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Perry Nuclear Power Plant
Docket No. 50-440
Submittal of Technical Specification Bases, Revision 4

Ladies and Gentlemen:

Pursuant to the requirements of Section 5.5.11.d of the Perry Nuclear Power Plant (PNPP) Technical Specifications, a copy of the Technical Specification Bases, Revision 4 is hereby submitted. This submittal reflects the changes made subsequent to the changes reported in the Revision 3 Technical Specification Bases submittal, which was provided by letter dated September 6, 2001. The changes are identified by sidebars.

If you have questions or require additional information, please contact Mr. Vernon K. Higaki, Manager - Regulatory Affairs, at (440) 280-5294.

Very truly yours,

Enclosure

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III

A001

TECHNICAL SPECIFICATIONS BASES
REVISION 4
LIST OF EFFECTIVE PAGES

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BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes less than two thirds of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and also to provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel, in order to prevent elevated clad temperatures and resultant clad perforation.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2 (continued)

with the SL within 2 hours. These actions will include restoring reactor vessel water level in accordance with the Plant Emergency Instructions (e.g., manually initiating the ECCS or depressurizing the reactor vessel). The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

Per 10 CFR 50.36(c)(1)(i)(A), operation must not be resumed until authorized by the Nuclear Regulatory Commission.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II" (latest approved revision).
3. 10 CFR 100.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the Winter of 1972 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. A weld overlay repair was performed on a feedwater nozzle to safe-end weld (1B13-N4C-KB), using a different Code Edition (Ref. 9), which did not affect this maximum transient pressure limit. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is currently designed to ASME Code, Section III, 1983 Edition, including addenda through the Winter of 1984 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping, 1650 psig for discharge piping between the pump and the discharge valve, and 1550 psig beyond the discharge valve. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.
SAFETY LIMITS	The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping, 1650 psig for discharge piping between the pump and the discharge valve, and 1550 psig beyond the discharge valve. The most limiting of these allowances is the 110% of the suction piping design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured in the reactor steam dome.
APPLICABILITY	SL 2.1.2 applies in all MODES.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. These actions will include restoring reactor vessel water level in accordance with the Plant Emergency Instructions (e.g., manually initiating the ECCS or depressurizing the reactor vessel). The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, and GDC 15.
2. ASME, Boiler and Pressure Vessel Code, Section III.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000.
4. 10 CFR 100.
5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, Winter of 1972.
6. ASME, Boiler and Pressure Vessel Code, 1983 Edition, Addenda, Winter of 1984.
7. Deleted
8. Deleted
9. USAR Table 3.2-7.

BASES

LCO 3.0.4 (continued)

provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in the other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

(continued)

BASES

LCO 3.0.6 (continued)

systems' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions. However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.10, "Safety Function Determination Program" (SFDP), ensures loss of safety function is detected and appropriate actions are taken. Upon failure to meet two or more LCOs concurrently, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

LCO 3.0.6 addresses support systems that have an LCO specified in the TS. For support systems that do not have an LCO specified in the TS, the following guidance applies.

In most cases, the non-TS support system has two subsystems, each supporting just one TS division of safety equipment. The duration of a maintenance activity on such a non-TS support system is limited by the Required Action Completion Times of the supported TS system(s). In this case, because

(continued)

BASES

LCO 3.0.6 (continued)

the outage time of the non-TS support system is limited by the supported system TSs, the plant is temporarily allowed to depart from the single-failure design criterion, but the licensee may not rely solely on the TS limitations. The licensee must still assess and manage risk in accordance with 10 CFR 50.65(a)(4).

In some cases, the non-TS support system has two redundant 100 percent capacity subsystems, each capable of supporting both TS divisions, e.g., M23/24, M28, M32, and P47. Loss of one support subsystem does not result in a loss of support for either division of TS equipment. Both TS divisions remain operable, despite a loss of support function redundancy, because the TS definition of operability does not require a TS subsystem's necessary support function to meet the single-failure design criterion. Thus, no TS limits the duration of the non-TS support subsystem outage, even though the single-failure design requirement of the supported TS systems is not met. However, by assessing and managing risk in accordance with 10 CFR 50.65(a)(4), the licensee can determine an appropriate duration for the maintenance activity. Use of administrative controls to implement such a risk-informed limitation is an acceptable basis for also allowing a temporary departure from the design-basis configuration during such maintenance. Although not expected, were a licensee to determine that its risk assessment would permit the support subsystem to be inoperable for more than 90 days, then the licensee would have to evaluate the maintenance configuration as a change to the facility under 10 CFR 50.59, including consideration of the single-failure design criterion.

For the unusual non-TS support system design configuration described, the preceding is a clarification of the previous staff position (GL 80-30) regarding when a temporary departure from the single-failure design criterion is allowed. This allowance would be permitted regardless of whether the maintenance is corrective or preventive.

When a non-TS support subsystem is unexpectedly found to be in a degraded or non-conforming condition, the licensee must make a prompt determination of operability (functionality), as discussed in Generic Letter 91-18. If the non-TS support subsystem is determined to be inoperable (non-functional), then the licensee must determine whether the subsystem's

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BASES

LCO 3.0.6 (continued)

support function is actually needed to support OPERABILITY of the TS supported systems. If the support function is required, then the risk-management strategies of the TS and 10 CFR 50.65(a)(4), as described above for planned maintenance, will determine the appropriate actions and time limits to return the non-TS subsystem to operable (functional) status. If the non-TS support function cannot be maintained, then enter the LCO(s) of the TS supported system(s).

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Special Operations LCOs in Section 3.10

(continued)

BASES

SR 3.0.2 (continued)

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. An example is in the Primary Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified

(continued)

BASES

SR 3.0.3 (continued)

Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met. This delay period provides adequate time to complete Surveillances that have been inadvertently missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions etc.) is discovered to not have been performed when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 also provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend

(continued)

BASES

SR 3.0.3 (continued)

Surveillance intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the inadvertently missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance.

The risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, 'Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.' This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide and in the standard which it endorses, NUMARC 93-01, Revision 3, 'Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.' The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

(continued)

BASES

LCO (continued)

satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in Shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert (using all available insertion methods) by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note that allows a stuck control rod to be bypassed in the Rod Action Control System (RACS) to allow continued operation. SR 3.3.2.1.9 provides additional requirements when control rods are bypassed in RACS to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, verification that the separation criteria

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

are met must be performed immediately. The stuck control rod separation criteria are that the stuck control rod may not occupy a location adjacent to a "slow" control rod. The description of "slow" control rods is provided in LCO 3.1.4 "Control Rod Scram Times". In addition, the control rod must be disarmed within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CRDM. The control rod can be

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

isolated from scram by isolating the hydraulic control unit from scram and normal drive and withdraw pressure, yet still maintain cooling water to the CRD. A control rod can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rod can be disarmed by disconnecting power from all four directional control valve solenoids.

Monitoring of the insertion capability for each withdrawn control rod must also be performed within 24 hours. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. The allowed Completion Time of 24 hours provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. Required Action A.2 has a modified time zero Completion Time. The 24 hour Completion Time for this Required Action starts when the withdrawn control rod is discovered to be stuck and THERMAL POWER is greater than the actual low power setpoint (LPSP) of the rod pattern controller (RPC), since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RPC (LCO 3.3.2.1, "Control Rod Block Instrumentation").

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod

(continued)

BASES

ACTIONS

A.1, A.2, A.3, and A.4 (continued)

also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 7).

B.1

With two or more withdrawn control rods stuck, the plant should be brought to MODE 3 within 12 hours. Isolating the control rod from scram prevents damage to the CRDM. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. With a control rod not coupled to its associated drive mechanism, insert the control rod drive mechanism to accomplish recoupling. Verify recoupling by withdrawing the control rod and observing any indicated response of the nuclear instrumentation and demonstrating that the control rod drive will not go to the overtravel position. Required Action C.1 is modified by a Note that allows control rods to be bypassed in the RACS if required to allow insertion of the inoperable control rods and continued operation. SR 3.3.2.1.9 provides additional requirements when the control rods are bypassed to ensure compliance with the CRDA analysis.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

D.1 and D.2

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 19.0% RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 7) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Required Action D.1 is utilized when the control rods that are violating the generic BPWS separation criteria cannot be restored to an OPERABLE condition. Required Action D.1, "Restore compliance with BPWS", means to provide an analysis which demonstrates that the control rod worths of a proposed or existing rod pattern are no more than the control rod worths determined in the generic BPWS analysis. Under Required Action D.1, even after compliance with BPWS is restored through new analysis, the control rods remain inoperable per LCO 3.1.6 unless they can be moved to meet the generic separation criteria (see Required Action D.2). Required Action D.2, "Restore control rod to OPERABLE status", means to move one or both control rods back into pattern such that they can be re-declared OPERABLE, or, if the rod is inoperable for reasons other than a pattern deviation, resolve that inoperability and then move the rod to be in compliance with the generic BPWS analysis.

A Note has been added to the Condition to clarify that the Condition is not applicable when > 19.0% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

(continued)

BASES

ACTIONS (continued)

E.1

If any Required Action and associated Completion Time of Condition A, C, or D are not met or nine or more inoperable control rods exist, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above 19.0% RTP (i.e., no CRDA considerations) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Control rod patterns analyzed in Reference 2 follow the generic banked position withdrawal sequence (BPWS) analysis described in Reference 8. The BPWS is applicable from the condition of all control rods fully inserted to 19.0% RTP (Ref. 1). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are defined to minimize the maximum incremental control rod worths without being overly restrictive during normal plant operation. The generic BPWS analysis (Ref. 8) also evaluated the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.

Rod pattern control satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the generic BPWS analysis. This LCO only applies to OPERABLE control rods. For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.

