Westinghouse Non-Proprietary Class 3

WCAP-16294-NP-A
Revision 1

June 2010

Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-16294-NP-A
Revision 1

Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs

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Work Performed Under Shop Order MUHP-3015 and PA-LSC-0194

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March 29, 2010

Mr. Biff Bradley, Director
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SUBJECT: FINAL SAFETY EVALUATION FOR NUCLEAR ENERGY INSTITUTE TOPICAL REPORT WCAP-16294-NP, REVISION 0, "RISK-INFORMED EVALUATION OF CHANGES TO TECHNICAL SPECIFICATION REQUIRED ENDSTATES FOR WESTINGHOUSE NSSS [NUCLEAR STEAM SUPPLY SYSTEM] PWRs [PRESSURIZED WATER REACTORS]," AUGUST 2005 (TAC NO. MD5134)

Dear Mr. Bradley:

By letter dated September 9, 2005, as supplemented by letters dated December 12, 2007, and November 26, 2008, Nuclear Energy Institute (NEI) submitted Topical Report (TR) WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation Of Changes To Technical Specification Required Endstates For Westinghouse NSSS PWRs," to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated December 10, 2009, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-16294-NP, Revision 0, was provided for your review and comment. By letter dated January 12, 2010, the NEI commented on the draft SE. The NRC staff's disposition of NEI's comments on the draft SE are discussed in Enclosure 2.

The NRC staff has found that TR WCAP-16294-NP, Revision 0, is acceptable for referencing in licensing applications for Westinghouse designed pressurized water reactors to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that NEI publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed final SE after the title page. Also, the accepted version must contain historical review information, including NRC requests for additional information (RAI) and your responses after the title page. The accepted version shall include an "-A" (designating accepted) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.

2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, NEI and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Thomas B. Blount, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No.: 689

Enclosures:
1. Final SE
2. Resolution of NEI Comments

cc w/encl: See next page

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Sincerely,

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Thomas B. Blount, Deputy Director
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ADAMS ACCESSION NO.:
Final SE Letter: ML100770137
Final SE: ML100770146
Comment Resolution Table: ML100770127

***No major changes to SE input.** **NRR-043**

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-16294-NP, REVISION 0, "RISK-INFORMED EVALUATION OF CHANGES TO TECHNICAL SPECIFICATION REQUIRED ENDSTATES FOR WESTINGHOUSE NSSS [NUCLEAR STEAM SUPPLY SYSTEM] PWRs [PRESSURIZED WATER REACTORS]," NUCLEAR ENERGY INSTITUTE

PROJECT NO. 689

1.0 INTRODUCTION AND BACKGROUND

By letter dated September 9, 2005 (Reference 1), the Nuclear Energy Institute (NEI) submitted topical report (TR) WCAP-16294, Rev. 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (Reference 2), for U.S. Nuclear Regulatory Commission (NRC) staff review and approval. The NRC staff issued requests for additional information (RAI) to NEI, by letters dated June 13, 2007 (Reference 3) and October 23, 2008 (Reference 4). Subsequently, by letters dated December 12, 2007 (Reference 5), November 26, 2008 (Reference 6), and November 20, 2009 (Reference 21) the NEI submitted its response, accompanied by proposed markups to TR WCAP-16294-NP. The November 20, 2009, letter contains revisions to the TS and Bases, and additional markups of TR WCAP-16294-NP.

