditions are allowed by the LCO for required systems.

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case Design Basis Accidents which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The Class 1E UPS inverters are part of the distribution system and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or postulated DBA. The battery powered inverters provide an uninterruptible supply of AC electrical power to the Class 1E AC instrument and control buses, even if the normal power supply from the 480 VAC is deenergized. OPERABILITY of the inverters requires that the Class 1E instrument and control buses be powered by the inverter with output voltage and frequency within tolerances, and the power input to the inverter from a 125 VDC station battery. This ensures the availability of

sufficient inverter power sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (fuel handling accidents, inadvertent reactor vessel draindown).

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(continued)

BASES (continued)

- APPLICABILITY The inverters required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies 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 are available;
  - 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.

Class 1E UPS inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.3, "Inverters – Operating."

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1 and A.2

If one or more required (per LCO 3.8.6, Distribution Systems – Shutdown) inverters are inoperable, the remaining OPERABLE inverters may be capable of supporting required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By allowance of the option to declare required features inoperable with associated inverter(s) inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., suspend CORE ALTERATIONS, movement of irradiated fuel

assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations

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(continued)

BASES

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ACTIONS

A.1 and A.2 (continued)

involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

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 inverters and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a regulating transformer.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.4.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and Class 1E AC instrument and control buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the Class 1E AC instrument and control buses. The 7 day Frequency takes into account the effectiveness of the voltage and frequency instruments, the redundant capability of the inverters, and other indications available in the

control room that alert the operator to inverter malfunctions.

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REFERENCES

1. Chapter 6, "Engineered Safety Features."
  2. Chapter 15, "Accident Analysis."
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 Distribution Systems – Operating

BASES

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BACKGROUND

The onsite Class 1E and DC and UPS electrical power distribution system is divided by division into four independent AC and DC electrical power distribution subsystems (Divisions A, B, C, and D).

The Class 1E AC distribution Divisions A and D each consists of one 208/120 V bus. The Class 1E AC distribution Divisions B and C each consists of two 208/120 V buses. The buses are normally powered from separate inverters which are connected to the respective Division Class 1E battery banks. The backup source provided for each Division for the Class 1E AC instrument and control buses is a Class 1E regulating transformer providing regulated output to the Class 1E AC instrument and control buses through a static transfer switch and a manual bypass switch. Power to the transformer is provided by the nonsafety related Main AC Power System. Additional description of this system may be found in the Bases for Specification 3.8.3, "Inverters – Operating."

The Class 1E DC distribution Divisions A and D each consists of one 125 VDC bus. The Class 1E DC distribution Divisions B and C each consists of two 125 VDC buses. The buses for the four Divisions are normally powered from their associated Division battery chargers. The backup source for each Class 1E DC bus is its associated Class 1E battery bank. Additionally, there is one installed spare Class 1E battery bank and one installed spare Class 1E battery charger, which can provide backup power to a Class 1E DC bus in the event that one of the battery banks or one of the chargers is out of service. Additional description of this system may be found in the Bases for Specification 3.8.1, "DC Sources Operating."

The list of all required distribution buses is presented in Table B 3.8.5-1 and shown in Section 8.3.2 (Ref. 1).

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(continued)

BASES (continued)

APPLICABLE  
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 2) and Chapter 15 (Ref. 3), assume engineered safety features (ESFs) are OPERABLE. The Class 1E 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 the ESFs 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 Technical Specifications 3.2 (Power Distribution Limits), 3.4 (Reactor Coolant System), and 3.6 (Containment Systems).

The OPERABILITY of the Class 1E AC and DC electrical power distribution systems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of Class 1E AC and DC power distribution systems OPERABLE during accident conditions in the event of:

- a. An assumed loss of all offsite and onsite AC power sources; and
- b. A worst case single failure.

The Class 1E AC and DC electrical power distribution system satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The required power distribution subsystems listed in Table B 3.8.5-1 ensure the availability of Class 1E AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. The Division A, B, C, and D Class 1E AC and DC electrical power distribution subsystems are required to be OPERABLE.

Maintaining the Division A, B, C, and D AC and DC electrical power distribution subsystems OPERABLE ensures that the redundancy incorporated into the design of the ESFs is not defeated. Three of the four Class 1E AC and DC power

(continued)

BASES

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LCO  
(continued)

distribution subsystems are capable of providing the necessary electrical power to the associated ESF components. Therefore, a single failure within any subsystem or within the electrical power distribution subsystems will not prevent safe shutdown of the reactor.

OPERABLE Class 1E DC electric power distribution subsystems require the associated buses, motor control centers, and electrical circuits to be energized to their proper voltage from either the associated battery bank or charger. The spare battery bank and/or chargers may be used by one subsystem for OPERABILITY. OPERABLE Class 1E AC electrical power distribution subsystems require the associated buses to be energized to their proper voltages and frequencies from the associated inverter or regulating transformer.

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APPLICABILITY

The Class 1E AC and DC electrical power distribution subsystems are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

- a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and
- b. Adequate core cooling is provided, and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The Class 1E AC and DC electrical power distribution subsystem requirements for MODES 5 and 6 are covered in the Bases for Specification 3.8.6, "Class 1E Distribution Systems – Shutdown."

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ACTIONS

A.1

With one division of the Class 1E AC instrument and control bus inoperable the remaining Class 1E AC instrument and control buses have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

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BASES

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ACTIONS

A.1 (continued)

Because of the passive system design and the use of fail-safe components, the remaining Class 1E AC instrument and control buses have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one AC instrument and control bus against the risks of a forced shutdown. Additionally, the Completion Time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC Power, which would have Required Action Completion Times shorter than 6 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected division; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 6 hour Completion Time takes into account the importance to safety of restoring the Class 1E AC instrument and control bus to OPERABLE status, the passive design of the ESF systems, the redundant capability afforded by the other OPERABLE Class 1E AC instrument and control buses, and the low probability of a DBA occurring during this period which requires more than two OPERABLE AC instrument and control buses.

