Progress Energy

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: ITS 5.6.2.17

April 15, 2004
3F0404-03

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Crystal River Unit 3 – Technical Specifications Bases Control Program

Dear Sir:

Florida Power Corporation, doing business as Progress Energy Florida, Inc., hereby submits the changes that were made to the Crystal River Unit 3 (CR-3) Improved Technical Specifications (ITS) Bases as required by ITS 5.6.2.17. The attachments provide revisions to the CR-3 ITS Bases that will update NRC copies of the ITS.

Attachment A provides the instructions for updating the CR-3 ITS Bases. Attachment B provides the CR-3 ITS and Bases Lists of Effective Pages. Attachment C provides the replacement pages for the CR-3 ITS Bases.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing & Regulatory Programs at (352) 563-4883.

Sincerely,

SC Powell

S. C. Powell
Supervisor
Licensing & Regulatory Programs

SCP/ff

Attachments:
A. Instructions for Updating the Crystal River Unit 3 ITS and Bases
B. CR-3 ITS and Bases Lists of Effective Pages
C. Replacement CR-3 ITS Bases Pages

xc: Regional Administrator, Region II (w/o Attachment C)
Senior Resident Inspector (w/o Attachment C)
NRR Project Manager (w/o Attachment C)

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Power Line Street
Crystal River, FL 34428

A001

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT A

INSTRUCTIONS FOR UPDATING

THE CRYSTAL RIVER UNIT 3 ITS AND BASES

INSTRUCTIONS FOR UPDATING
THE CRYSTAL RIVER UNIT 3
IMPROVED TECHNICAL SPECIFICATIONS

4/15/04

Page(s) to be Removed		Page(s) to be Added	
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PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT B

CR-3 ITS AND BASES LISTS OF EFFECTIVE PAGES

IMPROVED TECHNICAL SPECIFICATIONS

List of Effective Pages
(Through Amendment 211)

Amendment Nos. 159, 164, 166, 171, 173, 181, 189 and 190 amended the CR-3 Operating License, only, and did not effect changes to the ITS LCOs or Bases.

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PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT C

REPLACEMENT CR-3 ITS BASES PAGES

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial condition assumptions used in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the worth of a potential ejected rod, the assumptions of available SDM, and the initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in FSAR Section 1.4, Criterions 6, "Reactor Core Design" and 29, "Reactivity Shutdown Capability", (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Limits on regulating rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are not violated.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). To achieve this approximately linear relationship, the regulating rod groups are withdrawn and operated in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are normally controlled automatically by the Integrated Control System (ICS) but can also be controlled manually. They are capable of adding reactivity quickly compared with borating or diluting the Reactor Coolant System (RCS).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together,

(continued)

BASES

BACKGROUND (continued)

LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and maintain the minimum required SDM in MODES 1 and 2.

This LCO is required to minimize fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accidents requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) or anticipated operational occurrences (Condition II). The LCOs governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Refs. 7 and 8); and

c. During an ejected rod accident, the fuel enthalpy must not exceed 280 cal/gm (Ref. 3).

(continued)

BASES

ACTIONS (continued)

D.1

If the regulating rods cannot be restored to within the acceptable operating limits for the original THERMAL POWER, or if the power reduction cannot be completed within the associated Completion Time, then the plant must be is placed in a MODE in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours or 4 hours is required depending on whether or not the CONTROL ROD drive sequence alarm is OPERABLE. Verification that the sequence and overlap are within limits at a 12 hour Frequency is sufficient to ensure these limits are preserved. The 4 hour Completion time is acceptable because little rod motion occurs in 4 hours due to fuel burnup, and the probability of a deviation occurring simultaneously with an inoperable sequence monitor in this relatively short time frame is low. Both Frequencies take into account the level of information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

With an OPERABLE regulating rod insertion limit alarm, verification of the regulating rod insertion limits as specified in the COLR at a Frequency of 12 hours is sufficient to detect regulating rod banks that may be approaching the group insertion limits, because little rod

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.2 (continued)

motion due to fuel burnup occurs in 12 hours. If the insertion limit alarm becomes inoperable, verification of the regulating rod group position at a Frequency of 4 hours is sufficient to detect whether the regulating rod groups may be approaching or exceeding their group insertion limits. Both Frequencies take into account the level of information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS or estimated critical boron concentration is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and make the reactor critical.

REFERENCES

1. FSAR, Section 1.4.
2. 10 CFR 50.46.
3. FSAR, Section 14.2.2.4.
4. FSAR, Section 3.1.2.2.
5. FSAR, Section 14.
6. CR-3 COLR.
7. BAW-10143P-A, Rev. 1, "BWC Correlation of Critical Heat Flux", April 1985.
8. BAW-10241P-00, BHTP DNB Correlation Applied with LYNXT.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

The insertion limits of the APSRs are initial condition assumptions in all safety analyses that are affected by core power distributions. The applicable criterion for these power distribution design requirements are FSAR Section 1.4, Criterion 6, "Reactor Core Design", (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Limits on APSR insertion have been established, and all APSR positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is maintained.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in Reference 2. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. The APSRs are not required for reactivity insertion rate on trip or SDM and, therefore, they do not trip upon a reactor trip.

This LCO is required to minimize fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) or anticipated operational occurrences (Condition II). Acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Refs. 4 and 5); and

c. During an ejected rod accident, the fuel enthalpy must not exceed 280 cal/gm (Ref. 3).

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result should an accident occur simultaneously with violation of one or more of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

Operation at the APSR insertion limits may approach the maximum allowable linear heat generation rate or peaking factor with the allowed QPT present.

The APSR insertion limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on APSR physical insertion as defined in the COLR must be maintained because they serve the function of controlling the power distribution within an acceptable range.

Measurement system-independent limits for APSR insertion are provided in the COLR. Measurement system-dependent maximum allowable setpoints are derived by adjustment of the measurement system-independent limits to allow for THERMAL POWER level uncertainty and rod position errors.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions.

Because the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the positions specified in the COLR.

B.1

If the APSRs cannot be restored to their intended positions within the associated Completion Time, then the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

Verification that the APSRs are within their insertion limits at a 12 hour Frequency is sufficient to ensure that the APSR insertion limits are preserved. The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The probability of a deviation occurring simultaneously with a non-functioning APSR position computer alarm is low in this relatively short time frame. Also, the Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

REFERENCES

1. FSAR, Section 1.4.

2. 10 CFR 50.46.

(continued)

BASES

REFERENCES (continued)

3. FSAR, Section 14.2.2.4.
4. BAW-10143P-A, Rev. 1, "BWC Correlation of Critical Heat Flux", April 1985.
5. BAW-10241P-00, BHTP DNB Correlation Applied with LYNXT.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits given in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions in the safety analyses related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum linear heat rate (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a zirconium water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value

(continued)

BASES

BACKGROUND (continued)

during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on AXIAL POWER IMBALANCE are determined directly by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution.

Core protection from the effects of excessive AXIAL POWER IMBALANCE is accomplished in a tiered approach. The operating envelope limits addressed by this Specification form the first line of defense, are monitored by the operator and compensatory actions initiated when AXIAL POWER IMBALANCE is not within the acceptable region. The AXIAL POWER IMBALANCE operating limit envelope contained in the COLR represents the setpoint for which the core power distribution would either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss of flow accident with the maximum allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

The next line of defense is the Reactor Protection System (RPS) Flux/Delta Flux/Flow trip setpoints. The trip setpoints are addressed in LCO 3.3.1 and are developed by conservatively adjusting the AXIAL POWER IMBALANCE operating limits to account for measurement system inaccuracies and other potential errors. The trip setpoints ensure the reactor will be automatically shutdown prior to exceeding a Safety Limit.

The last line of defense is the Safety Limit itself (Reference Section 2.0). If the AXIAL POWER IMBALANCE protective limit specified in the COLR is exceeded, then the operator must take additional actions to preclude conditions under which DNB could occur. Even with AXIAL POWER IMBALANCE at the Safety Limit, damage to the fuel will not occur.

(continued)

BASES

BACKGROUND (continued)

Actual alarm setpoints are more restrictive than the maximum allowable setpoint values to provide additional conservatism between the actual alarm setpoints and the measurement system independent (operating) limit.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The LCOs based on power distribution, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," preclude core power distributions that would violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Refs. 2 and 3).

The regulating rod positions, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs governing the four process variables cited above. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The regulating rod insertion, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are monitored and controlled during power operation to ensure that the power distribution is within the bounds set by the safety analyses. The axial

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits; and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distributions.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account when the reload safety evaluation analysis is performed.

Operation at the AXIAL POWER IMBALANCE limit must be interpreted as operating the core at the maximum allowable $F_Q(Z)$ or $F^N_{\Delta H}$ peaking factors assumed as initial conditions for the accident analyses with the allowed QPT present.

AXIAL POWER IMBALANCE satisfies Criterion 2 of the NRC Policy Statement.