APPLICABILITY

In MODES 1 and 2, when THERMAL POWER is ≤ 19.0% RTP, the CRDA is a Design Basis Accident (DBA) and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > 19.0% RTP, there is no credible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 1). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

With one or more OPERABLE control rods not in compliance with the prescribed control rod sequence, action may be taken to either correct the control rod pattern or declare the associated control rods inoperable within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, rods inserted for the purpose of suppressing a fuel defect, conducting single rod scram time testing (Ref. 9), or a power reduction to ≤ 19.0% RTP before establishing the correct control rod pattern. The number of OPERABLE control rods not in compliance with the prescribed sequence is limited to eight to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the control rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note, which allows control rods to be bypassed in Rod Action Control System (RACS) in accordance with SR 3.3.2.1.9 to allow the affected control rods to be returned to their correct position. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. OPERABILITY of control rods is determined by compliance with LCO 3.1.3; LCO 3.1.4, "Control Rod Scram Times"; and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than

(continued)

BASES

REFERENCES (continued)

5. 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

9. USAR 7.6.1.5.C.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in the USAR, Chapters 4, 6, and 15, and in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operations that determine APLHGR limits are presented in USAR, Chapters 4, 6, and 15, and in References 1, 2, 3, and 4.

APLHGR limits are developed as a function of exposure and the various operating core flow and power states. Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 5) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. $MAPFAC_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram signals are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 23.8% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 6. The ECCS/LOCA analysis assumes the existence of MAPFAC.

LOCA analyses are then performed to ensure that the above determined APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A discussion of the analysis code is provided in Reference 7. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, the MAPFAC multiplier is limited to a maximum value which is specified in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO	The APLHGR limits specified in the COLR are a function of exposure and are a result of DBA analyses. For two recirculation loops operating, the limit is determined by multiplying the smaller of the $MAPFAC_f$ and $MAPFAC_p$ factors times the exposure dependent APLHGR limits. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by the smallest of $MAPFAC_f$, $MAPFAC_p$, and the limiting value specified for single recirculation loop operation in the COLR, which has been determined by a specific single recirculation loop analysis (Ref. 2).
APPLICABILITY	The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 4.7% to 14.2% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function provides rapid scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < 23.8% RTP, the reactor operates with substantial margin to the APLHGR limits; thus, this LCO is not required.
ACTIONS	A.1 If any APLHGR exceeds the required limit, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limit(s) such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limit and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 4, 5, and 6).

Flow dependent MCPR limits ($MCPR_f$) are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 7). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. 8). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 23.8% RTP and the previously mentioned bypass power level.

Pressure Regulator Out of Service (PROOS) option is an analysis using the Pressure Regulator Downscale Failure (PRDF) at off-rated conditions. At full power, the PRDF is bounded by other pressurization transients. However, as the reactor power at the beginning of the transient decreases, the impact of the PRDF to MCPR increases.

During a PRDF transient, the pressure regulator closes the turbine control valves. This increases pressure, which increases power in the reactor. When the reactor is at full power, the pressure and power increases quickly, causing a SCRAM. As the reactor power is decreased, the power is further from the SCRAM setpoint so it takes more time to SCRAM. This longer time to SCRAM increases the amount of specific heat in the fuel and impacts the CPR. There is a range of initial reactor power where the CPR is no longer bounded by the normal $MCPR_p$ limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	There are two independent channels in the pressure regulating system and the PRDF transient is not applicable when both channels are operable. The COLR identifies the range of the modified MCPR limits and the new limits. These limits may be incorporated by either a revision to the monitoring system or appropriate administrative limits. The MCPR satisfies Criterion 2 of the NRC Policy Statement.
LCO	The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.
APPLICABILITY	The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23.8% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23.8% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in USAR Chapters 6 and 15.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in the USAR, Chapters 4, 6, and 15, and in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 1).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during AOOs for operation with LHGR up to the operating limit LHGR specified in the COLR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

The LHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs(Refs. 3 and 4). Flow dependent Thermal-Mechanical LHGR Limits are determined using the three dimensional BWR simulator code (Ref. 5) to analyze slow flow runout transients. The flow dependent multiplier for the Thermal-Mechanical LHGR Limits is dependent on the maximum core flow runout capability. Thermal-Mechanical LHGR Limit curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

The LHGR limits are primarily derived from fuel design evaluations and transient analyses that are assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required LHGR limits increases. This trend continues down to the power range of 4.7% to 14.2% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function provides rapid scram initiation during any significant transient, thereby effectively removing any LHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels < 23.8% RTP, the reactor operates with substantial margin to the LHGR limits; thus, this LCO is not required

The LHGR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

(continued)

BASES (continued)

APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At THERMAL POWER levels < 23.8% RTP, the reactor is operating with substantial margin to the LHGR limits and this LCO is not required.

ACTIONS

A.1

If any LHGR exceeds the required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action is taken to restore the LHGR(s) to within required limit(s) such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the LHGR(s) to within its limit and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limit within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23.8% RTP within 4 hours. The allowed

(continued)

BASES

ACTIONS	B.1 (continued) Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23.8% RTP in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.2.3.1 The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23.8% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 23.8% RTP is achieved, is acceptable given the large inherent margin to operating limits at lower power levels.
REFERENCES	1. NUREG-0800, "Standard Review Plan," Section 4.2, II.A.2(g), Revision 2, July 1981. 2. USAR, Chapter 15, Appendix 15B. 3. USAR, Chapter 15, Appendix 15F. 4. USAR, Chapter 15, Appendix 15E. 5. NEDO-30130-P-A, "Steady State Nuclear Methods," April 1985.

BASES

ACTIONS

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

would be HPCS. For Required Action B.1, redundant automatic initiation capability is lost if either (a) one or more Function 1.a channels and one or more Function 2.a channels are inoperable and untripped, or (b) one or more Function 1.b channels and one or more Function 2.b channels are inoperable and untripped.

For Divisions 1 and 2, since each inoperable channel would have Required Action B.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated Division of low pressure ECCS, DG and AEGT System to be declared inoperable. However, since channels in both Divisions are inoperable and untripped, and the Completion Times started concurrently for the channels in both Divisions, this results in the affected portions in both Divisions of ECCS, DG and AEGT System being concurrently declared inoperable.

For Required Action B.2, redundant automatic initiation capability is lost if two Function 3.a or two Function 3.b channels are inoperable and untripped in the same trip system. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the feature(s) associated with the inoperable, untripped channels must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1 and Required Action B.2), the two Required Actions are only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Although a total loss of initiation capability for 24 hours is allowed by Required Action B.3 during MODES 4 and 5, additional controls are imposed in ORM 6.2.9. Notes are also provided (Note 2 to Required Action B.1 and Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed.

(continued)

ACTIONS (continued)

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function (or in some cases, within the same monitored parameter) result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions 1.c, 1.d, 1.e, 2.c, and 2.d (i.e., low pressure ECCS). For Functions 1.c and 2.c, redundant automatic initiation capability is lost if the Function 1.c and Function 2.c channels are inoperable. For Functions 1.d, 1.e, and 2.d, redundant automatic initiation capability is lost if the Function 1.d and 1.e channels and the Function 2.d channels are inoperable. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated Division to be declared inoperable. However, since channels in both Divisions are inoperable, and the Completion Times started concurrently for the channels in both Divisions, this results in the affected portions in both Divisions being concurrently declared inoperable. For Functions 1.c and 2.c, the affected portions of the Division are LPCI A and LPCI B, respectively. For Functions 1.d, 1.e, and 2.d, the affected portions of the Division are the low pressure ECCS pumps (Divisions 1 and 2, respectively).

In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. As noted (Note 1), the Required Action is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Although a total loss of automatic initiation capability for 24 hours is allowed by Required Action(s) B.3 and C.2 during MODES 4 and 5, additional controls are imposed in ORM 6.2.9.

Note 2 states that Required Action C.1 is only applicable for Functions 1.c, 1.d, 1.e, 2.c, and 2.d. The Required Action is not applicable to Functions 1.h, 2.f, and 3.h (which also require entry into this Condition if a channel in these Functions is inoperable), since they are the Manual Initiation Functions and are not assumed in any accident or

(continued)

ACTIONS

E.1 and E.2 (continued)

channels within the LPCS and LPCI Pump Discharge Flow-Low (Bypass) Functions result in redundant automatic initiation capability being lost for the feature(s). For Required Action E.1, the features would be those that are initiated by Functions 1.f, 1.g, and 2.e (e.g., low pressure ECCS). Redundant automatic initiation capability is lost if three of the four channels associated with Functions 1.f, 1.g, and 2.e are inoperable. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected low pressure ECCS pump to be declared inoperable. However, since channels for more than one low pressure ECCS pump are inoperable, and the Completion Times started concurrently for the channels of the low pressure ECCS pumps, this results in the affected low pressure ECCS pumps being concurrently declared inoperable.

In this situation (loss of redundant automatic initiation capability), the 7 day allowance of Required Action E.2 is not appropriate and the feature(s) associated with each inoperable channel must be declared inoperable within 1 hour after discovery of loss of initiation capability for feature(s) in both Divisions. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Although a total loss of initiation capability for 7 days is allowed by Required Action E.2 during MODES 4 and 5, additional controls are imposed in ORM 6.2.9. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to low pressure ECCS Functions. Required Action E.1 is not applicable to HPCS Functions 3.f and 3.g since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 4 and considered acceptable for the 7 days allowed by Required Action E.2.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action E.1, the Completion Time only begins upon discovery that three channels of the Function (Pump

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1.6 (continued)

Surveillance under the conditions that apply during a plant outage and the potential for unplanned transients if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. USAR, Section 5.2.
2. USAR, Section 6.3.
3. USAR, Chapter 15.
4. NEDC-30936-P-A, "BWR Owners' Group Technical Specification Improvement Analyses for ECCS Actuation Instrumentation, Part 2," December 1988.
5. Plant Data Book, Tab R, Section 6.2.9.

BASES

BACKGROUND (continued)

The primary containment and drywell isolation instrumentation has inputs to the trip logic from the isolation Functions listed below.

1. Main Steam Line Isolation

Most Main Steam Line Isolation Functions receive inputs from four channels. The outputs from these channels are combined in one-out-of-two taken twice logic to initiate isolation of all main steam isolation valves (MSIVs). The outputs from the same channels are arranged into two two-out-of-two logic trip systems to isolate all MSL drain valves. One logic trip system isolates one valve in the inboard MSL drain line. The other trip system isolates the second valve in the inboard MSL drain line.