The proposed risk-informed TS changes are intended to maintain or improve safety while reducing unnecessary burden and to make TS requirements consistent with the Commission's other risk-informed regulatory requirements. TR WCAP-16294-NP identifies and evaluates new TS Required Action end states for a number of TS Limiting Conditions for Operation (LCOs), using a risk-informed approach, consistent with Regulatory Guides 1.174 and 1.177 (References 7 and 8). The end states are currently defined based on placing the unit into a Mode or condition in which the TS LCO is not applicable. Mode 5 is the current endstate for LCOs that are applicable in Modes 1 through 4. The risk of the transition from Mode 1 to Modes 4 or 5 depends on the availability of alternating current (AC) sources. During the realignment from Mode 4 to Mode 5, there is an increased potential for loss of shutdown (SD) cooling and loss of inventory events. Decay heat removal following a loss of offsite power event in Mode 5 is dependent on AC power for SD cooling whereas, in Mode 4, the turbine-driven auxiliary feedwater pump will be available. Therefore, transitioning to Mode 5 is not always the appropriate endstate from a risk perspective. Thus, for specific TS conditions, this TR justifies Mode 4 as an acceptable alternate endstate to Mode 5. The proposed change to the TSs would allow time to perform short-duration repairs which would otherwise necessitate exiting the original Mode of operation. The Mode 4 TS end state is applied, and risk is assessed and managed in accordance with 10 CFR 50.65. Modified end states are limited to conditions where: (1) entry into the shutdown mode is for a short interval, (2) entry is initiated by inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical.

1.1 Proposed Action

As summarized in the following table, the requested TS changes would permit an end state of hot shutdown (Mode 4) rather than an end state of cold shutdown (Mode 5) for the following TS action requirements that are contained in STS.

Proposed Changes To End States	
TS Condition	**Title**
3.3.2-B 3.3.2-C 3.3.2-K	Engineered Safety Feature Actuation System Instrumentation
3.3.7-C	Control Room Emergency Filtration System Actuation Instrumentation
3.3.8-D	Fuel Building Air Cleanup System Actuation Instrumentation
3.4.13-B	RCS Operational Leakage
3.4.14-B	RCS Pressure Isolation Valve Leakage
3.4.15-E	RCS Leakage Detection Instrumentation
3.5.3-C	Emergency Core Cooling System (ECCS) – Shutdown
3.5.4-C	Refueling Water Storage Tank
3.6.4A-B	Containment Pressure (Atmospheric, Dual, and Ice Condenser) *
3.6.4B-B	Containment Pressure (Subatmospheric) *
3.6.5A-B	Containment Air Temperature (Atmospheric and Dual) *
3.6.5B-B	Containment Air Temperature (Ice Condenser) *
3.6.5C-B	Containment Air Temperature (Subatmospheric) *
3.6.6A-B 3.6.6A-E	Containment Spray and Cooling Systems (Atmospheric and Dual)
3.6.6B-F	Containment Spray and Cooling Systems (Atmospheric and Dual)
3.6.6C-B	Containment Spray System (Ice Condenser)
3.6.6D-B	Quench Spray System (Subatmospheric)
3.6.6E-F	Recirculation Spray System (Subatmospheric)
3.6.7-B	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)
3.6.8-B	Shield Building (Dual and Ice Condenser) *
3.6.11-B	Iodine Cleanup System (Atmospheric and Subatmospheric)
3.6.12-B	Vacuum Relief Valves (Atmospheric and Ice Condenser)
3.6.13-B	Shield Building Air Cleanup System (Dual and Ice Condenser)
3.6.14-B	Air Return System (Ice Condenser)
3.6.15-B	Ice Bed (Ice Condenser) *
3.6.16-D	Ice Condenser Doors (Ice Condenser) *
3.6.17-C	Divided Barrier Integrity (Ice Condenser) *

Proposed Changes To End States	
TS Condition	**Title**
3.6.18-C	Containment Recirculation Drains (Ice Condenser)
3.7.7-B	Component Cooling Water System
3.7.8-B	Service Water System
3.7.9-C	Ultimate Heat Sink
3.7.10-C	Control Room Emergency Filtration System
3.7.11-B	Control Room Emergency Air Temperature Control System
3.7.12-C	ECCS Pump Room Exhaust Air Cleanup System
3.7.13-C	Fuel Building Air Cleanup System
3.7.14-C	Penetration Room Exhaust Air Cleanup System
3.8.1-G	Alternating Current (AC) Sources – Operating
3.8.4-D	Direct Current (DC) Sources – Operating
3.8.7-B	Inverters – Operating
3.8.9-D	Distribution Systems – Operating

** As noted in the November 26, 2008, letter from the NEI, the proposed endstate changes to these TS were withdrawn and the TR WCAP-16294-NP will be revised to delete the discussions associated with these proposed changes.*

In addition to the items noted in the above table, the NRC staff also reviewed 6.4.22a, "Recirculation Fluid pH Control System."