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The AXIAL POWER IMBALANCE operating limit envelope contained in the COLR represents the setpoint for which the core power distribution would either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss of flow accident with the allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is $\geq$ 40% RTP to prevent the core power distribution from exceeding the LOCA and loss of flow assumptions used in the accident analyses. Applicability of these limits at < 40% RTP in MODE 1 is not required because the combination of AXIAL POWER IMBALANCE with the maximum ALLOWABLE THERMAL POWER level will not

(continued)

BASES

ACTIONS

B.1 (continued)

Completion Time of 2 hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and on operating experience regarding the amount of time required to reach 40% RTP from RTP without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints in the operating procedure are derived from their corresponding measurement system independent limits (in the COLR) by adjusting for both the system observability errors and instrumentation errors. Although they are based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system independent limits to alarm setpoints assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Minimum Incore Detector System consists of detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Figure B 3.2.3-1 (Minimum Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

SR 3.2.3.1

If the plant computer becomes inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the AXIAL POWER IMBALANCE. Although these systems do not provide a direct calculation and display of the AXIAL POWER IMBALANCE, a 1 hour Frequency provides reasonable time between calculations for detecting any trends in the AXIAL POWER IMBALANCE that may exceed its alarm setpoint and for undertaking corrective action.

When the AXIAL POWER IMBALANCE alarm is OPERABLE, the operator receives an alarm if the AXIAL POWER IMBALANCE increases to its alarm setpoint. Verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE limits are not violated. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, would likely be discovered by the operator before the specified limits are violated.

REFERENCES

1. 10 CFR 50.46.
2. BAW-10143P-A, Rev. 1, "BWC Correlation of Critical Heat Flux", April 1985.
3. BAW-10241P-00, BHTP DNB Correlation Applied with LYNXT.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits given in the COLR.

Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis. Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow, or other accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum linear heat rate (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a zirconium water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

(continued)

BASES

BACKGROUND (continued)

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on QPT are determined directly by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate core power distribution assumptions used in the accident analysis. The error adjusted maximum allowable alarm setpoints (measurement system dependent limits) for QPT are specified in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition I) and anticipated operational occurrences (Condition II). The LCOs based on power distribution (LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4) preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Refs. 3 and 4).

QPT is one of the process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result if an accident occurs with

(continued)

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.2.4.1</u> (continued) After a THERMAL POWER increase following restoration of the QPT to within the steady state limit, QPT must be determined to remain within the steady state limit at the increased THERMAL POWER level. This is accomplished by monitoring QPT for 12 consecutive hourly intervals or until verified acceptable at $\geq$ 95% RTP to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the steady state limit again. In case QPT exceeds the steady state limit for more than 24 hours or exceeds the transient limit (Condition A, B, or D), the potential for this xenon redistribution is greater.
REFERENCES	1. 10 CFR 50.46. 2. BAW 10122A, Rev. 1, "Normal Operating Controls", May 1984. 3. BAW-10143P-A, Rev. 1, "BWC Correlation of Critical Heat Flux", April 1985. 4. BAW-10241P-00, BHTP DNB Correlation Applied with LYNXT.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking Factors

BASES

BACKGROUND

The purpose of this LCO is to establish limits that constrain the core power distribution within design limits during normal operation (Condition I) and during anticipated operational occurrences (Condition II) such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation at THERMAL POWER within specified acceptable fuel design limits.

$F_Q(Z)$ is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling Systems (ECCS) analysis. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. Because $F_Q(Z)$ is a ratio of local power densities, it is related to the maximum local (pellet) power density in a fuel rod. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaking that would exceed the loss of coolant accident (LOCA) linear heat rate (LHR) limits derived from the analysis of the ECCS.

The $F^N_{\Delta H}$ limit is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient. $F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum departure from nucleate boiling ratio (DNBR) occurs to the average fuel rod power. Because $F^N_{\Delta H}$ is a ratio of integrated powers, it is related to the maximum total power produced in a fuel rod. Operation within the $F^N_{\Delta H}$ limits given in the COLR prevents departure from nucleate boiling (DNB) during a postulated loss of forced reactor coolant flow accident.

Measurement of the core power peaking factors using the Incore Detector System to obtain a three dimensional power distribution map provides direct confirmation that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, and may be used to verify that the power peaking factors remain bounded when one or more normal operating parameters exceed their limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The limits on $F_Q(Z)$ are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 1). Higher cladding temperatures could cause severe cladding failure by oxidation due to a zirconium water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

The limits on $F^N_{\Delta H}$ provide protection from DNB during a limiting loss of flow transient. Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition (Refs. 2 and 3).

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1. Nuclear

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1 (continued)

for exceeding both the power peaking factors assumed in the accident analyses due to operation with unanalyzed core power distributions and spatial xenon distributions beyond their analyzed ranges.

The measured value of F_Q is increased by 2.0% to account for manufacturing tolerances on the fuel and further increased by 7.5% to account for measurement uncertainty. $F^N_{\Delta H}$ is increased by 5% for measurement uncertainty.

REFERENCES

1. 10 CFR 50.46.

2. BAW-10143P-A, Rev. 1, "BWC Correlation of Critical Heat Flux", April 1985.

3. BAW-10241P-00, BHTP DNB Correlation Applied with LYNXT.

BASES

ACTIONS

A.1 (continued)

With one channel inoperable, the system cannot meet the single-failure criterion and still satisfy the dual functional criteria described above. Therefore, when one vector valve logic channel is inoperable, the channel must be restored to OPERABLE status within 72 hours. This Condition is analogous to having one EFW train inoperable; wherein a 72 hour Completion Time is provided by the Required Actions of LCO 3.7.5, "EFW System." As such, the Completion Time of 72 hours is based on engineering judgment.

B.1 and B.2

If Required Action A.1 cannot be met within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test demonstrates that the EFIC-EFW-vector valve logic is capable of performing its intended function. The Frequency is based on operating experience that demonstrates failure of more than one channel within the same 31 day interval is unlikely.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.15 Reactor Building (RB) Purge Isolation-High Radiation

BASES

BACKGROUND

The RB Purge Isolation-High Radiation Function closes the RB purge and RB mini-purge valves to isolate the RB atmosphere from the environment and minimize releases of radioactivity in the event an accident occurs.

The radiation monitoring system (RM-A1) measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The sensitive volume of the gas sampler is shielded with lead and monitored by a Geiger-Mueller detector. The air sample is taken from the center of the purge exhaust duct through a nozzle installed in the duct.

The monitor will alarm and initiate closure of the valves prior to exceeding the noble gas limits specified in the Offsite Dose Calculation Manual.

The closure of the purge and mini-purge valves ensures the RB remains as a barrier to fission product release.

APPLICABLE SAFETY ANALYSES

FSAR Chapter 14 LOCA analysis assumes RB purge and mini purge lines are isolated within 60 seconds following initiation of the event. Since the early 1980's, this isolation time has only been practically applicable to the mini-purge valves since the large purge valves are required to be sealed closed during the MODES of plant operation (1, 2, 3, and 4) in which LOCAs are postulated to occur. Even

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

for mini-purge valves, design requirements on these valves require closure times on the order of 5 seconds. Thus, the purge isolation time of the current plant design is conservative to the original safety analysis.

The signal credited for initiating purge isolation in the original safety analysis is the RB Pressure - High ESAS signal and not RB Purge Isolation - High Radiation instrumentation. As such, design basis LOCA mitigation is not a basis for including this instrumentation.

RB purge isolation on high radiation is only required to maintain 10 CFR 20 limits during normal operations. However, this is not a basis for requiring a Technical Specification. Therefore, this Specification is not required in MODES 1, 2, 3 and 4.

Closure of the purge valves on high radiation is also not credited as part of the fuel handling accident (FHA) inside containment. The activity from the ruptured fuel assembly is assumed to be instantaneously released to the atmosphere in the form of a "puff" type release. Therefore, this specification is not required if moving fuel that has not been recently irradiated (See B 3.9.3, APPLICABLE SAFETY ANALYSIS for the cycle-specific definition of recently irradiated fuel).

This specification is only required to minimize dose if moving fuel that has been recently irradiated.

LCO

One channel of RB Purge Isolation-High Radiation instrumentation is required to be OPERABLE to ensure safety analysis assumptions regarding RB isolation are bounded. Operability of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

(continued)

BASES

APPLICABILITY

The RB Purge Isolation-High Radiation instrumentation shall be OPERABLE whenever movement of recently irradiated fuel within the RB is taking place. These specified conditions are indicative of those under which the potential for a fuel handling accident, and thus radiation release, is the greatest. While in MODES 5 and 6, when handling of recently irradiated fuel in the RB is not in progress, the isolation system does not need to be OPERABLE because the potential for a significant radioactive release is minimal and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

ACTIONS

A.1

Condition A applies to failure of the high radiation purge isolation function during movement of recently irradiated fuel assemblies within containment.