The exception to this arrangement is the Main Steam Line Flow-High Function. This Function uses 16 flow channels, four for each steam line. If one or more of the 16 flow channels are inoperable, then the appropriate Conditions and Required Actions must be entered. One flow channel from each steam line inputs to one of four trip channels. Two trip channels make up each trip system, and both trip systems must trip to cause an MSL isolation. Each trip channel has four inputs (one per MSL), any one of which will trip the trip channel. The trip channels within a trip system are arranged in a one-out-of-two taken twice logic. Therefore, this is effectively a one-out- of-eight taken twice logic arrangement to initiate isolation of the MSIVs. Similarly, the 16 flow channels are connected into two two-out-of-two logic trip systems (effectively, two one- out-of-four twice logic). One logic trip system isolates one valve in the inboard MSL drain line. The other trip system isolates the second valve in the inboard MSL drain line.

2. Primary Containment and Drywell Isolation

Each Primary Containment Isolation Function receives inputs from four channels. The outputs from these channels are arranged into two two-out-of-two logic trip systems. One trip system initiates isolation of all inboard PCIVs and drywell isolation valves, while the other trip system initiates isolation of all outboard PCIVs. Each trip system

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.c. Main Steam Line Flow-High (continued)

ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits (for the design-basis Revised Accident Source Term (RAST) LOCA analysis, the licensing basis offsite dose limit is 25 rem TEDE (Ref. 11)).

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from each other, all four connected to one steam line would be able to detect the high flow. Four channels of Main Steam Line Flow-High Function for each unisolated MSL (two channels per trip system) are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL. If one or more of the 16 flow transmitters or associated flow channels are inoperable, then the appropriate Conditions and Required Actions must be entered.

The Allowable Value is chosen to ensure that offsite dose limits are not exceeded due to the break.

This Function isolates the Group 6 valves.

1.d. Condenser Vacuum-Low

The Condenser Vacuum-Low Function is provided to prevent overpressurization of the main condenser in the event of a loss of the main condenser vacuum. Since the integrity of the condenser is an assumption in offsite dose calculations, the Condenser Vacuum-Low Function is assumed to be OPERABLE and capable of initiating closure of the MSIVs. The closure of the MSIVs is initiated to prevent the addition of steam that would lead to additional condenser pressurization and possible rupture of the diaphragm installed to protect the turbine exhaust hood, thereby preventing a potential radiation leakage path following an accident.

Condenser vacuum pressure signals are derived from four pressure transmitters that sense the pressure in the condenser. Four channels of Condenser Vacuum-Low Function are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to prevent damage to the condenser due to pressurization, thereby ensuring its integrity for offsite dose analysis. As noted (footnote (a) to Table 3.3.6.1-1), the channels are not required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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

This Function isolates the Group 2 isolation valves.

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

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

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

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

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

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 100 limits (for the design-basis Alternative Source Term (AST) LOCA and fuel handling accident analyses, the licensing basis offsite dose limits are 25 rem TEDE and 6.3 rem TEDE, respectively (Ref. 11 and 13)).

The Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery) must be provided to ensure offsite dose limits are not exceeded. However, OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 12).

These Functions isolate the Group 8 valves.

2.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.h. Manual Initiation (continued)

are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific USAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

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

Four channels of the Manual Initiation Function are required to be OPERABLE in MODES 1, 2, and 3, and during OPDRVs or movement of recently irradiated fuel assemblies in primary containment, since these are the MODES in which the Primary Containment and Drywell Isolation automatic Functions are required to be OPERABLE. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 12).

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

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Core Isolation Cooling System Isolation

3.a. RCIC Steam Line Flow-High

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

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

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

(continued)

ACTIONS

A.1 (continued)

tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue with no further restrictions. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Action taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in all automatic isolation capability being lost for the associated penetration flow path(s). A "Function" is a line item in Table 3.3.6.1-1, e.g., main steam line isolation from line item 1.a., "Reactor Vessel Water Level - Low Low Low, Level 1." The MSL isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that both trip systems will generate a trip signal from the given Function on a valid signal. The other isolation Functions are considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that at least one automatic isolation valve in the associated penetration flow path(s) can receive an isolation signal from the given Function. Note that some penetrations only have a single automatic isolation valve or do not have redundant isolation channels, in which case the isolation capability is maintained if those single valves can receive an isolation signal from the given Function.

Also, some Functions consist of channels that monitor several different locations. Two examples are Function 1.c, Main Steam Line Flow - High, which needs to maintain the isolation capability for each of the four main steam lines, and Function 3.h, RHR Equipment Area Ambient Temperature - High, which needs to maintain the isolation capability based on monitoring ambient temperature in both RHR Equipment Areas #1 and #2. Condition B does not include the Manual Initiation Functions (Functions 1.h,

(continued)

ACTIONS

B.1 (continued)

2.h, 3.k, 4.1, and 5.e), since they are not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

It is important to note that when determining Technical Specification compliance for each specific associated trip system and channel, the Technical Specification Bases should not be the sole document referenced. Other source documents should be referenced-such as the Plant Data Book, plant procedures, and controlled mechanical and electrical schematic drawings.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and if allowed (i.e., plant safety analysis allows operation with an MSL isolated), plant operation with the MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.7 (continued)

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The semiannual Frequency is based on the reduced drift and the design features inherent in digital systems (Ref. 10).

For Function 1.e, "Main Steam Line Pipe Tunnel Temperature High", this SR is applicable only to the Division 1 and 2 ambient temperature channels. Divisions 3 and 4 are monitored by analog instrument channels, which are functionally tested on a quarterly basis by SR 3.3.6.1.2.

REFERENCES

1. USAR, Section 6.3.

2. USAR, Chapter 15.

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

4. USAR, Section 9.3.5.

5. NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," June 1989.

6. NEDC-30851-P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

7. USAR, Section 15.1.3.

8. USAR, Section 15.6.

9. NEDO-32291, "System Analyses for Elimination of Selected Response Time Testing Requirements," January 1994.

10. Letter PY-CEI/NRR-1654L, "License Amendment Request: Replacement of Selected Analog Leak Detection System Instruments with GE NUMAC Leak Detection Monitors," November 22, 1993.

(continued)

BASES

REFERENCES (continued)

11. Amendment No. 103 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1.

12. USAR, Section 15.7.6

13. Amendment No. 122 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.3.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 3.

SR 3.3.6.3.3

The calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.6.3-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.

The Frequency of 92 days is based on the reliability analysis of Reference 3.

SR 3.3.6.3.4 and SR 3.3.6.3.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.6.3.4 and SR 3.3.6.3.5 are based on the assumption of the magnitude of equipment drift in the setpoint analysis.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Recirculation (CRER) System Instrumentation

BASES

BACKGROUND

The CRER System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent CRER subsystems are each capable of fulfilling the stated safety function. The instrumentation and controls for the CRER System automatically initiate action to isolate the main control room to minimize the consequences of radioactive material in the control room environment.

In the event of a Reactor Vessel Water Level-Low Low Low, Level 1, Drywell Pressure-High, or Control Room Ventilation Radiation Monitor signal, the CRER System is automatically started in the emergency recirculation mode. The control room air is then recirculated through the charcoal filter to reduce the concentration of airborne radioactive contaminants.

The CRER System instrumentation has two trip systems: one trip system initiates one CRER subsystem, while the second trip system initiates the other CRER subsystem (Ref. 1). Each trip system receives input from the Functions listed above. The Functions are arranged as follows for each trip system. The Reactor Vessel Water Level-Low Low Low, Level 1 and Drywell Pressure-High are arranged together in a one-out-of-two taken twice logic. The Control Room Ventilation Radiation Monitor is arranged in a one-out-of-one logic. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a CRER System initiation signal to the initiation logic.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CRER System to maintain the habitability of the control room is explicitly assumed for certain accidents as discussed in the USAR safety analyses (Refs. 2 and 3). CRER System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A (for the design-basis Alternative Source Term (AST) LOCA and fuel handling accident analyses, the licensing basis Control Room dose limit is 5 rem TEDE (Ref. 7 and 8)).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Control Room Ventilation Radiation Monitor (continued)

that control room personnel are protected during a LOCA, fuel handling event involving recently irradiated fuel, or a vessel draindown event. Due to radioactive decay, handling of fuel only requires OPERABILITY of this Function when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 9). OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, this Function is not required to be OPERABLE. During MODES 4 and 5, when these specified conditions are not in progress (e.g., OPDRVs), the probability of a LOCA or significant fuel damage is low; thus, the Function is not required.

ACTIONS

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

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent.

(continued)

ACTIONS

A.1 (continued)

Each time an inoperable channel is discovered, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

B.1 and B.2

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

(continued)

ACTIONS

B.1 and B.2 (continued)

provided the associated Function is still maintaining CRER System initiation capability. A Function is considered to be maintaining CRER System initiation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate an initiation signal from the given Function on a valid signal. This would require one trip system to have two channels, each OPERABLE or in trip. In this situation (loss of CRER System initiation capability), the 24 hour allowance of Required Action B.2 is not appropriate. If the Function is not maintaining CRER System initiation capability, both CRER subsystems must be declared inoperable within 1 hour of discovery of loss of CRER System initiation capability in both trip systems. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition D must be entered and its Required Actions taken.

C.1 and C.2

The control room ventilation radiation monitor signal is considered a diverse signal from the redundant Drywell Pressure-High, and Reactor Vessel Water Level-Low Low Low, Level 1 signals. Because of this, the control room ventilation radiation monitor is not required to be redundant. Therefore, a Completion Time of 7 days is provided to permit restoration of the inoperable channel to OPERABLE status as long as an alternate means of monitoring the control room atmosphere for radiation has been established in the first 24 hours. Acceptable alternate means include a portable continuous noble gas monitor or the control room area radiation monitor. If the inoperable channel cannot be restored to OPERABLE status within the Completion Time, Condition D must be entered and its Required Actions taken.

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 7.3.1.1.7.

2. USAR, Section 6.4.

3. USAR, Chapter 15.

4. GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

7. Amendment No. 103 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1; and Letter, D. Pickett (NRC) to L. Myers (FENOC), "Issuance of Exemption from 10 CFR Part 50, Appendix A, General Design Criterion 19", dated March 26, 1999.