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(continued)

BASES

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ACTIONS

A.1 (continued)

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System division (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the AC instrument and control inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one division AC instrument and control bus. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DC bus is inoperable and subsequently restored OPERABLE, the LCO may already have been not met for up to 6 hours. This could lead to a total of 12 hours, since initial failure of the LCO, to restore the AC instrument and control distribution system. At this time, a DC circuit could again become inoperable, and AC instrument and control distribution restored OPERABLE. This could continue indefinitely.

The Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition A was entered. The 12 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

B.1

With one Division of the Class 1E DC electrical power distribution subsystem inoperable, the remaining Divisions have the capacity to support a safe shutdown and to mitigate all DBAs, based on conservative analysis.

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(continued)

BASES

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ACTIONS

B.1 (continued)

Because of the passive system design and the use of fail-safe components, the remaining Divisions have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst case single failure. The 6 hour Completion Time is reasonable based on engineering judgement balancing the risks of operation without one Division against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown; and, if necessary, prepare and effect an orderly and safe shutdown.

The 6 hour Completion Time is also consistent with the time specified for restoration of one (of four) Protection and Safety Monitoring System division (LCO 3.3.2, ESFAS Instrumentation). Depending on the nature of the DC electrical power distribution subsystem inoperability, one supported division of instrumentation could be considered inoperable. Inoperability of a PMS division is similar to loss of one DC electrical power distribution subsystem. In both cases, actuation of the safety functions associated with one of the four subsystems/divisions may no longer be available.

This 6 hour limit is shorter than Completion Times allowed for most supported systems which would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 6 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions to restore power to the affected division; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

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BASES

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ACTIONS

B.1 (continued)

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an AC instrument and control bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 6 hours. This could lead to a total of 6 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored OPERABLE. This could continue indefinitely.

This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." This will result in establishing the "time zero" at the time the LCO was initially not met, instead of the time Condition B was entered. The 12 hour Completion Time is an acceptable limitation on this potential to fail to meet the LCO indefinitely.

C.1

With two divisions of AC instrument and control buses inoperable, the remaining OPERABLE buses are capable of supporting the minimum safety functions necessary to shut down the unit and maintain it in the safe shutdown condition. Overall reliability is reduced, however, since an additional single failure could result in the minimum required ESF functions not being supported. Therefore, the required divisions of AC instrument and control buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated [inverter via inverted DC, inverter using internal AC source, or Class 1E constant voltage transformer].

Condition C represents two divisions of AC instrument and control vital buses without power; potentially both the DC source and the associated AC source are nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all noninterruptable power. It is,

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BASES

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ACTIONS

C.1 (continued)

therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining buses and restoring power to the affected buses.

This 2 hour time limit is more conservative than Completion Times allowed for the vast majority of components that are without adequate AC instrument and control power. Taking exception to LCO 3.0.2 for components without adequate vital AC power, which would have the Required Action Completion Times shorter than 2 hours if declared inoperable, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) and not allowing stable operations to continue);
- b. The potential for decreased safety by requiring entry into numerous Applicable Conditions and Required Actions for components without adequate AC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time takes into account the importance to safety of restoring the AC instrument and control buses to OPERABLE status, the redundant capability afforded by the other OPERABLE buses, and the low probability of a DBA occurring during this period.

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(continued)

BASES

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ACTIONS

C.1 (continued)

The second Completion Time for Required Action C.1 establishes a limit on the maximum allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition C is entered while, for instance, a DC bus is inoperable and subsequently returned to OPERABLE, the LCO may already have been not met for up to 12 hours. This could lead to a total of 14 hours, since initial failure of the LCO, to restore the bus distribution system. At this time, a DC train could again become inoperable, and AC bus distribution restored to OPERABLE. This could continue indefinitely.

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

D.1

With two divisions of DC electrical power distribution subsystems inoperable, the remaining DC electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming no single failure. The overall reliability is reduced, however, because a single failure in the remaining DC electrical power distribution subsystem could result in the minimum required ESF functions not being supported. Therefore, the [required] DC buses must be restored to OPERABLE status within 2 hours by powering the bus from the associated battery or charger.

Condition D represents two subsystems without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative

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(continued)

BASES

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ACTIONS

D.1 (continued)

that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining divisions and restoring power to the affected divisions.

This 2 hour limit is more conservative than Completion Times allowed for the vast majority of components that would be without power. Taking exception to LCO 3.0.2 for components without adequate DC power, which would have Required Action Completion Times shorter than 2 hours, is acceptable because of:

- a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue;
- b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected divisions; and
- c. The potential for an event in conjunction with a single failure of a redundant component.

The 2 hour Completion Time for DC buses is consistent with Regulatory Guide 1.93 (Ref. 3).

The second Completion Time for Required Action D.1 establishes a limit on the maximum time allowed for any combination of required distribution subsystems to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition D is entered while, for instance, an AC instrument and control bus is inoperable and subsequently returned OPERABLE, the LCO may already have been not met for up to 12 hours. This could lead to a total of 14 hours, since initial failure of the LCO, to restore the DC distribution system. At this time, an AC train could again become inoperable, and DC distribution restored to OPERABLE. This could continue indefinitely.

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(continued)

BASES

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ACTIONS

D.1 (continued)

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

E.1 and E.2

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

F.1

With two Divisions with inoperable distribution subsystems that result in a loss of safety function, adequate core cooling, containment OPERABILITY and other vital functions for DBA mitigation would be compromised, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.5.1

This Surveillance verifies that the Class 1E AC and DC electrical power distribution subsystems are functioning properly, with the required circuit breakers and switches properly aligned. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the Class 1E AC and DC electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

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(continued)

BASES (continued)

Table B 3.8.5-1 (page 1 of 1)  
Class 1E AC and DC Electrical Power Distribution System

| TYPE                                 | VOLTAGE | DIVISION A*            | DIVISION B*            | DIVISION C*            | DIVISION D*            |
|--------------------------------------|---------|------------------------|------------------------|------------------------|------------------------|
| DC Buses                             | 125 Vdc | IDSA-DS-1              | IDSB-DS-1<br>IDSB-DS-2 | IDSC-DS-1<br>IDSC-DS-2 | IDSD-DS-1              |
| DC Distribution Panels               | 125 Vdc | IDSA-DD-1<br>IDSA-DK-1 | IDSB-DD-1<br>IDSB-DK-1 | IDSC-DD-1<br>IDSC-DK-1 | IDSD-DD-1<br>IDSD-DK-1 |
| AC Instrumentation and Control Buses | 120 Vac | IDSA-EA-1              | IDSB-EA-1<br>IDSB-EA-3 | IDSC-EA-1<br>IDSC-EA-3 | IDSD-EA-1              |

\* Each Division of the AC and DC electrical power distribution systems is a subsystem.