The immediate Completion Time is consistent with the loss of RB isolation capability under conditions in which the fuel handling accidents involving handling recently irradiated fuel are possible and the high radiation function is required to provide automatic action to terminate the release.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR is the performance of the CHANNEL CHECK for the RB purge isolation-high radiation instrumentation once every 12 hours. The CHANNEL CHECK is a comparison of the parameter indicated on the radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of

(continued)

BASES

FUNCTION	CHANNEL A	CHANNEL B
15. Steam Generator Water Level (Operating Range)	OTSG A: SP-17-LI1 or SP-17-LIR OTSG B: SP-21-LI1 or SP-21-LIR	OTSG A: SP-18-LI1 OTSG B: SP-22-LI1
16. Steam Generator Pressure	OTSG A: MS-106-PI1 or MS-106-PIR, OTSG B: MS-110-PI1 or MS-110-PIR	OTSG A: MS-107-PI1 or MS-107-PIR OTSG B: MS-111-PI1 or MS-111-PIR
17. Emergency Feedwater Tank Level	EF-98-LI1	EF-99-LI1
18a. Core Exit Temperature (Thermocouple) Quadrant WX XY YZ ZW	 IM-5G-TE/IM-6C-TE IM-9E-TE/IM-13G-TE IM-9H-TE/IM-100-TE IM-3L-TE/IM-60-TE	 IM-7F-TE/IM-2G-TE IM-10C-TE/IM-11G-TE IM-10M-TE/IM-13L-TE IM-4N-TE/IM-6L-TE
18b. Core Exit Temperature (Recorder)	RC-171-TR	RC-172-TR
19. Emergency Feedwater Flow	OTSG A: EF-25-FI1 OTSG B: EF-23-FI1	OTSG A: EF-26-FI1 OTSG B: EF-24-FI1
20. Low Pressure Injection Flow	DH-1-FI1	DH-1-FI2
21. Degrees of Subcooling	As Displayed on EMCO-38 *<u>Note:</u> Entry into LCO 3.3.17 is required if any of the following Hardware or RECALL Points are Out of Service.* <u>Hardware</u> Multiplexers EMCO-17/18/19 Comm. HUBs EMCO-07/20 Computers EMCO-21/40 Monitor EMCO-38 Transmitter EMCO-72 Receiver EMCO-73 Recorder RC-171-TR <u>RECALL Points</u> RCS Pressure LR RECL-243 RCS Pressure WR RECL-4 T-Hot RECL-17/239	As Displayed on EMCO-39 *<u>Note:</u> Entry into LCO 3.3.17 is required if any of the following Hardware or RECALL Points are Out of Service.* <u>Hardware</u> Multiplexers EMCO-26/27/28 Comm. HUBs EMCO-08/29 Computers EMCO-30/41 Monitor EMCO-39 Transmitter EMCO-74 Receiver EMCO-75 Recorder RC-172-TR <u>RECALL Points</u> RCS Pressure LR RECL-40 RCS Pressure WR RECL-5 T-Hot RECL-18/240
22. Emergency Diesel Generator kW Indication	EGDG-1A Wattmeter SSF-AH Main control board indicator	EGDG-1B Wattmeter SSF-AX Main control board indicator
23. LPI Pump Run Status	ESFA-LX3 (Red Light) or ESFA-HU (ES Light Matrix "A")	ESFB-LX3 (Red Light) or ESFB-HU (ES Light Matrix "B")
24. DHV-42 and DHV-43 Open Position	ESFA-KN3 (Red Light)	ESFB-KN3 (Red Light)
25. HPI Pump Run Status	HPI Pump 1A: ESFA-MF7 (Red Light) or ESFA-AH (ES Light Matrix "A") <u>OR</u> HPI Pump 1B: ESFA-MN7 (Red Light) or ESFA-AJ (ES Light Matrix "A")	HPI Pump 1C: ESFB-MF7 (Red Light) or ESFB-AH (ES Light Matrix "B") <u>OR</u> HPI Pump 1B: ESFB-MV7 (Red Light) or ESFB-AJ (ES Light Matrix "B")
26. RCS Pressure (Low Range)	RC-147-PI1	RC-148-PI1

NOTES: For Function 18a, each quadrant requires at least 2 OPERABLE detectors, one from each channel. OPERABILITY of only one detector for any quadrant constitutes entry into Condition A of LCO 3.3.17. Any quadrant with no OPERABLE detector constitutes entry into Condition C if LCO 3.3.17. Separate Condition entry is allowed for each quadrant.

For Function 25, OPERABILITY of indication is required only for the one ES selected HPI pump in each channel.

(continued)

BASES

LCO (continued)

The following list is a discussion of the specified instrument Functions listed in Table 3.3.17-1.

1. Wide Range Neutron Flux

Two wide-range neutron flux monitors are provided for post-accident reactivity monitoring over the entire range of expected conditions. Each monitor provides indication over the range of 10^{-8} to 100% log rated power covering the source, intermediate, and power ranges. Each monitor utilizes a fission chamber neutron detector to provide redundant main control board indication. A single channel provides recorded information in the control room. The control room indication of neutron flux is considered one of the primary indications used by the operator following an accident. Following an event the neutron flux is monitored for reactivity control. The operator ensures that the reactor trips as necessary and that emergency boration is initiated if required. Since the operator relies upon this indication in order to take specified manual action, the variable is included in this LCO. Therefore, the LCO deals specifically with this portion of the string.

2. Reactor Coolant System (RCS) Hot Leg Temperature

Two wide range resistance temperature detectors (RTD's), one per loop, provide indication of reactor coolant system hot leg temperature (T_H) over the range of 120° to 920°F. Each T_H measurement provides an input to a control room indicator. Channel B is also recorded in the control room. Since the operator relies on the control room indication following an accident, the LCO deals specifically with this portion of the string.

T_H is a Type A variable on which the operator bases manual actions required for event mitigation for which no automatic controls are provided.

Following a steam generator tube rupture, the affected steam generator is to be isolated only after T_H falls below the saturation temperature corresponding to the pressure setpoint of the main steam safety valves. For event monitoring, once the RCP's are tripped, T_H is used along with the core exit temperatures and RCS cold leg temperature to measure the temperature rise across the core for verification of core cooling.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

water relief through the pressurizer safety valves. Since prevention of water relief is a goal for abnormal transient operation, rather than an acceptance criterion, the nominal value for the pressurizer level limit is not required to be adjusted for instrument error. The analysis performed to substantiate the 290" upper limit (Ref. 3) assumed the reactor tripped on high RCS pressure (consistent with historical assumptions for this event). Had the anticipatory reactor trip (ART) on loss of both feedwater pumps been modeled, the reactor would have tripped much sooner in the event, terminating the nuclear chain reaction sooner, thereby limiting RCS heatup (and insurge). Thus, there is margin in the analysis to substantiate the use of the nominal value as acceptance criterion, however additional margin is conservatively applied by administratively adjusting the limit for level measurement uncertainty (Ref. 4).

Evaluations performed for the design basis large break loss of coolant accident (LOCA), also assume the maximum level assumed for the LMFW event. The pressurizer level assumed for the LOCA is the partial basis for the volume of reactor coolant released to the containment following the accident. The containment analysis performed using the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits. Parametric evaluations of this analysis indicate the sensitivity to pressurizer volume is small.

The requirement for redundant emergency power supplies is based on NUREG-0578 (Ref. 1). The intent is to allow maintaining the reactor coolant in a subcooled condition with natural circulation for an undefined, but extended, time period after a loss of offsite power.

The maximum pressurizer water level limit satisfies Criterion 2 of the NRC Policy Statement. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0578 (Ref. 1), is the reason for providing a limit on this feature.

LCO

For the pressurizer to be OPERABLE, water level must be maintained ≤ 290 inches and a minimum of 252 kW of pressurizer heaters are to be capable of being powered from each emergency power supply. Limiting the maximum operating water level preserves the steam space for pressure control and ensures the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients.

(continued)

BASES

LCO (continued)

The minimum heater capacity required is sufficient to maintain the system adequately subcooled when accounting for heat losses through the pressurizer insulation and minimal margin for pressurizer steam space leakage. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The capability to provide a minimum of 252 kW of pressurizer heaters from each emergency power supply requires that both an adequate amount of heaters and the circuits that supply power from each of the emergency power supplies to the pressurizer heaters are OPERABLE. The 480 V buses and supply breakers listed in Table B 3.4.8-1 must be available in order to satisfy this LCO.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, the Applicability has been designated for MODES 1 and 2. For additional conservatism, the Applicability is also extended down to include MODE 3.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Decay Heat Removal System is in service. Therefore the LCO is not applicable in the MODES.

ACTIONS

A.1

With pressurizer water level in excess of the maximum limit, action must be taken to restore pressurizer level to within the bounds assumed in the analysis. The 1 hour Completion Time is considered reasonable for adjusting makeup and letdown or taking level control to hand and decreasing level to within limit.

B.1

If there is < 252 kW of heaters capable of being powered from each emergency power supply, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the low probability of a loss of offsite power during this period. Pressure control will be maintained during this time using normal non-1E powered heaters.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

If pressurizer heater capability or water level cannot be restored within the allowed Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within 6 hours and MODE 4 within the following 6 hours. The Completion Times are reasonable, based on operating experience, to reach the specified MODES from full power conditions in an orderly manner and without challenging plant systems.