8. Amendment No. 122 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1.

9. USAR, Section 15.7.6

BASES

ACTIONS (continued)

E.1 and E.2

If any Required Action and associated Completion Time of Condition A or B are not met in MODE 4 or 5, with the RHR Shutdown Cooling System not isolated, the operator must immediately initiate action to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path can be isolated).

Alternately, action must be immediately initiated to either restore one electric power monitoring assembly to OPERABLE status for the inservice power source supplying the required instrumentation powered from the RPS bus (Required Action E.2). Required Action E.2 is provided because the RHR Shutdown Cooling System may be needed to provide core cooling. All actions must continue until the applicable Required Actions are completed.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2.1

A CHANNEL FUNCTIONAL TEST is performed on each overvoltage, undervoltage, and underfrequency channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted in the Surveillance, the CHANNEL FUNCTIONAL TEST is only required to be performed while the plant is in a condition in which the loss of the RPS bus will not jeopardize steady state power operation (the design of the system is such that the power source must be removed from service to conduct the Surveillance). The 24 hours is intended to indicate an outage of sufficient duration to allow for scheduling and proper performance of the Surveillance. The 184 day Frequency and the Note in the Surveillance are based on guidance provided in Generic Letter 91-09 (Ref. 2). This surveillance can be performed under other plant conditions provided that the risk is evaluated pursuant to 10 CFR 50.65(a)(4) (Ref. 3).

SR 3.3.8.2.2

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2.2 (continued)

The Frequency is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.2.3

Performance of a system functional test demonstrates a required system actuation (simulated or actual) signal. The logic of the system will automatically trip open the associated power monitoring assembly circuit breaker. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E circuit breakers is included as part of this test to provide complete testing of the safety function. If the breakers are incapable of operating, the associated electric power monitoring assembly would be inoperable.

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 an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. USAR, Section 8.3.1.1.5.

2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electric Protective Assemblies in Power Supplies for the Reactor Protection System."

3. 10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figure 3.4.11-1 contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing. The heatup curve provides limits for both heatup and criticality. Curves are provided which are valid for up to 22 EFPY and 32 EFPY.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185-82 (Ref. 3) and 10 CFR 50, Appendix H

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.10 (continued)

material specimens, in accordance with ASTM E 185-82 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves in Figure 3.4.11-1 will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests For Light-Water Cooled Nuclear Power Reactor Vessels," July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. GE-NE-0000-0000-8763-01, Revision 0, "Pressure-Temperature Curves For FirstEnergy Corporation, Using the K_{Ic} Methodology, Perry Unit 1," April 2002.
8. USAR, Section 15.4.4.
9. GE Services Information Letter, SIL No. 517 Supplement 1, "Analysis Basis for Idle Recirculation Loop Startup."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.7 (continued)

passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ADS valves divert steam flow upon opening. Sufficient time is therefore allowed, after the required pressure and flow are achieved, to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer. Reactor startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual activation after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. SR 3.5.1.6 and the LOGIC SYSTEM FUNCTIONAL TEST performed in SR 3.3.5.1.6 overlap this Surveillance to provide complete testing of the assumed safety function.

The Frequency of 24 months on a STAGGERED TEST BASIS ensures that the solenoids for each ADS valve are alternately tested. The 24 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 14). The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

SR 3.5.1.8

This SR ensures that the ECCS RESPONSE TIMES are within limits for each of the ECCS injection and spray subsystems. This SR is modified by a note which identifies that the associated ECCS actuation instrumentation is not required to be response time tested. Response time testing of the remaining subsystem components is required. This is supported by Reference 15. Response time testing acceptance criteria are included in Reference 16.

ECCS RESPONSE TIME tests are conducted every 24 months. The 24 month Frequency is based on the need to perform this

(continued)

BASES (continued)

LCO

Primary containment OPERABILITY is maintained by limiting leakage to < 1.0 L_a, except prior to the first unit startup after performing a required leakage test in accordance with the Primary Containment Leakage Rate Testing Program. At this time, the applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Individual leakage rates specified for the primary containment air locks are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment leakage limits are not required to be met in MODES 4 and 5 to prevent leakage of radioactive material from primary containment, (refer to LCO 3.6.1.10, "Primary Containment-Shutdown").

ACTIONS

A Note has been provided to indicate that when the Inclined Fuel Transfer System (IFTS) blind flange is unbolted for removal or re-installation, entry into associated Conditions and Required Actions may be delayed for up to 20 hours per 12 month period. This note only applies to the IFTS penetration and not to any other Primary Containment penetration. During removal and re-installation of the blind flange, a temporary condition will exist where the bolting will be loosened, hydraulic jacks will spread the flange faces, and normally about one half of the bolts will be removed while the blind is rotated. This condition is expected to exist for no more than 20 hours (10 hours to rotate out the blind and an additional 10 hours to re-install the blind). Upon expiration of the 20 hour allowance for this maintenance activity, if the IFTS blind flange has not yet been re-bolted, the applicable Condition must be entered and the Required Actions taken. With the bolts removed, the seismic restraint for the IFTS penetration is potentially challenged. The risk is to the bellows assembly, as exact displacements are not quantified. Failure of the ASME Class 2 bellows could result in a potential bypass of

(continued)

ACTIONS (continued)

Containment. This Note is based on a risk analysis (Ref. 9) of the time required to perform IFTS blind flange removal or installation. That analysis demonstrated that a 20 hour allowance per 12 month period does not significantly reduce the probability that the Primary Containment will be OPERABLE when necessary. Therefore, the total number of hours that the blind flange is unbolted per 12 month period shall be tracked to ensure the 20 hour assumption in the risk analysis is maintained. The 20 hour duration conservatively limits the seismic risk associated with the unbolted IFTS flange, yet provides adequate time to complete flange rotation.

A.1

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

B.1 and B.2

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Primary Containment Leakage Rate Testing Program. Failure to meet air lock leakage testing (SR 3.6.1.2.1 and SR 3.6.1.2.4), secondary containment bypass leakage (SR 3.6.1.3.9), resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.6), main steam isolation valve leakage (SR 3.6.1.3.10), or hydrostatically tested valve leakage (SR 3.6.1.3.11) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of the Primary Containment Leakage Rate Testing Program. The Frequency is required by the Primary Containment Leakage Rate Testing Program. An one-time exception to NEI 94-01, Section 9.2.3 has been approved by the NRC such that the first Type A test performed after the July, 1994 Type A test must be completed no later than June 29, 2009 (Reference 10).

The Appendix J, Option A exemptions approved to date are listed below. Appendix J, Option A exemptions that are applicable to Appendix J, Option B, may be utilized for Appendix J, Option B testing, unless they have been specifically revoked by the NRC (Ref. 3). Additionally, Bechtel Topical Report BN-TOP-1 may be utilized for ILRTs with a duration of less than 24 hours as noted in Reference 5 and in the Primary Containment Leakage Rate Testing Program.

a. Section III.D.2(b)(ii) - The air lock seal leakage test of Section III.D.2(b)(iii) of Appendix J may be substituted (following normal air lock door opening) for the full-pressure test provided that no maintenance has been performed that would affect the air locks sealing capability (Reference 6).

b. Section III.D.3 - A one time schedular Exemption was issued to permit Type C testing of certain containment isolation valves to exceed the two year interval, so that these tests could by conducted during the first refueling outage (Reference 7).

(continued)

BASES

REFERENCES (continued)

6. PNPP Safety Evaluation Report Supplement 7, Section 6.2.6 "Containment Leakage Testing," November 1985.

7. Letter from NRC (T. Colburn) to CEI (A. Kaplan), "Exemption from 10 CFR Part 50, Appendix J", dated January 22, 1988.

8. Letter from NRC (J. Hopkins) to Centerior Services Company (D. Shelton), "Issuance of Exemption from the Requirements of 10 CFR Part 50, Appendix J - Perry Nuclear Power Plant, Unit 1", dated December 4, 1995.

9. Letter PY-CEI/NRR-2614L, "License Amendment Request Pursuant to 10 CFR 50.90: Inclined Fuel Transfer System (IFTS)," March 14, 2002.

10. Letter from NRC (S. Sands) to FENOC (W. Kanda), "Issuance of Amendment 126", dated April 8, 2003.

BASES

BACKGROUND (continued)

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.20% by weight of the containment and drywell air per 24 hours at the calculated maximum peak containment pressure (P_a) of 7.80 psig. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the intermediate building.

Primary containment air locks satisfy Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the air locks. However, it was determined that a Specification should remain in place per Criterion 4 to address operations with the potential for draining the reactor vessel (OPDRVs) and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the air locks during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

LCO

As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air locks are required to be OPERABLE. For each 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

(continued)

BASES

LCO (continued)	allows only one air lock door to be open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single OPERABLE door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from primary containment.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining OPERABLE primary containment air locks in MODE 4 or 5 to ensure a control volume is only required during situations for which significant releases of radioactive material can be postulated; such as during operations with a potential for draining the reactor vessel (OPDRVs), or during movement of recently irradiated fuel assemblies in the primary containment. OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the primary containment air locks are not required to be OPERABLE. Due to radioactive decay, handling of fuel only requires primary containment air lock OPERABILITY when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. However, even though the air locks are not required to be OPERABLE during handling of fuel that is not recently irradiated, there are still controls provided to ensure the ability to close a door in an air lock should the need arise. Closure of a door, even though it is not OPERABLE, would reduce the potential for gross unfiltered leakage. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 4).
ACTIONS	The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component.

(continued)

BASES

ACTIONS (continued)

If the outer door is inoperable, then it may be easily accessed for most repairs. If the inner door is the one that is inoperable, then 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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

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

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

PCIVs satisfy Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the primary containment. However, it was determined that Specifications should remain in place per Criterion 4 to address operations with the potential for draining the reactor vessel (OPDRVs) and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the primary containment during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

(continued)

BASES (continued)

LCO

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

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated automatic valves are required to actuate on an automatic isolation signal. Primary containment purge supply and exhaust valves are not qualified to close under accident conditions and therefore must be sealed closed (inboard) or blocked to prevent full opening (outboard valves) to be OPERABLE.