(continued)

BASES (continued)

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REFERENCES

1. Section 8.3.2, "DC Power Systems."
  2. Chapter 6, "Engineering Safety Features."
  3. Chapter 15, "Accident Analyses."
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.6 Distribution System – Shutdown

BASES

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BACKGROUND            A description of the Class 1E AC instrument and control bus and Class 1E DC electrical power distribution system is provided in the Bases for Specification 3.8.5, "Distribution System – Operating."

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APPLICABLE  
SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The Class 1E AC and DC electrical power sources and associated power distribution systems are designed to provide sufficient capacity, redundancy, and reliability to ensure the availability of necessary power to the ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the minimum Class 1E AC and DC electrical power sources and associated power distribution subsystems during MODES 5 and 6, and during movement of irradiated fuel assemblies 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 power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.

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(continued)

BASES

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| APPLICABLE<br>SAFETY ANALYSES<br>(continued) | The Class 1E AC and DC electrical power distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii). |
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|-----|--|
| 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 electrical distribution system necessary to support OPERABILITY of required systems, equipment, and components—all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY. |
|-----|--|

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

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|---------------|--|
| APPLICABILITY | The Class 1E AC and DC electrical power distribution subsystems are required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that: |
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- a. Systems 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;
- b. Systems needed to mitigate a fuel handling accident are available;

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(continued)

BASES

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APPLICABILITY  
(continued)

- 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 and refueling condition.

The Class 1E AC and DC electrical power distribution subsystem requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.5, "Distribution Systems – Operating."

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ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1 and A.2

If one or more required Class 1E DC or Class 1E AC instrument and control bus electrical power distribution subsystems are inoperable, the remaining OPERABLE divisions may be capable of supporting required features to allow continuation of CORE ALTERATIONS, fuel movement, and/or 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 will be implemented in accordance with the affected equipment LCO Required Actions. In many instances this would likely involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, any activities that could potentially result in inadvertent draining of the reactor vessel, and operations involving positive reactivity additions that could result in loss of required SDM (Mode 5) or boron concentration (Mode 6)). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than that what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in

RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions will 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 unit safety systems.

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

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BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.6.1

This Surveillance verifies that the Class 1E AC and DC electrical power distribution subsystems are functioning properly, with the required circuit breakers and switches properly aligned. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems and other indications available in the control room that alert the operator to subsystem malfunctions.

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REFERENCES

1. Chapter 6, "Engineered Safety Features."
  2. Chapter 15, "Accident Analysis."
  3. Section 8.3.2, "DC Power Systems."
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.7 Battery Parameters

BASES

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**BACKGROUND** LCO 3.8.7, Battery Parameters, delineates the limits on electrolyte temperature, level, float voltage and specific gravity for the DC power source batteries. A discussion of these batteries and their OPERABILITY requirements is provided in the Bases for LCO 3.8.1, "DC Sources - Operating," and LCO 3.8.2, "DC Sources - Shutdown." In addition to the limitations of this Specification, the [licensee controlled program] also implements a program specified in Specification 5.5.17 for monitoring various battery parameters that is based on the recommendations of IEEE Standard 450-1995, "IEEE Recommended Practice For Maintenance, Testing, And Replacement Of Vented Lead-Acid Batteries For Stationary Applications" (Ref. 3).

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**APPLICABLE SAFETY ANALYSES** The initial conditions of Design Basis Accident (DBA) and transient analyses in Chapter 6 (Ref. 1), and Chapter 15 (Ref. 2), assume engineered safety features are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for safety related and vital control instrumentation loads including monitoring and main control room emergency lighting during all MODES of operation. It also provides power for safe shutdown when all the onsite and offsite AC power sources are lost.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining at least three of the four Divisions of DC sources OPERABLE during accident conditions, in the event of:

- a. An assumed loss of all offsite and onsite AC power sources; and
- b. A worst case single failure.

Battery parameters satisfy the Criterion 3 of 10 CFR  
50.36(c)(2)(ii).

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LCO

Battery parameters must remain within acceptable limits to ensure availability of the required DC power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. Battery parameter limits are conservatively established, allowing continued DC electrical system function even with limits not met. Additional preventative maintenance, testing, and monitoring performed in accordance with the [licensee controlled program] is conducted as specified in Specification 5.5.17.

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BASES (continued)

APPLICABILITY The battery parameters are required solely for the support of the associated DC electrical power subsystems. Therefore, battery parameter limits are only required when the DC power source is required to be OPERABLE. Refer to the Applicability discussion in Bases for LCO 3.8.1, and LCO 3.8.2.

ACTIONS

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

With one or more cells in one or more batteries in one Division  $< [2.07]$  V, the battery cell is degraded. Within 2 hours verification of the required battery charger, OPERABILITY is made by monitoring the battery terminal voltage (SR 3.8.1.1) and of the overall battery state of charge by monitoring the battery float charge current (SR 3.8.7.1). This assures that there is still sufficient battery capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of one or more cells in one or more batteries  $< [2.07]$  V, and continued operation is permitted for a limited period up to 24 hours.

Since the Required Actions only specify "perform," a failure of SR 3.8.1.1 or SR 3.8.7.1 acceptance criteria does not result in this Required Action not met. However, if one of the SRs is failed the appropriate Condition(s), depending on the cause of the failures, is entered. If SR 3.8.7.1 is failed then there is not assurance that there is still sufficient battery capacity to perform the intended function and the battery must be declared inoperable immediately.