In the case of water level, reducing THERMAL POWER and RCS Tave will tend to restore level and also reduce the thermal energy of the reactor coolant mass for potential LOCA mass and energy releases.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires that pressurizer water level be monitored every 12 hours in order to verify operation is maintained below the nominal upper limit. The 12 hour Frequency has been shown by operating experience to be sufficient to regularly assess the level for deviations and trends, and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.8.2

This SR verifies minimum redundant pressurizer heater capacity is capable of being powered from its associated emergency power supply. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 24 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. NUREG-0578, July 1979, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations."
2. NUREG 0737, "Clarification of TMI Action Plan Requirements", November, 1980.
3. B&W Topical Report 51-1200406-00, January 1991.
4. Calculation M97-0064, Revision 3, "Pressurizer Level vs. Tave for Power Operations."

Table B 3.4.8-1
Pressurizer Heater Emergency Power Source Circuit Components

TYPE	TRAIN A	TRAIN B
480 V Buses	Reactor Aux Bus 3A	Plant Aux Bus 3 Reactor Aux Bus 3B
Breakers	3321 3395 3355	3222 3312 3392 3396 3356
Pressurizer Heater MCC Breakers	Pressurizer Heater MCC 3A	Pressurizer Heater MCC 3B
	1A - Pressurizer Heater Control Transformer A-1 2A - Pressurizer Heater Control Transformer A-2 1C - Pressurizer Heater Group 7 2C - Pressurizer Heater Group 8 3C - Pressurizer Heater Group 9	1A - Pressurizer Heater Control Transformer B-1 1B - Pressurizer Heater Control Transformer B-2 1D - Pressurizer Heater Group 10 2C - Pressurizer Heater Group 11 3C - Pressurizer Heater Group 12 4C - Pressurizer Heater Group 13

BASES

APPLICABLE SAFETY ANALYSES

The DBAs analyzed for dose consequences that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE so that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure (P_a) following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. L_a is 0.25% of containment air weight per day and P_a is 54.2 psig, resulting from the limiting design basis LOCA.

The dose acceptance criteria applied to DBA releases of radioactive material to the environment are given in 10 CFR 50.67 (Ref. 4).

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

(continued)

BASES

LCO (continued)

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time (Ref. 5). This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is 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 containment.

Each air lock contains equalization valves that are operated in conjunction with the associated air lock doors. The equalization valves are integral to the air lock design and are driven by the mechanical operating system associated with the air lock door. OPERABILITY of an air lock door requires the associated equalization valve to be OPERABLE. Therefore OPERABILITY of the equalization valves is addressed by this LCO and not by LCO 3.6.3, "Containment Isolation Valves." A failure of the equalization valve would result in the associated air lock door being declared inoperable and LCO 3.6.2 Condition A would be applicable.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component or for emergencies involving personnel safety. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). In this context, repairs include follow-up actions to an initial failure of the air lock door seal SR in order to determine which air lock door(s) is faulty. There are circumstances where an at-power containment entry would be required during the period of time that one air lock was inoperable. In this case, entry would be made through the OPERABLE air lock if ALARA conditions permit. However, the

(continued)

BASES

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the requirements related to the control of leakage from containment during major accidents. This LCO is intended to ensure the containment leakage rates do not exceed the values assumed in the safety analysis. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring containment isolation is applicable to this LCO.

The DBAs analyzed for dose consequences that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 3). In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential leakage paths to the environment through containment isolation valves (including containment purge valves) are minimized.

The dose acceptance criteria applied to accidental releases of radioactive material to the environment are given in 10 CFR 50.67 (Ref. 8).

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times. SR 3.3.5.4 addresses the response time testing requirements.

The single-failure criterion required in the safety analyses was considered in the original design of the containment purge valves. Two valves in a series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

The containment purge valves may be unable to close in the environment following a LOCA. Therefore, each of the 48 inch-purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single-failure criterion remains applicable to the containment purge valves because of failure in the control circuit associated with each valve. Again, the 48 inch purge system valve design prevents a single failure from compromising containment OPERABILITY as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to control of containment leakage rates during a DBA.

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 48 inch purge valves must be maintained sealed closed in MODES 1, 2, 3 and 4. The valves covered by this LCO are listed in the FSAR (Ref. 4).

The normally closed isolation valves are considered OPERABLE when manual valves are closed, check valves have flow through the valve secured, blind flanges are in place, and closed systems are intact.

Purge valves with resilient seals must meet additional leakage rate requirements addressed as part of this Specification. All other containment isolation valve leakage rate testing is addressed by LCO 3.6.1, "Containment," as part of Type C testing.

(continued)

BASES

LCO (continued)	This LCO provides assurance that the containment isolation valves and purge valves will perform their designated safety functions to control leakage from the containment during accidents.
APPLICABILITY	In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

The following terms are defined for the purpose of implementing this Specification:

- penetration flowpath: The piping which passes through the RB liner such that a portion of the system inside the RB can communicate with the portion outside the RB. A penetration passes through the imaginary plane established by the RB liner.
- unisolated: The state of a penetration flowpath whereby the operating fluid (liquid or gas) of the system is capable of passing freely through the imaginary plane established by the RB liner.
- penetration flow path with two containment isolation valves: Penetrations where the primary flow path is isolated by valves both inside and outside the containment building. The primary flow path does not include test connections, vents or drains.
- penetration flow path with only one containment isolation valve and a closed system: Penetrations where the primary flow path has a closed system on one side of containment and is isolated by a valve on the other. The primary flow path does not include test connections, vents or drains.

The ACTIONS are modified by a Note allowing penetration flow paths, except for 48 inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under

(continued)

BASES

ACTIONS (continued)

administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

Note 2 has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path.

The ACTIONS are further modified by a Note 3, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

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

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable (except for purge valve leakage not within limit), the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the valve used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic

(continued)

ACTIONS

A.1 and A.2 (continued)

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

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

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

B.1 and B.2

With all containment isolation valves in one or more penetration flow paths inoperable (except for 48 inch purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation

(continued)

ACTIONS

B.1 and B.2 (continued)

barrier that cannot failure. Isolation be adversely affected by a single active barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action B.2. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves or those with one containment isolation valve and no closed system. Condition A of this Specification addresses the condition of one containment isolation valve inoperable in a penetration flow path with two containment isolation valves.

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

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable or the closed system breached, inoperable valve must be restored to OPERABLE status or affected penetration flow path must be isolated. The method of isolation must include the use of at least one

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths utilizing a closed system.

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

(continued)

BASES

ACTIONS (continued)

D.1

In the event one or more containment 48 inch purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits within 24 hours. The specified time is a reasonable period for restoring the valve leakage to within limits, provided overall containment leakage rate remains within limits. With the purge valve seal degraded such that leakage exceeds the limits, there is an increased potential for the same mechanism that caused the initial degradation to cause further degradation. If left unchecked, this could result in a loss of containment OPERABILITY. Thus, the 24 hour Completion Time is necessary to limit the length of time the plant can operate in this condition.

E.1 and E.2

If the Required Actions and associated Completion Times are not met, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 48 inch containment purge valve is required to be verified sealed closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent or spurious opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to maintain offsite doses to within licensing basis limits. Therefore, these valves are required to be in the sealed closed position during MODES 1, 2, 3, and 4. A containment purge valve that is sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 6), related to containment purge valve use during unit operations. In the event purge valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.

SR 3.6.3.2

This SR ensures that the 6 inch post accident hydrogen purge valves are closed as required or, if open, open for an allowable reason. The SR is not required to be met when the post accident hydrogen purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The post accident hydrogen purge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency for verifying valve position is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and is not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for valves outside containment is relatively easy, a 31 day Frequency, based on engineering judgment was chosen to provide added assurance

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3 (continued)

of the correct positions. The SR specifies that valves open under administrative controls are not required to meet the SR during the time the valves are open.

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

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange that is located inside containment and is not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these valves and flanges are typically inaccessible during reactor operation, are operated under administrative controls and the probability of their misalignment is low. The SR specifies that valves open under administrative controls are not required to meet the SR during the time they are open.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.2 (continued)

occurring between surveillances and has been shown to be acceptable through operating experience.

It is preferable to run the fans in slow speed for this SR since this provides additional confidence the post-accident containment cooling train circuitry is OPERABLE.

SR 3.6.6.3

Verifying that each RB spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the RB Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.4

Verifying an emergency design cooling water flow rate of ≥ 1780 gpm to each required containment cooling system heat exchangers (fan cooling coils) ensures the design flow rate assumed in the safety analysis is being achieved. The SR verifies that, with the SW System in the post-accident ES alignment, adequate flow is provided to the heat exchangers to remove the design basis reactor building heat load. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. While the heat exchangers can be aligned to the SW System during normal operations, other critical normal-running SW loads make it impractical to verify accident flow rate to the heat exchangers with the plant on-line. On an ES actuation, these normal-running loads are isolated and the SW flow

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.4 (continued)

normally supplied to them re-directed to the post-accident loads. The 24 month Frequency was also considered acceptable based upon the existence of other Technical Specification Surveillance Requirements. A degradation in heat exchanger performance between performances of this SR would likely be seen as an increase in RB temperature (monitored once per 12 hours in accordance with SR 3.6.5.1). The 1780 gpm cooling water flow rate does not include the flow to the motor cooler or any allowance for instrument flow uncertainty.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic RB spray valve that is not locked, sealed, or otherwise secured in the correct position, actuates to its correct position and that each RB spray pump starts upon receipt of an actual or simulated actuation signal. The 24 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The SR is modified by a note indicating the SR is not applicable in the identified MODE. This is necessary in order to make the requirements for automatic system response consistent with those for the actuation instrumentation.