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

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

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE and the primary containment purge valves are not required to be sealed closed in MODES 4 and 5. Certain valves are required to be OPERABLE, however, to prevent inadvertent reactor vessel draindown and release of radioactive material during a postulated fuel handling

(continued)

BASES

APPLICABILITY (continued)

accident involving handling of recently irradiated fuel. These valves are those whose associated instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.) Due to radioactive decay, handling of fuel only requires containment isolation valve OPERABILITY when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 5).

ACTIONS

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

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SURVEILLANCE REQUIREMENT

SR 3.6.1.3.1 (continued)

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

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

SR 3.6.1.3.2

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.3 (continued)

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

SR 3.6.1.3.4

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

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

A fourth Note addresses removal of the Inclined Fuel Transfer System (IFTS) blind flange in MODES 1, 2, and 3 for up to 60 days per cycle. The 60 days per operating cycle is

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BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.4 (continued)

a risk-informed duration that provides the option of performing testing and maintenance of the IFTS during MODES 1, 2 or 3 prior to an outage. However, it is not meant for the movement of fuel. Removal of the IFTS blind flange during MODES 1, 2 or 3 requires the upper pool IFTS gate to be installed and requires the Fuel Handling Building Fuel Transfer Pool water level to be ≥ 40' above the bottom of the pool which ensures sufficient submergence of water over the bottom gate valve in the transfer tube to prevent direct communication between the Containment Building atmosphere and the Fuel Handling Building atmosphere, even upon occurrence of the peak post-accident pressure, P_a. Forty feet (40') above the bottom of the pool is equivalent to 22' 8 ¼" above the top of the flange for the IFTS bottom gate valve, which is approximately 3' 10" more water than needed to counteract the peak accident pressure of 7.8 psig. Also, since the IFTS drain piping does not have the same water seal as the transfer tube, administrative controls are required to ensure that the drain flow path can be quickly isolated whenever necessary.

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

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BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.4 (continued)

The upper Containment pool gate (both inner and outer gates) between the IFTS pool and the dryer storage pool is required to be installed prior to IFTS blind flange removal during MODES 1, 2 or 3. With this gate installed, should a failure of an IFTS tube component occur the amount of water drained to the lower pools will be limited. Therefore, installing the upper pool IFTS gate provides single failure protection of upper pool water inventory for supporting the SPMU system. If the IFTS gate was not installed, the potential would exist to drain the upper pool volume, reducing the inventory available to the SPMU system to support make up to the suppression pool, which supports the ECCS design function during a LOCA. Reduced suppression pool volume and increased suppression pool temperature could result in a subsequent loss of suction pressure for the ECCS.

Also, to account for the upper containment pool water loss that would result from all leakage sources, including leakage through the upper Containment pool gate and leakage through the Fuel Pool Cooling and Clean-up (FPCC) siphon breaker supply lines; when the IFTS blind flange is removed in MODES 1,2 or 3, the upper containment pool level shall be maintained at ≥ 22 ft - 9 inches; and to account for possible leakage, the suppression pool is to be raised to ≥ 17 ft - 11.7 inches. These levels were determined via engineering calculation. Also, as a leakage prevention measure, the fuel transfer and storage pool supply isolation valve (G41-F0524) shall be closed to isolate the normal flow of FPCC supply water to the IFTS pool area.

Additional regulatory commitments to the NRC are required when the IFTS blind flange is removed in MODES 1, 2 or 3. These prerequisite administrative controls are controlled by plant procedures and are 1) the lower fuel transfer pool gates must be removed, and 2) Fuel Handling Building closure shall be in effect. Removal of the lower fuel transfer pool gates ensures control room monitoring exists for spent fuel pool level, which would assist in detecting a change in the fuel transfer pool water level in the event of an IFTS component failure. Establishing administrative controls for Fuel Handling Building closure when the IFTS blind flange is removed ensures that the Fuel Handling Area exhaust ventilation subsystem is in operation.

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SURVEILLANCE REQUIREMENT

SR 3.6.1.3.4 (continued)

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

SR 3.6.1.3.5

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

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BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.3.5 (continued)

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

SR 3.6.1.3.6

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

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

SR 3.6.1.3.7

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

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

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

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

SR 3.6.1.3.10

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

If work is performed on a valve in a Main Steam Line following the satisfactory performance of as-found testing, the post maintenance testing must ensure that Main Steam Line leakage does not exceed 100 scfh and total leakage does not exceed 250 scfh, prior to entering Modes 1, 2, or 3.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.10 (continued)

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

SR 3.6.1.3.11

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 (continued)

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

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

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

SR 3.6.1.3.12

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

The SR is modified by a Note stating that this SR is only required to be met in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when the purge valves are required to be capable of closing (e.g., during OPDRVs), pressurization concerns are not present, thus the purge valves can be fully open. The 24 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

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

REFERENCES

1. USAR, Chapter 15.
2. USAR, Section 6.2.
3. Plant Data Book, Tab G.
4. 10 CFR 50, Appendix J, Option B.
5. USAR, Section 15.7.6

BASES

SURVEILLANCE REQUIREMENT (continued)

SR 3.6.1.6.2

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

The 24 month Frequency is based on the need to perform this Surveillance during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

REFERENCES

1. GESSAR-II, Appendix 3B, Attachment A, Section 3BA.8.
2. USAR, Section 7.6.1.11.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Feedwater Leakage Control System (FWLCS)

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

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

The FWLCS satisfies Criterion 3 of the NRC Policy Statement.

LCO

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

(continued)

BASES

BACKGROUND (continued)	This Specification ensures that the performance of the primary containment, in the event of a fuel handling accident involving handling of recently irradiated fuel, or reactor vessel draindown, provides an acceptable leakage barrier to contain fission products, thereby minimizing offsite doses.
APPLICABLE SAFETY ANALYSES	The safety design basis for the primary containment is that it contain fission products to limit doses at the site boundary to within limits. The primary containment OPERABILITY in conjunction with the automatic closure of selected OPERABLE containment isolation valves (LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," and LCO 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"), assures a leak tight fission product barrier. The fuel handling accident calculations do not credit the primary or secondary containment; all gaseous fission products released from the water pool over the damaged fuel bundles are assumed to be immediately discharged directly to the environment (Ref. 2). During MODES 4 and 5, there are no accident analyses that credit the primary containment. However, it was determined that Specifications should remain in place per Criterion 4 to address operations with the potential for draining the reactor vessel (OPDRVs) and fuel handling accidents. Criterion 3 of the NRC Policy Statement would apply if dose calculations are revised to credit the primary containment during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).
LCO	Primary containment OPERABILITY is maintained by providing a contained volume to limit fission product escape following a fuel handling accident involving handling of recently irradiated fuel, or an unanticipated water level excursion. Compliance with this LCO will ensure a primary containment configuration, including the equipment

(continued)

BASES

LCO (continued)	hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Since offsite dose analyses conservatively assume LOCA leakage pathways and rates, the isolation and closure times of automatic containment isolation valves supports an OPERABLE primary containment during shutdown conditions. Furthermore, normal operation of the inclined fuel transfer system (IFTS) without the IFTS blind flange installed is considered acceptable for meeting Primary Containment-Shutdown OPERABILITY. Leakage rates specified for the primary containment and air locks, addressed in LCO 3.6.1.1 and LCO 3.6.1.2 are not directly applicable during the shutdown conditions addressed in this LCO.
APPLICABILITY	In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining an OPERABLE primary containment in MODE 4 or 5 to ensure a control volume, is only required during situations for which significant releases of radioactive material can be postulated; such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of Primary Containment when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 2). OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the primary containment is not required to be OPERABLE.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

The containment vacuum breakers satisfy Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the containment. However, it was determined that Specifications should remain in place per Criterion 4 to address operations with the potential for draining the reactor vessel (OPDRVs) and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the containment during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

LCO

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

APPLICABILITY

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

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

(continued)

BASES

APPLICABILITY (continued)

When handling recently irradiated fuel in the primary containment, and during operations with a potential for draining the reactor vessel (OPDRVs) the primary containment is required to be OPERABLE. Containment vacuum breakers are therefore required to be OPERABLE during these evolutions to protect the primary containment against an inadvertent initiation of the Containment Spray System. Due to radioactive decay, handling of fuel only requires OPERABILITY of Containment Vacuum Breakers when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 2). Since OPDRVs assume that one or more fuel assemblies are loaded into the core, this LCO would not be applicable for OPDRVs if no fuel is in the reactor vessel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.11.3 (continued)

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

Performance of this SR includes a CHANNEL CALIBRATION of the isolation valve actuation instrumentation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. USAR, Section 6.2.1.1.4.2.
2. USAR, Section 15.7.6.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.12 Containment Humidity Control

BASES

BACKGROUND

Primary containment temperature and humidity are initial condition inputs into the analysis that evaluates the initiation of RHR containment spray during normal plant operation. A curve was determined of initial primary containment average temperature and humidity which would maintain peak vacuum inside containment ≤ 0.72 psi (design is ≤ 0.80 psi) during the spray initiation event. This curve then determines the containment average temperature-to-humidity combinations that are acceptable whenever the conditions exist for the inadvertent containment spray initiation event (whenever the primary containment leak tight barrier has been established).

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for the inadvertent initiation of the RHR Containment Spray System. The initial containment average temperature and relative humidity have an effect on the results of this analyses. As long as the average temperature and relative humidity is maintained within the limits of Figure B 3.6.1.12-1, the design can adequately perform in the inadvertent containment spray event.

There is no need to monitor the containment average temperature-to-relative humidity when the primary containment is not OPERABLE (i.e., has large enough openings such that a vacuum would not be created during an RHR containment spray event).

The containment relative humidity satisfies Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the containment. However, it was determined that Specifications should remain in place per Criterion 4 to address OPDRVs and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the containment during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

(continued)

BASES (continued)

LCO

In the event RHR containment spray initiates during normal plant conditions, and while the primary containment is required to be OPERABLE, the initial average temperature and relative humidity must be within defined limits in order to assure proper response of the primary containment. When the primary containment is not OPERABLE, and contains sufficient openings such that a vacuum would not be created during a containment spray initiation, the average temperature and relative humidity are not required to be maintained within the prescribed limits.