B.1 and B.2

One or more batteries in one Division with float  $> [2]$  amps indicates that a partial discharge of the battery capacity has occurred. This may be due to a temporary loss of a battery charger or possibly due to one or more battery cells in a low voltage condition reflecting some loss of capacity. Within 2 hours verification of the required battery charger OPERABILITY is made by monitoring the battery terminal voltage. If the terminal voltage is found to be less than the minimum established float voltage there are two possibilities, the battery charger is inoperable or is operating in the current limit mode. Condition A addresses

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BASES

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ACTIONS

B.1 and B.2 (continued)

charger inoperability. If the charger is operating in the current limit mode after 2 hours that is an indication that the battery has been substantially discharged and likely cannot perform its required design functions. The time to return the battery to its fully charged condition in this case is a function of the battery charger capacity, the amount of loads on the associated DC system, the amount of the previous discharge, and the recharge characteristic of the battery. The charge time can be extensive, and there is not adequate assurance that it can be recharged within [12] hours (Required Action B.2). The battery must therefore be declared inoperable.

If the float voltage is found to be satisfactory but there are one or more battery cells with float voltage less than [2.07] V, the associated "OR" statement in Condition F is applicable and the battery must be declared inoperable immediately. If float voltage is satisfactory and there are no cells less than [2.07] V there is good assurance that, within [12] hours, the battery will be restored to its fully charged condition (Required Action B.2) from any discharge that might have occurred due to a temporary loss of the battery charger.

A discharged battery with float voltage (the charger setpoint) across its terminals indicates that the battery is on the exponential charging current portion (the second part) of its recharge cycle. The time to return a battery to its fully charged state under this condition is simply a function of the amount of the previous discharge and the recharge characteristic of the battery. Thus there is good assurance of fully recharging the battery within [12] hours, avoiding a premature shutdown with its own attendant risk.

If the condition is due to one or more cells in a low voltage condition but still greater than [2.07] V and float voltage is found to be satisfactory, this is not indication of a substantially discharged battery and [12] hours is a reasonable time prior to declaring the battery inoperable.

Since Required Action B.1 only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in the Required Action not met. However, if SR 3.8.1.1 is failed, the appropriate Condition(s), depending on the cause of the failure, is entered.

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

With one or more batteries in one Division with one or more cells electrolyte level above the top of the plates, but below the minimum established design limits, the battery still retains sufficient capacity to perform the intended function. Therefore, the affected battery is not required to be considered inoperable solely as a result of electrolyte level not met. Within 31 days the minimum established design limits for electrolyte level must be re-established.

With electrolyte level below the top of the plates there is a potential for dryout and plate degradation. Required Actions C.1 and C.2 address this potential (as well as provisions in Specification 5.5.17, Battery Monitoring and Maintenance Program). They are modified by a note that indicates they are only applicable if electrolyte level is below the top of the plates. Within 8 hours level is required to be restored to above the top of the plates. The Required Action C.2 requirement to verify that there is no leakage by visual inspection and the Specification 5.5.17.b item to initiate action to equalize and test in accordance with manufacturer's recommendation are taken from Annex D of IEEE Standard 450-1995. They are performed following the restoration of the electrolyte level to above the top of the plates. Based on the results of the manufacturer's recommended testing the batteries may have to be declared inoperable and the affected cells replaced.

D.1

With one or more batteries in one Division with pilot cell temperature less than the minimum established design limits, 12 hours is allowed to restore the temperature to within limits. A low electrolyte temperature limits the current and power available. Since the battery is sized with margin, while battery capacity is degraded, sufficient capacity exists to perform the intended function and the affected battery is not required to be considered inoperable solely as a result of the pilot cell temperature not met.

E.1

With one or more batteries in two or more Divisions with battery parameters not within limits there is not sufficient assurance that battery capacity has not been affected to the degree that the batteries can still perform their required function, given that redundant batteries are involved. With redundant batteries involved this potential could result in

a total loss of function on multiple systems that rely upon the batteries. The longer Completion Times specified for battery parameters on non-redundant batteries not within limits are therefore not appropriate, and the parameters must be restored to within limits in three Divisions within 2 hours.

#### F.1

With one or more batteries with any battery parameter outside the allowances of the Required Actions for Condition A, B, C, D, or E, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding battery must be declared inoperable. Additionally, discovering one or more batteries in one Division with one or more battery cells float voltage less than [2.07] V and float current greater than [2] amps indicates that the battery capacity may not be sufficient to perform the intended functions. The battery must therefore be declared inoperable immediately.

#### SURVEILLANCE REQUIREMENTS

#### SR 3.8.7.1

Verifying battery float current while on float charge is used to determine the state of charge of the battery. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a charged state. The float current requirements are based on the float current indicative of a charged battery. Use of float current to determine the state of charge of the battery is consistent with IEEE-450 (Ref. 3). The 7 day Frequency is consistent with IEEE-450 (Ref. 3).

This SR is modified by a Note that states the float current requirement is not required to be met when battery terminal voltage is less than the minimum established float voltage of SR 3.8.1.1. When this float voltage is not maintained the Required Actions of LCO 3.8.1 ACTION A are being taken, which provide the necessary and appropriate verifications of the battery condition. Furthermore, the float current limit of [2] amps is established based on the nominal float voltage value and is not directly applicable when this voltage is not maintained.

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BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.8.7.2 and SR 3.8.7.5

Optimal long term battery performance is obtained by maintaining a float voltage greater than or equal to the minimum established design limits provided by the battery manufacturer, which corresponds to [130.5] V at the battery terminals, or [2.25] Vpc. This provides adequate over-potential, which limits the formation of lead sulfate and self discharge, which could eventually render the battery inoperable. Float voltages in this range or less, but greater than [2.07] Vpc, are addressed in Specification 5.5.17. SRs 3.8.7.2 and 3.8.7.5 require verification that the cell float voltages are equal to or greater than the short term absolute minimum voltage of [2.07] V. The Frequency for cell voltage verification every 31 days for pilot cell and 92 days for each connected cell is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.3

The limit specified for electrolyte level ensures that the plates suffer no physical damage and maintains adequate electron transfer capability. The Frequency is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.4

This Surveillance verifies that the pilot cell temperature is greater than or equal to the minimum established design limit (i.e., [40]°F). Pilot cell electrolyte temperature is maintained above this temperature to assure the battery can provided the required current and voltage to meet the design requirements. Temperatures lower than assumed in battery sizing calculations act to inhibit or reduce battery capacity. The Frequency is consistent with IEEE-450 (Ref. 3).