SR 3.6.6.7

This SR requires verification that each required containment cooling train actuates upon receipt of an actual or simulated actuation signal. In the event of a LOCA, the air steam mixture density is much higher than normal air density. The units are not designed to handle the full flow rate at this condition. To operate the unit at full flow (motor at high speed) at this condition, will cause the motor to overload and trip. To guard the motor from overloading, the volumetric flow rate must be cut

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1 (continued)

Operating experience has shown significant differences between boron measured near the top of the pool and that measured elsewhere. As long as this SR is met, the analyzed events are fully bounded. The 7 day Frequency is acceptable because no major replenishment of pool water is expected to take place over this period of time.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 With Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI-931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

3. Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5% Enriched Mark B-11 Fuel in Checkerboard Arrangement with Water Holes, Holtec Report HI-992285, August 1999.

4. Criticality Evaluation of CR3 Spent Fuel Pool Storage Racks with Mark B-12 Fuel, Holtec Report HI-2022907, September 2002.

5. Progress Energy Engineering Change EC No. 52456, "Documentation of Acceptability to Receive and Store Mk-B/HTP Fuel".

B 3.7 PLANT SYSTEMS

B 3.7.15 Spent Fuel Assembly Storage

BASES

BACKGROUND

This document describes the Bases for the Spent Fuel Assembly Storage which imposes storage requirements upon irradiated and unirradiated fuel assemblies stored in the fuel storage pools containing high density racks. The storage areas, which are part of the Spent Fuel System, governed by this Specification are:

a. Fuel storage pool "A" and
b. Fuel storage pool "B".

In general, the function of the storage racks is to support and protect new and spent fuel from the time it is placed in the storage area until it is shipped offsite.

Spent fuel is stored underwater in either fuel storage pool A or B. Only fuel pool A has the capability to store failed fuel in containers. Spent fuel pool A features high density poison storage racks with a 10 1/2 inch center-to-center distance capable of storing 542 assemblies. Fuel pool A is capable of storing fuel with enrichments up to 5.0 weight percent U-235 (Ref. 1, 6, 7 and 8) without exceeding the criticality criteria of Reference 3 providing the fuel has sufficient burnup. New fuel will be placed into pool A only.

Spent fuel pool B also contains high density racks having a 9.11 inch center-to-center distance capable of storing 932 assemblies. Fuel pool B is capable of storing fuel with enrichments up to 5.0 weight percent U-235 (Ref. 2, 7 and 8) without exceeding the criticality criteria of Reference 3, providing the fuel has sufficient burnup and required storage configuration. New fuel will not be placed into pool B.

(continued)

BASES

BACKGROUND (continued)

Both of the spent fuel pools are constructed of reinforced concrete and lined with stainless steel plate. They are located in the fuel handling area of the auxiliary building.

New fuel storage requirements are addressed in Section 4.0, "Design Features".

APPLICABLE SAFETY ANALYSES

The function of the spent fuel storage racks are to support and protect spent fuel assemblies from the time they are placed in the pool until they are shipped offsite. The spent fuel assembly storage LCO was derived from the need to establish limiting conditions on fuel storage to assure sufficient safety margin exists to prevent inadvertent criticality. The spent fuel assemblies are stored entirely underwater in a configuration that has been shown to result in a reactivity of less than or equal to 0.95 under worse case conditions (Ref. 1, 2, 6, 7 and 8). The spent fuel assembly enrichment requirements in this LCO are required to ensure inadvertent criticality does not occur in the spent fuel pool.

Inadvertent criticality within the fuel storage area could result in offsite radiation doses exceeding 10 CFR 50.67 limits.

The spent fuel assembly storage satisfies Criterion 2 of the NRC Policy Statement.

LCO

Limits on the new and irradiated fuel assembly storage in high density racks were established to ensure the assumptions of the criticality safety analysis of the spent fuel pools is maintained.

Limits on initial fuel enrichment and burnup for both new and for spent fuel stored in pool A have been established. Two limits are defined:

1. Initial fuel enrichment must be less than or equal to 5.0 weight percent U-235, and

(continued)

BASES

LCO (continued)

2. For new, low irradiation, and spent fuel with initial enrichment less than or equal to 5.0 weight percent and greater than or equal to 3.5 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-1. Figure 3.7.15-1 presents two areas of required fuel assembly burnup as a function of initial enrichment. For fuel with enrichment-burnup combinations in the area above the curve, there are no restrictions on where the fuel can be stored. For fuel with enrichment-burnup combinations below the curve, the fuel must be stored in a one-out-of-two checkerboard configuration with water cells that contain no fuel. The acceptability of storing this fuel in the checkerboard configuration is documented in References 6, 7 and 8.

Fuel enrichment limits are based on avoiding inadvertent criticality in the spent fuel pool. The CR-3 spent fuel storage system was initially designed to a maximum enrichment of 3.5 weight percent. Enrichments of up to 5.0 weight percent are permissible for storage in spent fuel pool A as long as the fuel burnup is sufficient to limit the worst case reactivity in the storage pool to less than or equal to 0.95. Fuel burnup reduces the reactivity of the fuel due to the accumulation of fission product poisons. Reference 1 documents that the required burnup varies linearly as a function of enrichment with 10500 megawatt days per metric ton uranium (Mwd/mtU) required for fuel with 5.0 weight percent enrichment and 0 burnup required for 3.5 weight percent enriched fuel.

Similar types of restrictions have been established for Pool B.

1. Initial fuel enrichment must be ≤ 5.0 weight percent U-235, and

2. For fuel with initial enrichment ≤ 5.0 weight percent and ≥ 2.0 weight percent, fuel burnup must be within the limits specified in Figure 3.7.15-2.

(continued)

BASES

LCO (continued)	Fuel with burnup-enrichment combinations in the area above the upper curve has no restrictions on where it can be stored. Fuel with burnup-enrichment combinations in the area between the lower and upper curves must be stored in the peripheral cells of the pool. The peripheral cells are those that are adjacent to the walls of the spent fuel pool. Fuel with burnup-enrichment combinations in the area below the lower curve cannot be stored in Pool B, but must be stored in Pool A.
APPLICABILITY	In general, limiting fuel enrichment of stored fuel prevents inadvertent criticality in the storage pools. Inadvertent criticality is dependent on whether fuel is stored in the pools and is completely independent of plant MODE. Therefore, this LCO is applicable whenever any fuel assembly is stored in high density fuel storage locations.
ACTIONS	<u>A.1</u> Required Action A.1 is modified by a Note indicating LCO 3.0.3 does not apply. Since the design basis accident of concern in this Specification is an inadvertent criticality, and since the possibility or consequences of this event are independent of plant MODE, there is no reason to shutdown the plant if the LCO or Required Actions cannot be met. When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure 3.7.15-1 or Figure 3.7.15-2, immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance. The Immediate Completion Time underscores the necessity of restoring spent fuel pool fuel loading to within the initial assumptions of the criticality analysis.

(continued)

BASES

ACTIONS

A.1 (continued)

The ACTIONS do not specify a time limit for completing movement of the affected fuel assemblies to their correct location. This is not meant to allow an unnecessary delay in resolution, but is a reflection of the fact that the complexity of the corrective actions is unknown.

SURVEILLANCE REQUIREMENTS

SR 3.7.15.1

Verification by administrative means that initial enrichment and burnup of fuel assemblies in accordance with Figure 3.7.15-1 and Figure 3.7.15-2 is required prior to storage of spent fuel in storage pool A or pool B (as applicable). This surveillance ensures that fuel enrichment limits, as specified in the criticality safety analyses (Ref. 1, 2, 6, 7 and 8), are not exceeded. The surveillance Frequency (prior to storage in high density region of the fuel storage pool) is appropriate since the initial fuel enrichment and burnup cannot change after removal from the core.

REFERENCES

1. Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 with Fuel of 5.0% Enrichment, S. E. Turner, Holtec Report HI 931111, December 1993.

2. Criticality Safety Analysis of the Westinghouse Spent Fuel Storage Racks in Pool B of Crystal River Unit 3, S. E. Turner, Holtec Report HI-992128, May 1999.

3. NUREG 0800, Standard Review Plan, Section 9.1.1 and 9.1.2, Rev. 2, July 1981.

4. 10 CFR 50.67.

5. CR-3 FSAR, Section 9.6.

6. Criticality Safety Analysis of the Crystal River Unit 3 Pool A for Storage of 5% Enriched Mark B-11 Fuel in Checkerboard Arrangement With Water Holes, S. E. Turner, Holtec Report HI-992285, August 1999.

7. Criticality Evaluation of CR3 Spent Fuel Pool Storage Racks with Mark B-12 Fuel, Holtec Report HI-2022907, September 2002.