(continued)

BASES (continued)

APPLICABILITY

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

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

When handling recently irradiated fuel in the primary containment, and during operations with a potential for draining the reactor vessel (OPDRVs) the primary containment is required to be OPERABLE. Therefore, the proper relationship between containment average temperature and relative humidity must exist during these evolutions. Due to radioactive decay, handling of fuel only requires control over Containment humidity when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 2).

ACTIONS

A.1

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.12.1

Verifying that the primary containment average temperature and relative humidity are within limits ensures that operation remains within limits assumed in the primary containment analyses for initiation of RHR containment spray (Ref. 1).

The 24 hour Frequency of this SR is considered acceptable based on the observed slow rates of temperature and relative humidity changes within the primary containment due to the large volume of the primary containment.

REFERENCES

1. USAR, Section 6.2.1.1.4.2.
2. USAR, Section 15.7.6.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. Average temperature is determined by taking an average of the functional suppression pool water temperature channels. The 24 hour Frequency has been shown to be acceptable based on operating experience. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which testing will heat up the suppression pool, has been shown to be acceptable based on operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

1. USAR, Section 6.2.
2. USAR, Section 15.2.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

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

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

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

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

(continued)

BASES

BACKGROUND (continued)

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

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

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

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

f. The Annulus Exhaust Gas Treatment System is OPERABLE.

APPLICABLE SAFETY ANALYSES

There is one accident for which credit is taken for secondary containment OPERABILITY. This is a LOCA (Ref. 1). The secondary containment performs no active function in response to this limiting event; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the AEGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit secondary containment. However, it was determined that Specifications should remain in place per Criterion 4 to address OPDRVs and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit secondary containment during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

(continued)

LCO

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

(continued)

BASES (continued)

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPRDVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of Secondary Containment when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 2). OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the secondary containment is not required to be OPERABLE.

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

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

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

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

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accident for which the secondary containment boundary is required is a loss of coolant accident (Ref. 1). The secondary containment performs no active function in response to this limiting event, but the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Annulus Exhaust Gas Treatment (AEGT) System before being released to the environment.

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

SCIVs satisfy Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the secondary containment. However, it was determined that Specifications should remain in place per Criterion 4 to address OPDRVs and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the secondary containment during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

APPLICABILITY

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

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of secondary containment isolation valves when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 3). OPDRVs assume that one or more fuel assemblies are loaded into the core. Therefore, if the fuel is fully off-loaded from the reactor vessel, the SCIVs are not required to be OPERABLE.

ACTIONS

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

(continued)

BASES

ACTIONS (continued)

penetration can be rapidly isolated when the need for secondary containment isolation is indicated.

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

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

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

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

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

(continued)

BASES

BACKGROUND (continued)

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

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

APPLICABLE SAFETY ANALYSES

The design basis for the AEGT System is to mitigate the consequences of a loss of coolant accident. For all events analyzed, the AEGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The AEGT System satisfies Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the AEGT System. However, it was determined that Specifications should remain in place per Criterion 4 to address OPDRVs and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the AEGT System during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

LCO

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

(continued)

BASES (continued)

APPLICABILITY

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

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

(continued)

BASES

APPLICABILITY (continued)

other situations under which significant releases of radioactive material can be postulated, such as during movement of recently irradiated fuel assemblies in the primary containment, or during operations with a potential for draining the reactor vessel (OPDRVs). Due to radioactive decay, handling of fuel only requires OPERABILITY of the AEGT System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 5).

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

ACTIONS

A.1

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

B.1 and B.2

If the AEGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the

(continued)

ACTIONS

B.1 and B.2 (continued)

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

C.1, C.2.1 and C.2.2

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1 (continued)

filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required AEGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The AEGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing HEPA filter efficiency, charcoal adsorber efficiency and bypass leakage, system flow rate, and general operating parameters of the filtration system. (Note: Values identified in the VFTP are Surveillance Requirement values.) Specified test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

This SR verifies that each AEGT subsystem starts and isolation dampers open upon receipt of a manual initiation signal from the control room and an actual or simulated initiation and operates throughout its emergency operating sequence for the LOCA signal.

The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.5.1.6 overlaps this SR to provide complete testing of the safety function. This Surveillance can be performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. USAR, Section 6.5.3.
3. USAR, Section 15.6.5.
4. Regulatory Guide 1.52, Rev. 2.
5. USAR, Section 15.7.6.

BASES

BACKGROUND (continued)

c. The drywell equipment hatch is closed and sealed;

d. The drywell head is installed and sealed;

e. The Drywell Vacuum Relief System is OPERABLE except as provided in LCO 3.6.5.6, "Drywell Vacuum Relief System";

f. The drywell leakage rates are within the limits of SR 3.6.5.1.1;

g. The suppression pool is OPERABLE; and

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

This Specification is intended to ensure that the performance of the drywell in the event of a DBA meets the assumptions used in the safety analyses (Ref. 1).

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the drywell are presented in Reference 1. The safety analyses assume that for a high energy line break inside the drywell, the steam is directed to the suppression pool through the horizontal vents where it is condensed. Maintaining the pressure suppression capability assures that safety analyses remain valid and that the peak LOCA temperature and pressure in the primary containment are within design limits.

The drywell satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

Maintaining the drywell OPERABLE is required to ensure that the pressure suppression design functions assumed in the safety analyses are met. The drywell is OPERABLE if the drywell structural integrity is intact and the bypass leakage is within limits, except prior to the first startup after performing a required drywell bypass leakage test. At this time, the drywell bypass leakage must be ≤ 10% of the drywell bypass leakage limit.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ability of the CRER System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the USAR, Chapters 6 and 15 (Refs. 3 and 4, respectively). The emergency recirculation mode of the CRER System is assumed to operate following a loss of coolant accident, main steam line break, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active or passive failure will cause the loss of ability to recirculate air in the control room.

The CRER System satisfies Criterion 3 of the NRC Policy Statement in MODES 1, 2, or 3. During MODES 4 and 5, there are no accident analyses that credit the CRER System. However, it was determined that Specifications should remain in place per Criterion 4 to address OPDRVs and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the CRER System during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

LCO

Two independent and redundant subsystems of the CRER System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in a failure to meet the dose requirements of GDC 19 in the event of a DBA (for the design-basis Alternative Source Term (AST) LOCA and fuel handling accident analyses, the licensing basis Control Room dose limit is 5 Rem TEDE (Ref. 6 and 7)).

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

a. Fans are OPERABLE;

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

(continued)

BASES

LCO (continued)	c. Heater, demister, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained. In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.
APPLICABILITY	In MODES 1, 2, and 3, the CRER System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

(continued)

BASES

APPLICABILITY (continued)

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

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

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

Due to radioactive decay, handling of fuel only requires OPERABILITY of the Control Room Emergency Recirculation System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 4).

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

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 6.5.1.
2. USAR, Section 6.4.1.
3. USAR, Chapter 6.
4. USAR, Chapter 15.
5. Regulatory Guide 1.52, Revision 2, March 1978.
6. Amendment No. 103 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1; and Letter, D. Pickett (NRC) to L. Myers (FENOC), "Issuance of Exemption from 10 CFR Part 50, Appendix A, General Design Criterion 19", dated March 26, 1999.
7. Amendment No. 122 to Facility Operating License No. NPF-58, Perry Nuclear Power Plant, Unit 1.

B 3.7 PLANT SYSTEMS

B 3.7.4 Control Room Heating, Ventilating, and Air Conditioning (HVAC) System

BASES

BACKGROUND

The Control Room HVAC System provides temperature control for the control room.

The Control Room HVAC System consists of two independent, redundant subsystems that provide cooling and heating of recirculated control room air. Each subsystem consists of heating coils, cooling coils, fans, chillers with compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.

The Control Room HVAC System is designed to provide a controlled environment under both normal and accident conditions. The Control Room HVAC System operation in maintaining the control room temperature is discussed in the USAR, Sections 6.4 and 9.4.1 (Refs. 1 and 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the Control Room HVAC System is to maintain the control room temperature for a 30 day continuous occupancy.

The Control Room HVAC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room HVAC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single active failure of a component of the Control Room HVAC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control. The Control Room HVAC System is designed in accordance with Seismic Category I requirements. The Control Room HVAC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The Control Room HVAC System satisfies Criterion 3 of the NRC Policy Statement in MODES 1, 2, and 3. During MODES 4 and 5, there are no accident analyses that credit the Control Room HVAC System. However, it was determined that Specifications should remain in place per Criterion 4 to address OPDRVs and fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the Control Room HVAC during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

(continued)

BASES (continued)

LCO

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

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

APPLICABILITY

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

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

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

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

Due to radioactive decay, handling of fuel only requires OPERABILITY of the Control Room HVAC System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 3).

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

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

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

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

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

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analysis. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room HVAC System is not expected over this time period.

REFERENCES

1. USAR, Section 6.4.
2. USAR, Section 9.4.1.
3. USAR, Section 15.7.6.

B 3.7 PLANT SYSTEMS

B 3.7.7 Fuel Pool Water Level

BASES

BACKGROUND	The minimum water level in the spent fuel storage pools and upper containment fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident. A general description of the fuel handling building (FHB) spent fuel storage pools and upper containment fuel storage pool design is found in the USAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the USAR, Sections 15.7.4 and 15.7.6 (Refs. 2 and 3, respectively).
APPLICABLE SAFETY ANALYSES	The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the offsite radiological consequences (calculated Total Effective Dose Equivalent (TEDE) doses at the exclusion area and low population zone boundaries) are ≤ 25% of the 10 CFR 50.67 (Ref. 5) exposure guidelines. The Control Room is also evaluated to ensure doses are less than the 10 CFR 50.67 exposure guidelines. A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide 1.183 (Ref. 6). The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto stored fuel bundles. The consequences of a fuel handling accident inside the FHB and inside containment are documented in References 2 and 3, respectively. The water levels in the FHB spent fuel storage pools and upper containment fuel storage pools provide for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident. The fuel pool water level satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

REFERENCES

1. USAR, Section 9.1.2.
2. USAR, Section 15.7.4.
3. USAR, Section 15.7.6.
4. Deleted
5. 10 CFR 50.67.
6. Regulatory Guide 1.183, July 2000.