SR 3.8.7.6

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

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BASES

SURVEILLANCE  
REQUIREMENTS

SR 3.8.7.6 (continued)

Either the battery performance discharge test or the modified performance discharge test is acceptable for satisfying SR 3.8.7.6; however, only the modified performance discharge test may be used to satisfy the battery service test requirements of SR 3.8.1.3.

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

It may consist of just two rates; for instance the one minute rate for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the service test. Since the ampere-hours removed by a one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test must remain above the minimum battery terminal voltage specified in the battery service test for the duration of time equal to that of the service test.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 3) and IEEE-485 (Ref. 4). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. Furthermore, the battery is sized to meet the assumed duty cycle loads when the battery design capacity reaches this [80]% limit.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% of its expected life and capacity is < 100% of the manufacturer's rating, the Surveillance Frequency is reduced to 12 months. However, if the battery

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.8.7.6 (continued)

shows no degradation but has reached 85% of its expected life, the Surveillance Frequency is only reduced to 24 months for batteries that retain capacity  $\geq 100\%$  of the manufacturer's ratings. Degradation is indicated, according to IEEE-450 (Ref. 3), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is  $\geq [10\%]$  below the manufacturer's rating. These Frequencies are consistent with the recommendations in IEEE-450 (Ref. 3).

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment.

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BASES (continued)

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- REFERENCES
1. Chapter 6, "Engineered Safety Features."
  2. Chapter 15, "Accident Analyses."
  3. IEEE-450 1995, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications."
  4. IEEE-485-1983, June 1983.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.1 Boron Concentration

#### BASES

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#### BACKGROUND

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

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of  $k_{eff} \leq 0.95$  during fuel handling with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by procedures.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled down and depressurized, the vessel head is unbolted and slowly removed. The refueling cavity and the fuel transfer canal are then flooded with borated water from the In-containment Refueling Water Storage Tank (IRWST) by the use of the Spent Fuel Pool Cooling System (SFS).

During refueling, the water volumes in the RCS, the fuel transfer canal and the refueling cavity are contiguous. However, the soluble boron concentration is not necessarily the same in each volume. If additions of boron are required during refueling, the Chemical and Volume Control System (CVS) provides the borated makeup.

The pumping action of the Normal Residual Heat Removal System (RNS) in the RCS, the SFS pumps in the spent fuel pool and refueling cavity, and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the fuel transfer canal. The RNS is in operation during

(continued)

**BASES**

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**BACKGROUND**  
(continued)           refueling to provide forced circulation in the RCS, while the SFS is in operation to cool and purify the spent fuel pool and refueling cavity. Their operation assists in maintaining the boron concentration in the RCS, the refueling cavity, and fuel transfer canal above the COLR limit.

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**APPLICABLE SAFETY ANALYSES**   The boron concentration limit, specified in the COLR, is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the  $k_{eff}$  of the core will remain  $\leq 0.95$  during the refueling operation. Hence, at least a 5%  $\Delta k/k$  margin of safety is established during refueling.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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**LCO**                   The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling cavity and the transfer tube while in MODE 6. The boron concentration limit specified in the COLR ensures that a core  $k_{eff} \leq 0.95$  is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

---

**APPLICABILITY**       This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a  $k_{eff}$  of  $\leq 0.95$ . Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" ensures that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

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(continued)

BASES (continued)

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ACTIONS

A.1 and A.2

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

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude completion of actions to establish a safe condition, including moving a component to a safe position.

A.3

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

In determining the required combination of boration flow rate and concentration, no unique design basis accident (DBA) must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator shall begin boration with the best source available for plant operations.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.1.1

This SR verifies that the coolant boron concentration in the RCS, the refueling cavity and the fuel transfer canal is within the COLR limit. The boron concentration of the coolant in each volume is determined periodically by chemical analysis.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.1.1 (continued)

A minimum Frequency of once every 72 hours is a sufficient interval to verify the boron concentration. The surveillance interval is based on operating experience, isolation of unborated water sources in accordance with LCO 3.9.2, and the availability of the source range neutron flux monitors required by LCO 3.9.3.

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REFERENCES

1. Chapter 15, "Accident Analysis."
  2. NS-57.2, ANSI/ANS-57.2-1983, Section 6.4.2.2.3, American Nuclear Society, American National Standard, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," 1983.
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B 3.9 REFUELING OPERATIONS

B 3.9.2 Unborated Water Source Flow Paths

BASES

**BACKGROUND** During MODE 6 operation, all flow paths for reactor makeup water sources containing unborated water which are connected to the Reactor Coolant System (RCS) must be closed to prevent an unplanned dilution of the reactor coolant. At least one isolation valve in each flow path must be secured in the closed position.

The Chemical and Volume Control System is capable of supplying borated and unborated water to the RCS through various flow paths. Since a positive reactivity addition, made by reducing the boron concentration, is inappropriate during MODE 6, isolation of all unborated water sources prevents an unplanned boron dilution event.

**APPLICABLE SAFETY ANALYSES**

The possibility of an unplanned boron dilution event (Ref. 1) in MODE 6 is precluded by adherence to this LCO which requires that potential dilution sources be isolated. Closing the required valves during refueling operations prevents the flow of unborated water to the filled portions of the RCS. The valves are used to isolate unborated water sources. These valves have the potential to indirectly allow dilution of the RCS boron concentration in MODE 6. By isolating unborated water sources, a safety analysis for an uncontrolled boron dilution accident in accordance with the Standard Review Plan (Ref. 2) is not required in MODE 6.