8. Progress Energy Engineering Change EC No. 52456, “Documentation of Acceptability to Receive and Store Mk-B/HTP Fuel”.

B 3.7 PLANT SYSTEMS

B 3.7.18 Control Complex Cooling System

BASES

BACKGROUND	The Control Complex Cooling System provides temperature control for the control room and other portions of the Control Complex containing safety related equipment. The Control Complex Cooling System consists of two redundant chillers, associated chilled water pumps, and parallel duct mounted air heat exchangers that can receive chilled water from either chilled water pump. A train consists of a chiller and associated chilled water pump as well as a duct mounted heat exchanger that provide cooling of recirculated control complex air. The design of the Control Complex Cooling System contains features that allow either train chiller and associated chilled water pump to provide cooling capability to either duct mounted heat exchanger. Redundant chillers and chilled water pumps are provided for suitable temperature conditions in the control complex for operating personnel and safety related control equipment. The Control Complex Cooling System maintains the nominal temperature between 70°F and 80°F. A single chiller and associated chilled water pump will provide the required heat removal for either duct mounted heat exchanger. The Control Complex Cooling System operation to maintain control complex temperature is discussed in the FSAR, Section 9.7 (Ref. 1).
APPLICABLE SAFETY ANALYSIS	The Control Complex Cooling System consists of redundant, safety related components, with some common piping. The Control Complex Cooling System maintains the temperature between 70°F and 80°F. A single active failure of a Control Complex Cooling System component does not impair the ability of the system to perform as designed. The Control Complex Cooling System is designed in accordance with Seismic Category I requirements. The Control Complex Cooling System is capable of removing heat loads from the control room and other portions of the Control Complex containing safety related equipment, including consideration of equipment heat loads and

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

personnel occupancy requirements, to ensure equipment OPERABILITY.

The Control Complex Cooling System satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two redundant trains of the Control Complex Cooling System are required to be OPERABLE to ensure that at least one train is available, assuming a single failure disables one redundant component. A Control Complex Cooling train consists of a chiller and associated chilled water pump as well as a duct mounted heat exchanger that provides cooling of recirculated control complex air. All components of an OPERABLE train must be energized by the same train electrical bus. Total system failure could cause control complex equipment to exceed its operating temperature limits. In addition, the Control Complex Cooling System must be OPERABLE to the extent that air circulation can be maintained (See Specification 3.7.12).

APPLICABILITY

In MODES 1, 2, 3, and 4, the Control Complex Cooling System must be OPERABLE to ensure that the control complex temperature will not exceed equipment OPERABILITY requirements.

ACTIONS

A.1

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy and diversity of subsystems, the inoperability of one component in a train does not render the Control Complex Cooling System incapable of performing its safety function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the Control Complex Cooling System. The intent of this Condition is to maintain a combination of equipment such that the cooling capability equivalent to 100% of a single train remains available and in operation. The word available means: The functional system placed in service does not have to meet the requirement of an operable train. The functional system will have to provide at least 100% of the cooling capability of a single operable Control Complex Cooling train. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

(continued)

LCO (continued)

The 230 kV and 500 kV substations, while part of the offsite network, are not considered part of the circuit required by this LCO. The OPERABILITY of the circuit is supported by the substation provided the substation is capable of supplying the required post accident loads. Each EDG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ES bus on detection of bus undervoltage. This must be accomplished within 10 seconds. Each EDG must also be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ES buses. These capabilities are required to be met from a variety of initial conditions, such as the EDG in standby with the engine hot and the EDG in standby with the engine at ambient conditions. Proper sequencing of loads, including shedding of non-essential loads, is a required function for EDG OPERABILITY.

EDG OPERABILITY requires proper ventilation using EDG Air Handling System cooling fan(s) for each EDG in order to maintain the temperature of the EDG engine room and EDG control room within manufacturer's limits. Based on analysis, single fan or dual fan operation is acceptable dependent upon fan supply air temperature.

The AC sources in one train must be separate and independent (to the extent possible) of the AC sources in the other train. For the EDGs, separation and independence are complete. For the offsite AC sources, separation and independence exist to the extent practical. A circuit may be connected to more than one ES bus and not violate separation criteria. A circuit that is not connected to an ES bus is required to have the capability for the operator to transfer power to the ES buses in order to be considered OPERABLE.

APPLICABILITY

Two onsite and two offsite AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of anticipated operational occurrences (AOOs) or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

(continued)

BASES

APPLICABILITY (continued)	AC power requirements for MODES 5 and 6 are addressed in LCO 3.8.2, "AC Sources- Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis.

Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met (Condition F). However, if the remaining required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition C, for two offsite circuits inoperable, is entered.

A 2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated EDG will not result in a complete loss of safety function of redundant required features. These features are powered from the redundant AC electrical power train. Single train systems (from an electrical perspective), such as the turbine driven emergency feedwater pump, are not included.

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

a. The train has no offsite power supplying its loads; and

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

(continued)

ACTIONS

A.2 (continued)

If at any time during the existence of Condition A (one offsite circuit inoperable) both 'a' and 'b' above become met, this Completion Time begins to be tracked.

The remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E distribution system. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

A.3

According to the recommendations of Regulatory Guide 1.93 (Ref. 6), operation with one required offsite circuit inoperable should be limited to a period of time not to exceed 72 hours. In this condition, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. However, the remaining OPERABLE offsite circuit and EDGs are adequate to supply electrical power to the onsite Class 1E distribution system.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The 6 day (17 days with the alternate AC source available) Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failure to meet the LCO. If Condition A is entered while, for instance, an EDG is inoperable and that EDG is subsequently returned to OPERABLE status, LCO 3.8.1 may already have been not met for up to 14 days. This could lead to a total of 17 days, since initial failure to meet the LCO, to restore the offsite circuit.

(continued)

ACTIONS

A.3 (continued)

The 6 day and 17 day Completion Times provide limits on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently.

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

B.1

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

B.2

Required Action B.2 is intended to provide assurance that a loss of offsite power, during the period that a EDG is inoperable, does not result in a complete loss of safety function of critical redundant required features. These features are designed with redundant safety related trains. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable EDG. Single train systems (from an electrical perspective), such as the turbine driven emergency feedwater pump, are not included.

(continued)

BASES

ACTIONS B.2 (continued)

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

a. An EDG is inoperable; and

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

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

Declaring the required features inoperable within four hours from the discovery of items 'a' and 'b' existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the plant to transients associated with shutdown.

In this Condition, the remaining OPERABLE EDG and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. Thus, on a component basis, single-failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS

B.3.1 and B.3.2

Required Action B.3.1 provides an option to testing the OPERABLE EDG in order to avoid unnecessary testing. If it can be determined that the cause of the inoperable EDG does not exist on the OPERABLE EDG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other EDG, the other EDG would be declared inoperable upon discovery and Condition E of LCO 3.8.1 would be entered. If the common cause failure evaluation is indeterminate (the cause of the initial inoperable EDG cannot be confirmed not exist on the remaining EDG), performance of SR 3.8.1.2 is adequate to provide assurance of continued OPERABILITY of that EDG.

The Completion Time of 24 hours is reasonable to confirm that the OPERABLE EDG is not affected by the same problem as the inoperable EDG and is based on the recommendations of Generic Letter 84-15 (Ref. 7).

B.4

According to the recommendations of Regulatory Guide 1.93 (Ref. 6), operation with one EDG inoperable should be limited to a period not to exceed 72 hours. The completion time may be extended to 14 days if alternate AC (AAC) power is available or on a one-time basis as described in the footnote to the Completion Time. The alternate AC source must be capable of being aligned to the same bus as the inoperable EDG and must be capable of supporting loads required for safe shutdown of the reactor.

In Condition B, the remaining OPERABLE EDG, AAC source and offsite circuits are adequate to supply electrical power to the onsite Class 1E distribution system. The Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, the ability to perform online preventative maintenance, and the low probability of a DBA occurring during this period.

During online preventative maintenance that is planned to take over 72 hours, the following compensatory measures will be put in place prior to initiating the activity:

(continued)

ACTIONS (continued)

B.4 (continued)

Availability will be assured during an extended EDG AOT by the following:

- Starting the AAC and assuring proper operation prior to removing the EDG from service,
- Verifying every 72 hours that a 24-hour fuel supply is onsite, and
- Ensuring the AAC is electrically and mechanically ready for manual operation and can be aligned to supply the applicable safety-related bus with simple operator action every 72 hours.

CR-3 will perform procedure CP-253, "Power Operation Risk Assessment and Management," which requires both a deterministic and probabilistic evaluation of risk for the performance of all maintenance activities. This procedure uses the Level 1 PSA model to evaluate the impact of maintenance activities on CDF. CR-3 will not plan any maintenance that results in "Higher Risk" (Orange Color Code) during EDG maintenance.

ECCS equipment, emergency feedwater, Control Complex Cooling and auxiliary feedwater (FWP-7 and MTDG-1) will be designated administratively as "protected" (no planned maintenance or discretionary equipment manipulation). The term "discretionary equipment manipulation" is not intended to preclude manipulations required for normal operation of the plant, required surveillances or operator response to abnormal conditions.

Prior to initiating a planned EDG outage, CR-3 will verify the availability of offsite power to the 230 kV switchyard and ensure that the capability to power both ES busses is available from each of the two ES offsite power transformers (OPT and BEST).