The request is limited to inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable technical specification, and the primary purpose is to correct the inoperable component(s) and return to power operation as soon as is practical.

1.2 Related NRC Actions

Similar Topical Reports (TR) have been reviewed and approved by the NRC staff for three other plant types. Specifically, TR CE-NPSD-1186 for the Combustion Engineering Owners' Group (CEOG) was approved on July 17, 2001 (Reference 10), TR NEDC-32988 for the Boiling Water Reactor Owners' Group (BWROG) was approved on September 27, 2002 (Reference 11), and TR BAW-2441 for the Babcock and Wilcox Owners' Group (BWOG) was approved on August 25, 2006 (Reference 12).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance which the NRC staff considered in its review of TR WCAP-16294-NP, Revision 0 are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities.

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TS as part of the license. The TS ensure the operational capability of structures, systems and components that are required to protect the health and safety of the public. The Commission's regulatory requirements related to the content of the TS are contained in 10 CFR Section 50.36, Technical specifications. That regulation requires that the TS include items in the following specific categories: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. 10 CFR 50.36(c)(2) Limiting conditions for operation specifies the required end state for non-compliance with TS as: "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow the remedial action permitted by the technical specification until the condition can be met." However, the rule does not specify the particular requirements to be included in plant TSs. The NRC staff's guidance as to the content of TS for Westinghouse plants is established in NUREG-1431, Revision 3.0, "Standard Technical Specifications Westinghouse Plants" (STS) (Reference 9).

10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," requires that the reactor must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in 10 CFR 50.46(b).

10 CFR 50.65(a)(4), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", requires that "Before performing maintenance activities ... the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety." Regulatory Guide (RG) 1.182 (Reference 13) provides guidance on implementing the provisions of 10 CFR 50.65(a)(4) by endorsing a revised Section 11 to NUMARC 93-01 (Reference 14). Section 11 states, "The assessment is required for maintenance activities performed during power operations or during shutdown." Planning and scheduling of maintenance activities during shutdown should consider their impact on performance of key shutdown safety functions.

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

Criterion 16, "Containment Design", requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 35, "Emergency Core Cooling", requires a system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Criterion 38, "Containment Heat Removal" requires the establishment of a containment heat removal system that will rapidly reduce containment pressure and temperature following any loss-of-coolant accident. The containment heat removal system supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following a loss-of-coolant accident thus lessening the challenge to containment integrity. Meeting GDC 38 will help ensure that the containment can fulfill its role as the final barrier against the release of radioactivity to the environment.

Criterion 41, "Containment Atmosphere Cleanup", requires systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Criterion 60, "Control of Releases of Radioactive Materials to the Environment", requires the means to control the release of radioactive materials in gaseous and liquid effluents.

RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 7), describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (Reference 8), describes an acceptable risk-informed approach specifically for assessing proposed permanent AOT and Surveillance Test Interval TS changes. This RG also provides risk acceptance guidelines for evaluating the results of such assessments. RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed Completion Time (CT) TS change, as summarized below. Per RG 1.177, the improved STS use the terminology "completion times" and "surveillance frequency" in place of allowed outage time" and "surveillance test interval."

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174 and RG 1.177.
- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out-of-service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved. The objective of the Tier 2 evaluation is to ensure that appropriate restrictions on risk-significant configurations associated with the changes are in place.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk-significant plant configurations resulting from maintenance or other operational activities and appropriate compensatory measures are taken to avoid risk significant configurations that may not have been considered when the Tier 2 evaluation was performed. The risk impact of out-of-service equipment must be evaluated prior to performing any maintenance activity by the licensee.