The RCS boron concentration satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

**LCOs**

This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during MODE 6 and, thus, avoid a reduction in SHUTDOWN MARGIN.

(continued)

BASES (continued)

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APPLICABILITY In MODE 6, this LCO is applicable to prevent an unplanned boron dilution event by ensuring isolation of all sources of unborated water to the RCS.

In MODES 1 through 5, the requirements of LCO 3.1.9, "Chemical and Volume Control System (CVS) Demineralized Water Isolation Valves," apply.

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ACTIONS The ACTIONS Table has been modified by a Note which allows separate Condition entry for each unborated water source flow path.

A.1

Continuation of CORE ALTERATIONS is contingent upon maintaining the plant in compliance with this LCO. With any valve used to isolate unborated water sources not secured in the closed position, all operations involving CORE ALTERATIONS must be suspended immediately. The Completion Time of "Immediately" shall not preclude completion of actions to establish a safe condition, including movement of a component to a safe location.

Condition A has been modified by a Note to require that Required Action A.3 must be completed whenever Condition A is entered.

A.2

Preventing unplanned dilution of the reactor coolant boron concentration is dependent on maintaining the unborated water isolation valves secured closed. Securing the valves in the closed position verifies that the valves cannot be inadvertently opened. The Completion Time of "Immediately" requires an operator to initiate actions to close an open valve and secure the isolation valve in the closed position immediately. Once actions are initiated, they must be continued until the valves are secured in the closed position.

A.3

Due to the potential of having diluted the boron concentration of the reactor coolant, SR 3.9.1.1

(continued)

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BASES

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ACTIONS

A.3 (continued)

(verification of boron concentration) must be performed whenever Condition A is entered to verify that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.2.1

These valves are to be secured closed to isolate possible dilution flow paths. The likelihood of a significant reduction in the boron concentration during MODE 6 operations is remote due to the large mass of borated water in the refueling cavity and the fact that all unborated water source flow paths are isolated, precluding a dilution. The boron concentration is checked every 31 days during MODE 6 under SR 3.9.1.1. This surveillance demonstrates that the valves are closed through a system walkdown. The 31 day Frequency is based on engineering judgement and is considered reasonable in view of other administrative controls that will verify that the valve opening is an unlikely possibility.

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REFERENCES

1. Chapter 15, "Accident Analyses."
  2. NUREG-0800, Standard Review Plan, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the RCS."
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B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

**BACKGROUND** The source range neutron flux monitors are used to monitor the core reactivity during refueling operations. The source range neutron flux monitors are part of the Protection and Safety Monitoring System (PMS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

The source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux ( $1 \times 10^6$  cps) with a 5% instrument accuracy. The detectors also provide continuous visual and audible indication in the main control room and an audible alarm in the main control room and containment building.

**APPLICABLE SAFETY ANALYSES** Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as those associated with an improperly loaded fuel assembly. During initial fuel loading, or when otherwise required, temporary neutron detectors may be used to provide additional reactivity monitoring (Ref. 2). The potential for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2 (Ref. 1).

The source range neutron flux monitors satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

**LCO** This LCO requires two source range neutron flux monitors to be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

(continued)

BASES (continued)

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APPLICABILITY In MODE 6, the source range neutron flux monitors are required to be OPERABLE to determine possible changes in core reactivity. There are no other direct means available to monitor the core reactivity conditions. In MODES 2, 3, 4, and 5, the source range detectors and associated circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System Instrumentation."

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ACTIONS

A.1 and A.2

Redundancy has been lost if only one source range neutron flux monitor is OPERABLE. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

B.1

If no source range neutron flux monitors are OPERABLE, actions to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, actions shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

If no source range neutron flux monitors are OPERABLE, there is no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are discontinued, the core reactivity condition is stabilized and no changes are permitted until the source range neutron flux monitors are restored to OPERABLE status. This stable condition is confirmed by performing SR 3.9.1.1 to verify that the required boron concentration exists.

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable considering the low probability of a change in core reactivity during this time period.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is the comparison of the indicated parameter values monitored by each of these instruments. It is based on the assumption that the two indication channels should be consistent for the existing core conditions. Changes in core geometry due to fuel loading can result in significant differences between the source range channels, however each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for these same instruments in LCO 3.3.1, "Reactor Trip System Instrumentation."

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 24 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consisting of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 24 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 during the refueling outage.

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REFERENCES

1. Chapter 15, "Accident Analysis."
  2. Section 14.2.6.1, "Initial Fuel Loading."
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B 3.9 REFUELING OPERATIONS

B 3.9.4 Refueling Cavity Water Level

BASES

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BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum water level of 23 ft. above the top of the reactor vessel flange. During refueling, this maintains sufficient water level in containment, refueling cavity, refueling canal, fuel transfer canal, and spent fuel pool to retain iodine fission product activity in the event of a fuel handling accident (Ref. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to within the values reported in Chapter 15.

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APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling cavity and the refueling canal is an initial condition design parameter in the analysis of a fuel-handling accident in containment, as postulated by Regulatory Guide 1.183 (Ref. 1).

The fuel handling accident analysis inside containment is described in Reference 2. This analysis assumes a minimum water level of 23 feet.

Refueling Cavity Water Level satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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LCO

A minimum refueling cavity water level of 23 ft. above the reactor vessel flange is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within the values calculated in Reference 2.

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APPLICABILITY

Refueling Cavity Water Level is applicable when moving irradiated fuel assemblies in containment. The LCO minimizes the

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BASES

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APPLICABILITY  
(continued)

possibility of radioactive release due to a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel assemblies are not being moved in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel pool are covered by LCO 3.7.11, "Fuel Storage Pool Water Level."

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ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

With a water level of < 23 ft. above the top of the reactor vessel flange, all operations involving movement of irradiated fuel assemblies within containment shall be suspended immediately to ensure that a fuel handling accident cannot occur.

The suspension of fuel movement shall not preclude completion of movement to safe position.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.4 1

Verification of a minimum water level of 23 ft. above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgement and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions which make significant unplanned level changes unlikely.