BASES

BACKGROUND (continued)	With the boundaries in place, the FHB Ventilation Exhaust System will assure that any releases occurring as a result of a FHA are filtered.
APPLICABLE SAFETY ANALYSES	There are no accidents for which credit is taken for FHB OPERABILITY. Although there are no accident analyses that credit the FHB, it was determined that Specifications should remain in place per Criterion 4 to address fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) in the FHB (Ref. 1). Criterion 3 of the NRC Policy Statement would apply if dose calculations are revised to credit the FHB during handling of recently irradiated fuel.
LCO	An OPERABLE FHB provides a control volume into which fission products can be diluted and processed prior to release to the environment. For the FHB to be considered OPERABLE, it must provide proper air flow patterns to ensure that there is no uncontrolled release of radioactive material during a FHA involving handling of recently irradiated fuel in the FHB.
APPLICABILITY	In plant operating MODES, OPERABILITY of the FHB is not required since leakage from the primary containment will not be released into the FHB. Regardless of the plant operating MODE, anytime recently irradiated fuel is being handled in the FHB there is the potential for significant radioactive releases due to a FHA, and the FHB is required to mitigate the consequences. Due to radioactive decay, handling of fuel only requires OPERABILITY of the Fuel Handling Building when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 1).

(continued)

BASES (continued)

ACTIONS

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

A.1

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

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 and SR 3.7.8.2

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

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

REFERENCES

1. USAR, Section 15.7.4 and 15.7.6.

BASES

BACKGROUND (continued)	radiation condition, an alarm will occur in the control room, and the operating supply fan from the FHB Ventilation Supply System will trip. The exhaust subsystems remain operational to continue exhausting contaminated air from the fuel handling area through the charcoal filter trains, thus precluding any uncontrolled release of radioactivity to the outside environment.
APPLICABLE SAFETY ANALYSES	The design basis for the FHB Ventilation Exhaust System is to mitigate the consequences of a FHA involving handling of irradiated fuel. However, there are no longer any dose calculations which credit the FHB Ventilation Exhaust System to reduce, via filtration and adsorption, the radioactive material released to the environment. Although there are no accident analyses that credit the FHB Ventilation Exhaust System, it was determined that Specifications should remain in place per Criterion 4 to address fuel handling accidents. Criterion 3 would apply if dose calculations are revised to credit the FHB Ventilation Exhaust System during handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.
LCO	Following a FHA involving handling of recently irradiated fuel, a minimum of two FHB ventilation exhaust subsystems are required to maintain the FHB at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for three OPERABLE subsystems ensures operation of at least two FHB ventilation exhaust subsystems in the event of a single active failure.
APPLICABILITY	In plant operating MODES, OPERABILITY of the FHB Ventilation Exhaust System is not required since leakage from the primary containment will not be released into the FHB. Regardless of the plant operating MODE, anytime recently irradiated fuel is being handled in the FHB there is the potential for significant radioactive releases due to a FHA, and the FHB Ventilation Exhaust System is required to mitigate the consequences. Due to radioactive decay, handling of fuel only requires OPERABILITY of the Fuel

(continued)

BASES

APPLICABILITY (continued)	Handling Building Ventilation Exhaust System when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours. Although this Function retains APPLICABILITY during "movement of recently irradiated fuel", which could be interpreted to permit fuel handling before 24 hours of radiological decay if certain buildings and filtration systems are OPERABLE, this is not the case. Fuel handling during that period is prohibited since no dose calculations exist to address a fuel handling accident within the first 24 hours after the reactor core is sub-critical (Ref. 3).
ACTIONS	The Required Actions have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.9.2

This SR verifies that the required FHB ventilation exhaust filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The FHB Ventilation Exhaust System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 4) whenever recently irradiated fuel is going to be handled. The VFTP includes testing HEPA filter efficiency, charcoal adsorber efficiency and bypass leakage, system flow rate, and general operating parameters of the filtration system. (Note: Values identified in the VFTP are Surveillance Requirement values.) Specified test frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.9.3

This SR requires verification that each FHB ventilation exhaust subsystem can be started from the control room, and that the FHB ventilation exhaust system performs satisfactorily during an actual or simulated actuation of the FHA instrumentation. This SR will include calibration of the FHB ventilation exhaust radiation monitor (noble gas). This Surveillance can be performed with the reactor at power. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

SR 3.7.9.4

This SR requires the performance of a CHANNEL FUNCTIONAL TEST on the FHB ventilation exhaust radiation monitor (noble gas) to ensure the entire channel will perform its intended function.

The Frequency is based on plant operating experience with regard to channel operability and the recommendations of Generic Letter 93-05.

(continued)

BASES (continued)

REFERENCES	1.	10 CFR 50, Appendix A, GDC 41.
	2.	USAR, Section 6.2.3.
	3.	USAR, Section 15.7.4 and 15.7.6.
	4.	Regulatory Guide 1.52, Rev. 2.

ACTIONS

B.3.1 and B.3.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either Required Actions B.3.1 or B.3.2, the corrective actions program will continue to evaluate the common cause failure possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable time to confirm that the OPERABLE DG(s) are not affected by the same problem as the inoperable DG.

B.4

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

Entering an Emergency Diesel Generator (EDG), Division 1 or 2, Completion Time greater than or equal to 72 hours is considered a risk-informed Allowed Outage Time (AOT). Specification 5.5.13.1 requires the provisions of a Configuration Risk Management Program (CRMP) be adhered to when the risk-informed AOT is entered. The CRMP is controlled by plant administrative procedures.

A risk-informed AOT duration of up to 14 days may be taken in accordance with the CRMP for one EDG, Division 1 or 2, within a period of 365 days (one year). The basis of the "once per year" frequency is to avoid scheduling back to back EDG AOTs. In accordance with the CRMP and the acceptance criteria of RG 1.177, a qualitative assessment may be performed to assess time periods between AOTs of less than one year.

The risk-informed AOT is meant to be a pre-planned activity with a work scope that limits the risk to the plant by controlling the impact on risk significant components and systems. In an effort to reduce the risk to the plant, when planning or performing a EDG risk-informed AOT, positive measures will be taken to preclude subsequent testing or maintenance on those risk significant systems, subsystems, trains, components, and devices that depend on the Division 1 or 2 EDG as a source of emergency power.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.6 (continued)

standby power sources. This Surveillance provides assurance that each fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The design of the fuel transfer systems is such that pumps operate automatically in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing. Therefore, a 31 day Frequency is specified to correspond to the maximum interval for DG testing.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Transfer of each 4.16 kV ESF bus power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit. The 24 month Frequency of the Surveillance is based on engineering judgment taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

This SR has been modified by a Note. The reason for the Note is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

1) Maintenance; and

2) Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of this Surveillance is allowed 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the 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 the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load while maintaining a specified margin to the overspeed trip. The referenced load for the Division 1 DG is the 1400 kW low pressure core spray pump; for the Division 2 DG, the 729 kW residual heat removal (RHR) pump; and for the Division 3 DG the 2400 kW HPCS pump. This surveillance may be accomplished by: 1) tripping the DG output breaker with the associated single largest load while paralleled to offsite power, or while solely supplying the bus, or 2) tripping its associated single largest load with the DG solely supplying the bus. As required by IEEE-308 (Ref. 13), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

This SR has been modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

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

2) Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of this Surveillance is allowed 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the 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 the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

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

2) Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of this Surveillance is allowed 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the 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 the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the Division 1 and 2 nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

The DG auto-start times are derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and energization of permanent and auto-connected loads through the load sequence (individual load timers) is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, systems are not

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

capable of being operated at full flow, or RHR subsystems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of the connection and energization of these loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 24 month Frequency takes into consideration unit conditions required to perform the surveillance, and is consistent with the intent of Regulatory Guide 1.108 (Ref. 9) paragraph C.2.a to perform this test at refueling intervals.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for Division 1 and 2 DGs. For the Division 3 DG, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

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

2) Post maintenance testing that requires performance of portions of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

testing, and other unanticipated OPERABILITY concerns). Performance of portions of this Surveillance is allowed 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 this assessment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds for Divisions 1 and 2 and 13 seconds for Division 3) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability.

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations for Division 1 and 2 DGs. For the Division 3 DG, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation. The reason for Note 2 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

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

2) Post maintenance testing that requires performance of portions of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.12 (continued)

testing, and other unanticipated OPERABILITY concerns). Performance of portions of this Surveillance is allowed 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 this assessment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.13

This Surveillance demonstrates that DG non-critical protective functions (e.g., high jacket water temperature) are bypassed on an ECCS initiation test signal and critical protective functions trip the DG to avert substantial damage to the DG unit. The non-critical trips are bypassed during DBAs and provide alarms on abnormal engine conditions. These alarms provide the operator with necessary information to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG.

The 24 month Frequency is based on engineering judgment, taking into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

The SR is modified by a Note. The reason for the Note is that performing the Surveillance removes a required DG from service. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

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

3) Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of this Surveillance is allowed 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.13 (continued)

or operated independently for the 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 the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours-22 hours of which is at a load equivalent to the continuous rating of the DG, and 2 hours

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.16 (continued)

2) Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of this Surveillance is allowed 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the 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 the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.17

Demonstration of the test mode override ensures that the DG availability under accident conditions is not compromised as the result of testing. Interlocks to the LOCA sensing circuits cause the DG to automatically reset to ready-to-load operation if an ECCS initiation signal is received during operation in the test mode. Ready-to-load operation is defined as the DG running at rated speed and voltage with the DG output breaker open. These provisions for automatic switchover are required by IEEE-308 (Ref. 13), paragraph 6.2.6(2).

The requirement to automatically energize the emergency loads with offsite power is essentially identical to that of SR 3.8.1.12. The intent in the requirement associated with SR 3.8.1.17.b is to show that the emergency loading is not affected by the DG operation in test mode. In lieu of actual demonstration of connection and energization of loads, testing that adequately shows the capability of the emergency loads to perform these functions is acceptable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The 24 month Frequency takes into consideration unit conditions required to perform the surveillance, and is consistent with the intent of Regulatory Guide 1.108 (Ref. 9) paragraph C.2.a to perform this test at refueling intervals.