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 15). Guidance on evaluating probabilistic risk assessment (PRA) technical adequacy is provided in Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 16). More specific guidance related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decision Making: Technical Specifications," (Reference 17), which includes CT changes as part of risk-informed decision making.

In its ongoing program of improving TS, the NRC continues to consider methods to make use of risk and reliability information for defining and improving generic TS requirements. The NRC policy regarding the use of PRA technology is that the PRA methods and data should complement the NRC's deterministic approach and supports its traditional defense-in-depth philosophy. The PRA and associated analyses should be used to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guidance, license commitments, and staff practices.

In implementing risk-informed decision making, Section 19.2 of the SRP states that a risk-informed application should be evaluated to ensure that the proposed changes meet the following key principles:

- The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When a proposed change results in an increase in core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement, and
- The impact of the proposed change should be monitored using performance measurement strategies.

Defense-in-depth and safety margins are fundamental safety principles on which the plant design is based and cannot be compromised. Design basis accidents, safety margins and defense-in-depth constitute a combination of postulated challenges and failure events that are used in the plant design to demonstrate a safe plant response. Proposed changes must be evaluated to determine their impact on defense-in-depth and safety margin.

In this TR, the proposed changes need to meet the defense-in-depth philosophy that consists of the following elements:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.
- Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of physical barriers is not degraded.
- Defenses against human errors are maintained, and
- The intent of the General Design Criteria in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," is maintained.

3.0 TECHNICAL EVALUATION

The STS for Westinghouse plants defines the following six operational modes. Of specific relevance to TR WCAP-16294-NP, Revision 0, are Modes 4 and 5:

- Mode 1 - Power operation. Reactivity condition is such that k_{eff} [1] ≥ 0.99 and thermal power is greater than 5 percent of the rated thermal power.
- Mode 2- Startup - Reactivity condition is such that $k_{eff} \geq 0.99$ and thermal power is ≤ 5 percent of the rated thermal power.
- Mode 3 - Hot standby - Reactor reactivity condition is such that $k_{eff} < 0.99$ and the average reactor coolant system (RCS) temperature is $\geq 350^{\circ}F$.
- Mode 4 - Hot shutdown - The average RCS temperature is greater than 200°F and less than 350°F. The reactivity condition is such that $k_{eff} < 0.99$. The reactor vessel head closure bolts are fully tensioned.
- Mode 5 - Cold shutdown - The average RCS temperature is less than or equal to 200°F. The reactor vessel head closure bolts are fully tensioned.
- Mode 6 – Refueling - The reactor in this mode is shut down and one or more reactor vessel head closure bolts are less than fully tensioned.

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[1] The ratio of the number of thermal neutrons obtained at the end of the neutron cycle to the number of those initiating the cycle is called the effective multiplication factor and is denoted by k_{eff}. It is a significant parameter of the nuclear chain reaction because its value determines the rate of neutron level multiplication.
If $k_{eff} > 1$, the reactor is supercritical; the neutron population, the fission rate, and energy production are increasing exponentially.
If $k_{eff} = 1$, the reactor is critical; the neutron population is constant, as is the fission rate and the energy production. The nuclear chain reaction is sustained and controlled.
If $k_{eff} < 1$, the reactor is subcritical; the neutron population, the fission rate, and the energy production are decreasing exponentially.

End states for unit conditions are prescribed in the TS when Required Actions are not met or cannot be met. The current TS actions require placing the unit in cold shutdown (Mode 5) based on the expectation that this condition would result in the safest condition, since most design basis accidents and transients either cannot physically occur during shutdown, or would have significantly reduced plant impact and occur much less frequently due to the reduced temperatures and pressures in the plant. Accidents and transients unique to shutdown conditions were anticipated to be of less significance compared to the design bases events applicable to power operation. However, in the late 1980s and early 1990s, the NRC and licensees recognized the potential significance of events occurring during shutdown conditions, and guidance was issued to improve shutdown operation.