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(continued)

BASES (continued)

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- REFERENCES
1. Regulatory Guide 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."
  2. Section 15.7.4, "Fuel Handling Accident."
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B 3.9 REFUELING OPERATIONS

B 3.9.5 Containment Penetrations

BASES

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BACKGROUND

During movement of irradiated fuel assemblies within containment, potential releases of fission product radioactivity within containment are monitored and filtered or are restricted from escaping to the environment when the LCO requirements are met. Monitoring of potential releases of radiation is performed in accordance with Administrative Controls Section 5.5.3, "Radioactive Effluent Control Program." In MODES 1, 2, 3, and 4, containment operability is addressed in LCO 3.6.1, "Containment." In MODES 5 and 6, closure capability of containment penetrations is addressed in LCO 3.6.8, "Containment Penetrations." Since there is no potential for containment pressurization due to a fuel handling accident, the Appendix J leakage criteria and tests are not required in MODES 5 and 6.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.34. For a fuel handling accident, the AP1000 dose analysis does not rely on containment closure to meet the offsite radiation exposure limits. This LCO is provided as an additional level of defense against the possibility of a fission product release from a fuel handling accident.

The containment equipment hatches, which are part of the containment pressure boundary, provide a means for moving large equipment and components into and out of containment. During movement of irradiated fuel assemblies within containment, an equipment hatch is considered closed if the hatch cover is held in place by at least [four] bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.

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(continued)

BASES

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BACKGROUND  
(continued)

If the equipment hatch is open, an alternative barrier between the containment atmosphere and the outside atmosphere shall be in place. Each containment equipment hatch opens into a staging area in the auxiliary building. These staging areas contain doors that open to the radiologically controlled areas of the annex building. The annex building contains a door that opens to the outside atmosphere. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed. The alternate barrier may also consist of a temporary equipment hatch cover that provides equivalent isolation capability. The alternate boundary prevents the airborne fission products from being readily released to the atmosphere if the equipment hatches were open during a fuel handling accident.

If an equipment hatch is open during movement of irradiated fuel assemblies within containment, the containment air filtration system (VFS) shall be OPERABLE, and at least one exhaust fan shall be operating to provide for monitoring of air-borne radioactivity. This system services the containment, and upon detection of high radiation, also services the fuel handling area, the auxiliary building (including the staging areas), and the annex building. If high airborne radioactivity is detected in the area enclosed by the alternate barrier, the radiologically controlled area ventilation system (VAS) supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment, and the VAS exhaust is automatically aligned to the VFS exhaust subsystem. The operation of the VFS exhaust fans provides the system with the ability for monitoring of radioactivity releases from containment following a fuel handling accident and, if operating, will provide filtration of the containment atmosphere.

If a personnel air lock or spare containment penetration is open during movement of irradiated fuel assemblies within containment, then the containment air filtration system (VFS) shall be OPERABLE

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BASES

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BACKGROUND  
(continued)

and operating to monitor for the release of radioactivity and to provide filtration of the air inside containment. These penetrations open into the auxiliary building. Upon detection of high radiation in the exhaust air from the auxiliary building, VFS will provide filtered exhaust of these areas. Considering that these penetrations open into the auxiliary building and not directly to the atmosphere, and that the VFS is in operation, an alternate barrier to the release of radioactivity directly to the environment is provided.

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APPLICABLE  
SAFETY ANALYSES

For the AP1000, there are no safety analyses that require containment closure during movement of irradiated fuel assemblies within containment, other than those discussed in LCO 3.6.8. Fuel handling accidents, analyzed in Reference 1, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.4, "Refueling Cavity Water Level," ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 50.34. Standard Review Plan, Section 15.0.1 (Reference 2), defines the dose acceptance limit to be 25% of the limiting dose guideline values.

This specification is included as defense-in-depth.

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LCO

This LCO provides defense-in-depth against the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. This LCO requires that if an equipment hatch, personnel air lock, or spare containment penetration is open during movement of irradiated fuel assemblies within containment, then the containment air filtration system (VFS) shall be OPERABLE and operating to monitor for the release of radioactivity and to provide filtration of the air inside containment.

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## BASES

LCO  
(continued)

The VFS is OPERABLE when:

- a. One VFS exhaust fan is operating; the associated HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and air circulation can be maintained;
- b. An alternative barrier between the containment atmosphere and the outside atmosphere is in place. The alternate barrier may consist of the staging area in the auxiliary building, or may consist of the staging areas in the auxiliary building and the radiologically controlled areas in the annex building provided the doors from the annex building to the outside atmosphere are closed.

Doors in the alternate barrier which are normally closed may be opened for short periods of time for ingress and egress. The alternate barrier may also consist of a temporary equipment hatch cover that provides equivalent isolation capability.

## APPLICABILITY

The containment penetration requirements are applicable during movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Containment closure capability in MODES 5 and 6 are addressed by LCO 3.6.8.

## ACTIONS

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

The required status for the containment equipment hatch, air

locks or spare penetration is either closed, or open with the VFS OPERABLE and operating. The required status for the containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere is either closed by a manual or automatic isolation valve, blind

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(continued)

BASES

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ACTIONS

A.1 and A.2 (continued)

flange or equivalent, or capable of being closed by an OPERABLE Containment Isolation Signal. If the containment equipment hatch or air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.5.1

This Surveillance verifies that each of the containment penetrations required to be in its closed position is in that position or the VFS is OPERABLE and operating. For the VFS to be considered OPERABLE, this surveillance also requires that an alternate barrier is in place.

SR 3.9.5.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. The Frequency is in accordance with the Inservice Testing Program.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

SR 3.9.5.3

This SR verifies the ability of the VFS to maintain a negative pressure ( $\leq [-0.125]$  inches water gauge relative to outside atmospheric pressure) in the containment and the portions of the auxiliary and/or annex building that comprise the envelope defined as the alternate barrier. This surveillance is performed with the VFS in containment

operating. Doors in the alternate barrier which are normally closed may be opened for ingress and egress. The portion of the VAS which services the area enclosed by the alternate barrier is aligned to the VFS exhaust subsystem,

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.5.3 (continued)

and the VAS auxiliary/annex building supply fans and VFS containment purge supply fans not operating. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 3).