CR-3 will not initiate an EDG extended preventive maintenance outage if adverse weather, as designated by Emergency Preparedness procedures, is anticipated.

No elective maintenance will be scheduled in the switchyard that would challenge the availability of offsite power to the ES busses.

(continued)

ACTIONS (continued)

B.4 (continued)

A periodic fire watch will be established in fire areas that are considered risk-significant by the IPEEE, affect both EDGs or have increased risk significance due to EDG maintenance. The fire areas are listed in Table B 3.8.1-1.

The 17-day Completion Time for Required Action B.4 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failure to meet the LCO. Refer to the Bases for Required Action A.3 for additional information on this Completion Time.

(continued)

Table B 3.8.1-1

FIRE ZONE	EGDG -1A	EGDG -1B	ZONE DESCRIPTION	AUTO SUPPRESSION	FIRE WRAP	ZONE IGNITION FREQUENCY
AB-119-6A	x	x	NORTH HALLWAY	Wet-Pipe Sprinkler-dual level	1 hour, 3 hour	9.73E-05
AB-119-6E	x	x	EAST HALLWAY	Wet-Pipe Sprinkler-dual level	1 hour, 3 hour	1.73E-04
AB-119-6J (3)		x	CENTRAL HALLWAY	Wet-Pipe Sprinkler	3 hour	2.36E-04
AB-119-6K (2)	x		DECONTAMINATION ROOM	Wet-Pipe Sprinkler		1.02E-04
AB-119-7A	x	x	EMERGENCY DIESEL GENERATOR CONTROL ROOM 3B	Pre-Action Sprinkler	1 hour	1.73E-04
AB-119-7B (3)		x	EMERGENCY DIESEL GENERATOR ROOM 3B	Pre-Action Sprinkler		5.30E-03
AB-119-8A (2)	x		EMERGENCY DIESEL GENERATOR CONTROL ROOM 3A	Pre-Action Sprinkler		1.02E-04
AB-119-8B (2)	x		EMERGENCY DIESEL GENERATOR ROOM 3A	Pre-Action Sprinkler		5.30E-03
CC-108-102 (1)	x	x	HALLWAY AND REMOTE SHUTDOWN ROOM	None	3 hour	1.20E-04
CC-108-103	x	x	PLANT BATTERY ROOM 3B	None	3 hour	9.73E-05
CC-108-104	x	x	PLANT BATTERY ROOM 3A	None	3 hour	9.73E-05
CC-108-105 (1)	x	x	BATTERY CHARGER ROOM 3B	None	3 hour	4.03E-04
CC-108-106 (1)	x	x	BATTERY CHARGER ROOM 3A	None	3 hour	3.68E-04
CC-108-107 (1,3)		x	4160V ES SWITCHGEAR BUS ROOM 3B	None	3 hour	2.27E-04
CC-108-108 (1)	x	x	4160V ES SWITCHGEAR BUS ROOM 3A	None	3 hour	2.60E-04
CC-108-109 (1)	x	x	INVERTER ROOM 3B	None	3 hour	2.14E-04
CC-108-110	x	x	INVERTER ROOM 3A	None	3 hour	1.90E-04
CC-124-111 (1)	x	x	CRD & COMMUNICATION EQUIP ROOM	Wet-Pipe Sprinkler	1 hour, 3 hour	5.06E-04
CC-124-116 (3)		x	480V ES SWITCHGEAR BUS ROOM 3B	None	3 hour	1.90E-04
CC-124-117 (1,2)	x		480V ES SWITCHGEAR BUS ROOM 3A	None		2.04E-04
CC-134-118A (1)	x	x	CABLE SPREADING ROOM	Total Flooding Halon Room		9.73E-05
CC-145-118B (1)	x	x	CONTROL ROOM	None		1.24E-04

(1) Fire zone identified as risk significant per IPEEE
(2) Fire zone may have increased significance when EGDG-1B is in maintenance
(3) Fire zone may have increased significance when EGDG-1A is in maintenance

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Required Action C.1, which applies when both required offsite circuits are inoperable, is intended to provide assurance that a DBA, coincident with a worst-case single failure, will not result in a complete loss of redundant required safety functions. The Completion Time for declaring the redundant required features inoperable is 12 hours; reduced from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is no longer valid, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. Single train features (from an electrical perspective), such as the turbine driven emergency feedwater pump, are not included.

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

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

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

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

In order to ensure that the EDG is tested under load conditions that are as close to design basis conditions as possible, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the EDG would experience.

This SR is modified by two Notes. The reason for Note 1 is that during power operation, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following periodic governor replacement, corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed 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. However, the Note recognizes that should an unplanned event occur in MODES 1 or 2, following verification that the acceptance criteria of the SR are met, the event can be credited as a successful performance of this SR. Note 2 acknowledges this SR may be performed using component loads or it may be performed by paralleling the EDG with offsite power. When the SR is performed with the EDG carrying the 4160 Volt ES bus, the power factor of the EDG is a function of the reactive component of the loads powered from it, and as such, is not under direct control of the operator.

SR 3.8.1.9

Per the recommendations of Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(2), each EDG is required to demonstrate

(continued)

BASES

SURVEILLANCE

SR 3.8.1.9 (continued)

proper operation for the DBA loading sequence to ensure that voltage and frequency are maintained within the required limits. Under accident conditions prior to connecting the EDGs to their respective buses, all loads are shed except load block 1 feeder breakers that power Class 1E loads (referred to as "permanently connected" loads). Upon reaching the required voltage and frequency, the EDGs are auto-connected to their respective 4160 V buses. Loads are then sequentially connected to the bus by the automatic load sequencing relays. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the EDGs due to high motor starting currents.

The 10% load sequence time interval tolerance ensures that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ES equipment response times are not violated. Reference 2 provides a summary of the automatic loading of ES buses.

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

SR 3.8.1.10

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

This Surveillance demonstrates the EDG operation during a loss of offsite power actuation test signal in conjunction with an ES actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG 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 Frequency of 24 months takes into consideration plant conditions needed to perform the Surveillance.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs may be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations for EDGs. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and potentially challenge safety systems. However, Note 2 acknowledges that should an unplanned event occur in MODES 1, 2 or 3, following verification that the acceptance criteria of the SR are met, the event can be credited as a successful performance of this SR. Note 3 is an SR 3.0.4 type allowance to place the plant in MODE 4 for the purposes of performing this Surveillance. This is necessary in order to establish the pre-requisite plant configuration needed to perform the SR.

SR 3.8.1.11

This Surveillance demonstrates the EDGs are capable of synchronizing and accepting a load greater than or equal to the maximum expected steady state accident loads, which are the automatically connected accident loads and required manually applied accident loads. However, the upper limit of the 200 hour service rating is still available for flexibility in post accident EDG load management, including short duration loads. The test load band is provided to avoid routine overloading of the EDGs. Routine overloading may result in more frequent teardown inspections, in accordance with vendor recommendations, in order to maintain EDG OPERABILITY.

The 60 minute run time is provided to stabilize the engine temperature. This ensures that cooling and lubrication are adequate for extended periods of operation.

The 24 month Frequency takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. The reason for Note 2 is that during

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

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. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following periodic governor replacement, corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed 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. However, the Note acknowledges that credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3, July 1993.
4. FSAR, Chapter 6.
5. FSAR, Chapter 14.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. 1, August 1977.
10. Regulatory Guide 1.137, Rev. 1, October 1979.
11. ANSI C84.1-1982.
12. ASME, Boiler and Pressure Vessel Code, Section XI.
13. Deleted.

ACTIONS (continued)

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

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable Since this option may involve undesired administrative efforts, alternative conservative actions are provided. With the required offsite circuit inoperable, the minimum required diversity of AC power sources is not available. In this condition it is required to take actions to minimize the probability or the occurrence of postulated events. This is done by suspending CORE ALTERATIONS and initiating action to suspend operations involving positive reactivity additions. Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. Additionally, the Required Action to initiate action to suspend positive reactivity additions does not preclude actions to maintain or reduce RCS temperature, or to maintain or increase RCS inventory provided the required SDM is maintained.

Notwithstanding performance of the conservative Required Actions, the plant is still without sufficient AC power sources to operate in a safe manner. Therefore, action must be initiated to restore the minimum required AC power sources and continue until the LCO requirements are restored. The restoration of the required AC electrical power sources should be completed in a timely manner in order to minimize the time during which plant safety systems may be without sufficient power.

The immediate Completion Time of these ACTIONS is consistent with others requiring prompt attention.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from Specification 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.7 is not required to be met because with only one offsite circuit required to be OPERABLE, there will not be an alternate

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1 (continued)

source to transfer to. SR 3.8.1.9 and SR 3.8.1.10 are also not required to be met since the ability to respond to ES actuations in other than MODES 1, 2, 3 or 4 is not a requirement for CR-3.

This SR is modified by a Note. The NOTE indicates SRs 3.8.1.3, 3.8.1.8, and 3.8.1.11 are not required to be performed to comply with SR 3.8.2.1. The reason for the Note is to preclude situations in which the OPERABLE EDG would be paralleled with the offsite power network or rendered inoperable during performance of SRs. With limited AC sources available, a single event could compromise both the required offsite circuit and the EDG. It is the intent that these SRs must still be capable of being met, but actual performance is not required. Refer to the Bases for LCO 3.8.1 for a discussion of each SR.