This SR has been modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

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

2) Post maintenance testing that requires performance of portions of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of portions of this Surveillance is allowed 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 this assessment

SR 3.8.1.18

Under accident conditions, loads are sequentially connected to the bus by the time delay relays. The time delay relays control the permissive and starting signals to motor breakers to prevent overloading of the bus power supply due to high motor starting currents. The 10% load sequence time tolerance ensures that sufficient time exists for the bus power supply to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

The 24 month Frequency takes into consideration unit conditions required to perform the surveillance, and is consistent with the intent of Regulatory Guide 1.108 (Ref. 9) paragraph C.2.a to perform this test at refueling intervals.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance during these MODES would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.18 (continued)

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

2. Post maintenance testing that requires performance of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of this Surveillance is allowed 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 Surveillance, a successful Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the 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 the Surveillance is performed in MODE 1 or 2. Risk insights or deterministic methods may be used for this assessment.

SR 3.8.1.19

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

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.11, during a loss of offsite power actuation test signal in conjunction with an ECCS initiation signal. In lieu of actual demonstration of connection and energization of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.19 (continued)

the entire connection and loading sequence is verified. The verification for assuring that the auto-connected emergency loads are energized has a timing requirement associated with Division 3. Thus verification for Division 1 or 2 is simply a check that the auto-connected loads are energized, whereas the verification for Division 3 includes a check that the auto-connected loads are energized in ≤ 13 seconds.

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 24 months.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.19 (continued)

consistent with manufacturer recommendations for Division 1 and 2 DGs. For the Division 3 DG, standby conditions mean that the lube oil is heated by the jacket water and continuously circulated through a portion of the system as recommended by the vendor. Engine jacket water is heated by an immersion heater and circulates through the system by natural circulation. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge plant safety systems. Credit may be taken for unplanned events that satisfy this SR. Examples of unplanned events may include:

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

2) Post maintenance testing that requires performance of portions of this Surveillance in order to restore the component to OPERABLE, provided the maintenance was required, or performed in conjunction with maintenance required to reestablish OPERABILITY (e.g. post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns). Performance of portions of this Surveillance is allowed 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 this assessment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.20

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously.

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. 9). During operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation, and as a result, plant safety systems. Therefore, this Surveillance shall only be performed during shutdown.

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources-Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources-Operating."

APPLICABLE SAFETY ANALYSES

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

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

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

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

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

(continued)

BASES

LCO (continued)

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

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

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

(continued)

BASES

LCO (continued)

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

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

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

APPLICABILITY

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

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

b. Systems used to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the AC Sources when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours);

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

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

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

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

The 24 month Frequency of the Surveillance is based on engineering judgement, taking into consideration the desired unit conditions to perform the Surveillance. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provides an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR, provided visible corrosion is removed during performance of this Surveillance.

The 24 month Frequency of the Surveillance is based on engineering judgement, taking into consideration the desired unit conditions to perform the Surveillance. The 24 month Frequency is based on operating experience, and is consistent with a typical industry refueling cycle.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.8

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

The acceptance criteria for this Surveillance is consistent with IEEE-450 (Ref. 8) and IEEE-485 (Ref. 11). 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.

The Surveillance Frequency for this test is normally 60 months, or every 18 months if the battery shows degradation, or if the battery has reached 85% of its expected life. Degradation is indicated, according to IEEE-450 (Ref. 8), when the battery capacity drops more than 10% of rated capacity from its previous performance test, or is below 90% of the manufacturer's rating. These Frequencies are based on the recommendations in IEEE-450 (Ref. 8). The 60 month test is taken directly from the IEEE recommended surveillance interval. The 18 month interval has been the approved interval since initial licensing and meets the intent of the IEEE to perform the test at a reduced test interval when battery degradation or battery aging reaches the predetermined limits discussed above.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE Standard 308, 1978.
4. USAR, Section 8.3.2.
5. USAR, Chapter 6.
6. USAR, Chapter 15.
7. Regulatory Guide 1.93, December 1974.
8. IEEE Standard 450, 1995.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources-Shutdown

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

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

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

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

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

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

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

LCO

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

(continued)

BASES

APPLICABILITY (continued)

b. Required features used to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the DC Sources when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours);

(continued)

BASES

SURVEILLANCE REQUIREMENTS

Table 3.8.6-1 (continued)

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450 (Ref. 3). Footnote (c) to Table 3.8.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirmatory measurements may be made in less than 7 days.

REFERENCES

1. USAR, Chapter 6.
2. USAR, Chapter 15.
3. IEEE Standard 450, 1995.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems-Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution systems is provided in the Bases for LCO 3.8.7, "Distribution Systems-Operating."

APPLICABLE SAFETY ANALYSES

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

The OPERABILITY of the AC and DC electrical power distribution systems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

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

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

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

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident involving handling of recently irradiated fuel, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours.

The AC and DC electrical power distribution systems satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

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

Maintaining these portions of the AC and DC electrical power distribution systems energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling of recently irradiated fuel and inadvertent reactor vessel draindown).

APPLICABILITY

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

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

b. Required features used to mitigate a fuel handling accident involving handling of recently irradiated fuel are available (due to radioactive decay, handling of fuel only requires OPERABILITY of the Distribution Systems when the fuel being handled is recently irradiated, i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours);

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

(continued)

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level-Irradiated Fuel

BASES

BACKGROUND	The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft 9 inches above the top of the RPV flange. During refueling, this maintains a sufficient water level in the upper containment pool. Sufficient water is necessary to retain halogen (e.g., iodine) fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient halogen activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 50.67 limits, as provided by the guidance of Reference 1.
APPLICABLE SAFETY ANALYSES	During movement of irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for halogens. This relates to the assumption that 99.5% of the total halogens released from the pellet to cladding gap of all the dropped fuel assembly rods are retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the total fuel rod I-131 inventory and 5% of the other halogens (Ref. 1). Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft over irradiated assemblies in the RPV and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4). While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases. Based on this judgment, and the physical dimensions which preclude normal operation with water level 23 feet above the flange, a slight reduction in this water level is acceptable. RPV water level satisfies Criterion 2 of the NRC Policy Statement.
LCO	A minimum water level of 22 ft 9 inches above the top of the RPV flange is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference 1.
APPLICABILITY	LCO 3.9.6 is applicable during movement of irradiated fuel assemblies within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. Requirements for handling of new fuel assemblies or control rods (where water depth to the RPV flange is not of concern) are covered by LCO 3.9.7, "RPV Water - New Fuel or Control Rods." Requirements for fuel handling accidents in the spent fuel storage pools and upper fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level."
ACTIONS	<u>A.1</u> If the water level is < 22 ft 9 inches above the top of the RPV flange, all operations involving movement of irradiated fuel assemblies within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	<u>SR 3.9.6.1</u> Verification of a minimum water level of 22 ft 9 inches above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 (continued) during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2). The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.
REFERENCES	1. Regulatory Guide 1.183, July 2000. 2. USAR, Section 15.7.6. 3. Deleted 4. 10 CFR 50.67.

B 3.9 REFUELING OPERATIONS

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level-New Fuel or Control Rods

BASES

BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain halogen (e.g., iodine) fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient halogen activity would be retained to limit offsite doses from the accident to < 25% of 10 CFR 50.67 limits, as provided by the guidance of Reference 1.

APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for halogens. This relates to the assumption that 99.5% of the total halogens released from the pellet to cladding gap of all the dropped fuel assembly rods are retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 8% of the total fuel rod I-131 inventory and 5% of the other halogens (Ref. 1).

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft over irradiated assemblies in the RPV and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4).

The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	RPV water level satisfies Criterion 2 of the NRC Policy Statement.
LCO	A minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference 1.
APPLICABILITY	LCO 3.9.7 is applicable when moving new fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) over irradiated fuel assemblies seated within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, 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 storage pools and upper fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level." Requirements for handling irradiated fuel over the RPV are covered by LCO 3.9.6, "Reactor Pressure Vessel (RPV) Water Level-Irradiated Fuel."
ACTIONS	A.1 If the water level is < 23 ft above the top of irradiated fuel assemblies seated within the RPV, all operations involving movement of new fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.9.7.1 Verification of a minimum water level of 23 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

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

REFERENCES

1. Regulatory Guide 1.183, July 2000.
2. USAR, Section 15.7.6.
3. Deleted
4. 10 CFR 50.67.

BASES (continued)

ACTIONS

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

If one or more of the requirements of this Special Operations LCO are not met, the immediate implementation of these Required Actions commences activities which will restore operation consistent with the normal requirements for failure to meet LCO 3.3.1.1, LCO 3.3.8.2, LCO 3.9.1, LCO 3.9.2, LCO 3.9.4, and LCO 3.9.5 (i.e., all control rods fully inserted) or with the exceptions granted by this Special Operations LCO. The Completion Times for Required Action A.1, Required Action A.2.1, and Required Action A.2.2 are intended to require that these Required Actions be implemented in a very short time and carried through in an expeditious manner to either initiate action to restore the CRD and insert its control rod, or initiate action to restore compliance with this Special Operations LCO. Actions must continue until either Required Action A.2.1 or Required Action A.2.2 is satisfied.

SURVEILLANCE REQUIREMENTS

SR 3.10.5.1, SR 3.10.5.2, SR 3.10.5.3, SR 3.10.5.4, and SR 3.10.5.5

SR 3.10.5.1 verifies that all the control rods, other than the control rod withdrawn for the removal of the associated CRD, are fully inserted. This is required to ensure the SDM is within limits. SR 3.10.5.2 verifies that the local five by five array of control rods, other than the control rod withdrawn for the removal of the associated CRD, is disarmed, while the scram function for the withdrawn rod is not available. This is required to preclude the possibility of criticality. SR 3.10.5.3 verifies that a control rod withdrawal block has been inserted. This ensures that no other control rods can be inadvertently withdrawn under conditions when position indication instrumentation is inoperable for the withdrawn control rod. The Surveillance for LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," which is made applicable by this Special Operations LCO, is required in order to establish that this Special Operations LCO is being met. Also, SR 3.10.5.5 verifies that no other CORE ALTERATIONS are being made. This is required to ensure the assumptions of the safety analysis are satisfied.

Periodic verification of the administrative controls established by this Special Operations LCO is prudent to preclude the possibility of an inadvertent criticality. The

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