SR 3.9.5.4

The VFS should be checked periodically to ensure that it functions properly. As the operating conditions on this system are not severe, testing each train within 31 days prior to fuel movement provides an adequate check on this system. Operation of the heater dries out any moisture accumulated in the charcoal from humidity in the ambient air.

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REFERENCES

1. Section 15.7.4, "Fuel Handling Accident."
  2. NUREG-0800, Section 15.0.1, Rev. 0.
  3. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
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## B 3.9 REFUELING OPERATIONS

### B 3.9.6 Containment Air Filtration System (VFS)

#### BASES

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##### BACKGROUND

The radiologically controlled area ventilation system (VAS) serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system (VHS). If high airborne radioactivity is detected in the exhaust air from the fuel handling area, the auxiliary building, or the annex buildings, the VAS supply and exhaust duct isolation dampers automatically close to isolate the affected area from the outside environment and the containment air filtration exhaust subsystem starts. The VFS exhaust subsystem prevents exfiltration of unfiltered airborne radioactivity by maintaining the isolated zone at  $\leq [-0.125]$  inches water gauge pressure relative to the outside atmosphere. Monitoring of potential releases of radiation is performed in accordance with Administrative Controls Section 5.5.3, "Radioactive Effluent Control Program."

For a fuel handling accident, the AP1000 dose analysis does not rely on the OPERABILITY of the VAS or VFS exhaust subsystem to meet the offsite radiation exposure limits. This LCO is provided as an additional level of defense-in-depth against the possibility of a fission product release from a fuel handling accident in the fuel building. The plant vent radiation detectors monitor effluents discharged from the plant vent to the environment.

Each VFS exhaust subsystem includes one 100 percent capacity exhaust air filtration unit, and the associated exhaust fan, heater and ductwork.

The filtration units are connected to a ducted system with isolation dampers to provide HEPA filtration and charcoal adsorption of exhaust air from the containment, fuel handling area, radiologically controlled areas of the auxiliary and annex buildings. A gaseous radiation monitor is located downstream of the exhaust air filtration units to

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(continued)

BASES

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BACKGROUND  
(continued)

provide an alarm if abnormal gaseous releases are detected. The plant vent exhaust flow is monitored for gaseous, particulate and iodine releases to the environment. During conditions of abnormal airborne radioactivity in the fuel handling area, auxiliary and/or annex buildings, the VFS exhaust subsystem provides filtered exhaust to minimize unfiltered offsite releases.

The VAS is described in Reference 1 and the VFS is described in Reference 2.

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APPLICABLE  
SAFETY ANALYSES

The VFS is not required to mitigate the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident does not assume that the VFS provides a filtered exhaust, and its operation would reduce the consequences of the fuel handling accident.

This specification is included for defense-in-depth.

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LCO

One VFS exhaust subsystem is required to be OPERABLE to reduce the consequences of a fuel handling accident by filtering the fuel building atmosphere.

A VFS exhaust subsystem is considered OPERABLE when its associated:

- a. Exhaust fan is capable of operating;
  - b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function;
  - c. The associated heater and ductwork are capable of operating.
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(continued)

BASES (continued)

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APPLICABILITY      During movement of irradiated fuel in the fuel handling area, one VFS exhaust subsystem is OPERABLE to alleviate the potential consequences of a fuel handling accident.

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ACTIONS            LCO 3.0.3 is applicable while in MODE 1, 2, 3, or 4. Since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3 while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

LCO 3.0.8 is applicable while in MODE 5 or 6. Since irradiated fuel assembly movement can occur in MODE 5 or 6, the ACTIONS have been modified by a Note stating that LCO 3.0.8 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, the fuel movement is independent of shutdown reactor operations. Entering LCO 3.0.8 while in MODE 5 or 6 would require the optimization of plant safety, unnecessarily.

A.1

When the required VFS exhaust subsystem is inoperable during movement of irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.6 1

Each VFS exhaust subsystem should be checked 31 days prior to fuel movement in the fuel handling area to ensure that it functions properly. As the operating conditions on this subsystem are not severe, testing each subsystem within one month prior to fuel movement provides an adequate check on this system. Operation of the heater dries out any moisture accumulated in the charcoal from humidity in the ambient air.

SR 3.9.6 2

This SR verifies that the VAS fuel handling area subsystem

aligns to the VFS and that the VFS exhaust subsystem starts and operates on an actual or simulated actuation signal. During the post-accident mode of operation, the VAS fuel handling area subsystem aligns to the VFS filtered exhaust subsystem. The 24 month Frequency is consistent with Reference 4.

SR 3.9.6.3

This SR verifies the integrity of the fuel handling area of the auxiliary building enclosure. The ability of the VAS and VFS to maintain negative pressure ( $\leq [-0.125]$  inches water gauge relative to outside atmospheric pressure) in the

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(continued)

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**BASES**

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**SURVEILLANCE  
REQUIREMENTS****SR 3.9.6 3 (continued)**

fuel handling area of the auxiliary building is periodically tested to verify proper function of the VAS and VFS exhaust subsystem. During this surveillance, the VAS fuel handling area subsystem is aligned to the operating VFS exhaust subsystem. The fan for the VAS fuel handling area subsystem is off. In this configuration, the VFS exhaust subsystem is designed to maintain a negative pressure in the fuel handling area of the auxiliary building ( $\leq [-0.125]$  inches water gauge relative to outside atmospheric pressure), to prevent unfiltered and unmonitored leakage. Doors may be opened for short periods of time to allow ingress and egress. During this surveillance, the VAS may be servicing the remaining portions of the auxiliary and annex buildings. The Frequency of 24 months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. 5).

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**REFERENCES**

1. Section 9.4.3, "Radiologically Controlled Area Ventilation System."
  2. Section 9.4.7, "Containment Air Filtration System."
  3. Section 15.7.4, "Fuel Handling Accident."
  4. Regulatory Guide 1.52 (Rev. 2).
  5. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.
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