Additionally, the impact of entering the ACTIONS of this Specification during Surveillance testing was evaluated as part of ITS implementation and a unique situation was discovered relative to the monthly EDG testing performed as part of surveillance procedures SP-354A and SP-354B and the MODES and Conditions addressed by this Specification. The EDG tear down and inspection may be performed during MODES 5 and 6 outages. Thus, it was concluded CR-3 could coincidentally have only one OPERABLE EDG and be required to perform SP-354. This created a concern since certain portions of SP-354 render the EDG inoperable, requiring entry into the ACTIONS of this Specification. Required Action B.3 requires action be initiated immediately to restore the EDG to OPERABLE status. Thus, a situation is established in which SP-354 is performed, the ACTIONS are entered, SP-354 is immediately terminated (in order to restore OPERABILITY of the EDG). In this case, SP-354 could never be performed.

The need to perform SP-354 was evaluated in light of this concern. It was decided that since the periods of inoperability for the performance of the portions of SP-354 associated with SR 3.8.1.2 and 3.8.1.6 were brief and the SRs provided confidence in the OPERABILITY of the EDG, the SP should be performed. Expeditiously performing the portions of SP-354 associated with SR 3.8.1.2 and 3.8.1.6 during these MODES satisfies the intent of Required Action B.3 and is therefore acceptable. For MODE 5 and 6 surveillance testing performed in accordance with SP-354 to meet SR 3.8.1.3 will be performed at the frequency of SR 3.8.1.3 only when both EDG trains are operable.

REFERENCES

None.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

An accident which occurs during movement of recently irradiated fuel assemblies within containment will have any released radioactivity limited from escaping to the environment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, the requirement to isolate the containment from the outside atmosphere is less stringent than those established for MODES 1 through 4. In order to make this distinction, the penetration requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths for radioactivity are closed or capable of being closed by an OPERABLE containment purge or mini-purge valve.

The containment equipment hatch or outage equipment hatch (OEH) provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch or OEH must be held in place by at least four bolts. The required number of bolts is based on dead weight and is acceptable due to the low likelihood of a pressurization event. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door in the OEH (if installed) must always remain closed.

The containment air locks provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. However, during periods of unit shutdown when containment OPERABILITY is not required, the door interlock mechanism may be disabled, allowing both doors of an installed air lock to remain open for extended periods when frequent containment ingress and egress is necessary. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed.

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BASES

BACKGROUND (continued)

The requirements on containment penetration closure ensure that a release of fission product radioactivity to the environment from the containment will be limited. The closure restrictions are sufficient to limit fission product radioactivity release from containment due to a fuel handling accident involving handling recently irradiated fuel during refueling.

In MODE 6, it is necessary to periodically recirculate/exchange RB atmosphere in order to minimize radiation uptake during the conduct of refueling operations. The 48 inch purge valves are normally used for this purpose, but the mini-purge valves may be relied upon as well. Both valve types are automatically isolated on a unit vent-high radiation signal (from RM-A1). So long as one valve in the flow path is OPERABLE, these lines may remain unisolated during the subject plant conditions.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by a minimum of one isolation device. Isolation may be achieved by an automatic or manual isolation valve, blind flange, or equivalent. Equivalent isolation methods include use of a material (e.g., temporary sealant) that can provide a temporary, atmospheric pressure ventilation barrier for the other containment penetrations during fuel movements involving handling recently irradiated fuel.

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BASES

APPLICABLE SAFETY ANALYSES

During movement of fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling recently irradiated fuel. For Cycle 13 (including Refueling Outage 13) and future Cycles (including Refueling Outages) that are operated at a RATED THERMAL POWER of 2568 mWt, recently irradiated fuel is the fuel that has occupied part of a critical reactor core within the previous 72 hours. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Canal Water Level," in conjunction with the administrative limit on minimum decay time prior to irradiated fuel movement ensure that the release fission product radioactivity subsequent to a fuel handling accident results in doses that are within the requirements specified in 10 CFR 50.67 even without containment closure.

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity from containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere, including the equipment hatch or the Outage Equipment Hatch, to be closed except for penetrations containing an OPERABLE purge or mini-purge valve. For the containment purge and mini-purge valves to be considered OPERABLE, at least one valve in each penetration must be automatically isolable on an RB Purge-high radiation isolation signal.

The definition of "direct access from the containment atmosphere to the outside atmosphere" is any path that would allow for transport of containment atmosphere to any atmosphere located outside the containment structure. This includes the Auxiliary Building. As a general rule, closed or pressurized systems do not constitute a direct path

(continued)

BASES

LCO (continued)	between the RB and outside environments. All permanent and temporary penetration closures should be evaluated to assess the possibility for a release path to the outside environment. For the purpose of determining what constitutes a "direct access" path, no failure mechanisms should be applied to create a scenario which results in a "direct access" path. For example, line breaks, valve failures, power losses or natural phenomenon should not be postulated as part of the evaluation process.
APPLICABILITY	The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving fuel that has not been recently irradiated will result in doses that are well within the guideline values specified in 10 CFR 50.67 even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.
ACTIONS	<u>A.1</u> With the containment equipment hatch, OEH, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere not in the required status, including containment purge or mini-purge valve penetrations not capable of automatic isolation when the penetrations are unisolated, the plant must be placed in a condition in which the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude moving a component to a safe position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within the containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations.

As such, this surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity within containment will not result in a release of significant fission product radioactivity to the environment.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and mini-purge valve actuates to its isolation position on an actual or simulated high radiation signal. The 24 month Frequency is consistent with other similar instrumentation and valve testing requirements. The Surveillance ensures that the valves are capable of closing after a postulated fuel handling accident involving handling recently irradiated fuel to limit a release of fission product radioactivity from the containment. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements.

REFERENCES

1. FSAR, Section 14.2.2.3.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Refueling Canal Water Level

BASES

BACKGROUND

The movement of irradiated fuel assemblies within containment requires a minimum refueling canal water level of 156 ft plant datum. This maintains sufficient water level above the fuel contained in the vessel and the bottom of the fuel transfer canal, and the spent fuel pool to ensure iodine fission product activity is retained in the water to a level consistent with the dose analysis of a fuel handling accident (Ref. 4). Sufficient iodine activity would be retained to limit offsite doses from the accident to well within 10 CFR 50.67 limits (Ref. 3).

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the refueling canal is an assumed initial condition in the analysis of the fuel handling accident in containment. This relates to the assumption that 99% of the total iodine released from the fuel is retained by the refueling canal water. There are postulated drop scenarios where there is < 23 ft above the top of the fuel bundle and the surface. In particular, this is the case for the period of time during which the assembly travels between the cavity and the deep end of the refueling canal. During this time, there is potentially 21 feet of water between the reactor vessel flange (135 ft plant datum) and the surface of the pool. The iodine retention factors used in the dose assessment are still conservative at water levels of 21 feet above the damaged fuel (Ref. 4). The 156 ft value was chosen to be consistent with the level specified for LCO 3.7.13, "Fuel Storage Pool Water Level" and plant configuration.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel handling accident analysis inside containment is described in Reference 4. With a minimum water level of 23 ft above the stored fuel, and the administrative limit on minimum decay time of 72 hours prior to movement of irradiated fuel in the vessel, analyses demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water such that offsite doses are maintained within allowable limits (Ref. 3).

Refueling canal water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

A minimum refueling canal water level of 156 ft plant datum is required to ensure that the radiological consequences of a postulated fuel handling accident inside containment are within acceptable limits. This minimum level also ensures an adequate operational window between the surface of the pool and the transfer winch for the RB fuel handling equipment.

APPLICABILITY

This Specification is applicable when moving irradiated fuel assemblies within the containment. The LCO minimizes the potential of a fuel handling accident in containment which results in offsite doses greater than those calculated by the safety analysis. If irradiated fuel is not present in containment, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Water level requirements for fuel handling accidents postulated to occur in the spent fuel pool are addressed by LCO 3.7.13, "Fuel Storage Pool Water Level."

ACTIONS

A.1

With a refueling canal water level of < 156 ft plant datum, all movement of irradiated fuel assemblies shall be suspended immediately to preclude a fuel handling accident from occurring. The suspension of fuel movement shall not preclude completion of movement of a component to a safe position.

(continued)

BASES

ACTIONS

A.2

In addition to immediately suspending movement of irradiated fuel, actions to restore refueling canal water level must be initiated immediately. The immediate Completion Time is based on engineering judgment. When increasing refueling canal water level the boron concentration of the make-up and the effect of this concentration on the minimum specified in the COLR (Ref. LCO 3.9.1) must be considered.

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum refueling canal water level of 156 ft plant datum ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are assumed to result from a postulated fuel handling accident inside containment (Ref. 2).

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

REFERENCES

1. Deleted.

2. FSAR Section 14.2.2.3.

3. 10 CFR 50.67.

4. FPC Calculation N-00-0001.