The requested change to the TSs is to allow a Mode 4 end state rather than a Mode 5 end state for selected TS LCO actions. TR WCAP-16294-NP, Revision 0, provides a comparative qualitative assessment of the availability of plant equipment for decay heat removal and accident mitigation in Modes 4 and 5, and considers the likelihood and consequences of initiating events which may occur in these modes. A quantitative risk assessment of operation in these modes, including the risk associated with the transition from Mode 4 to Mode 5 and then back to Mode 4 to support the return to service, is also provided using a shutdown and transition PRA model developed to support the review of TR WCAP-16294-NP, Revision 0.

TR WCAP-16294-NP, Revision 0, concludes that the availability of SG heat removal capability in Mode 4, and the avoidance of transitioning the plant to and from SDC, makes Mode 4 the preferred endstate over Mode 5 for each of the proposed TS conditions being changed. This conclusion is further supported by quantitative risk analyses which demonstrate a reduction in plant risk by remaining in Mode 4 compared to the alternative of transitioning to and from Mode 5 in accordance with the existing TS requirements.

Both the qualitative and quantitative analyses of TR WCAP-16294-NP, Revision 0, support a Mode 4 end state. This conclusion is primarily due to the availability of SG cooling in Mode 4 via the turbine-driven Auxiliary Feedwater (AFW) pump which is not reliant upon AC power, compared to the use of SDC in Mode 5 which requires the availability of AC power. Further, the transition risks associated with establishing SDC alignments and the resulting potential for loss of inventory or loss of cooling events due to human error during such alignments are avoided by remaining in Mode 4.

This general assessment is applied as the basis for changing the required endstate from Mode 5 to Mode 4 for those TSs which govern plant equipment that is not included in the PRA models, supported by qualitative assessments of the plant impact of the unavailability of the TS equipment. For those TS covering plant equipment that is included in the PRA models, a quantitative risk assessment is also provided which assesses the comparative risk of completing repairs in Mode 4 or proceeding to Mode 5 for repairs and then returning to Mode 4 for plant startup, considering the available equipment for accident mitigation.

Changing the required endstate to Mode 4 will also result in increased unit availability by decreasing the time of shutdown. The additional time required to transition to Mode 5 from Mode 4 when shutting down and also to Mode 4 from Mode 5 when restarting can be eliminated with the end state change. A typical time for the transition from Mode 4 to Mode 5 during shutdown and from Mode 5 to 4 during startup is 24 hours. Therefore, this change will allow an availability increase of 24 hours.

Changing the end states allows continued operation with the LCO not met by removing the TS requirement to exit the LCO Applicability. In this case the requirements of LCO 3.0.4.a would apply unless otherwise stated. LCO 3.0.4.a allows entry into a MODE or other specified condition in the Applicability with the LCO not met when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

Thus, implementing modified end states requires adding a Note to the affected Required Actions to prevent using the allowance of LCO 3.0.4.a when entering Mode 4 from Mode 5. This is done to avoid unit operation in a condition that should be prohibited by TS since LCO 3.0.4.a allows entry into a mode or other specified condition in the Applicability when the associated Actions to be entered permit continued operation in the Mode or other specified condition in the Applicability for an unlimited period of time. Applying the allowance of LCO 3.0.4.a to modified end states was not analyzed in TR WCAP-16294; therefore, an appropriate limitation is applied by the addition of Notes to the affected TS Required Actions (Reference 21).

3.1 Technical Analysis

This section provides the NRC staff evaluation of the impact of each proposed end state change on defense-in-depth, and safety margins as applied to the corresponding safety systems. The NRC staff's evaluation approves only the proposed changes to the standard technical specifications as described below. The NRC staff finds that the TR used realistic assumptions regarding the plant conditions and the availability of various mitigating systems in analyzing the risks and considering the defense-in-depth and safety margins. Thus the NRC staff concludes that the TR uses realistic assumptions to justify the change in the endstate. However, during the proposed Mode 4 end state, due to the SI signal blockage and nonavailability of accumulators, operator actions will be required to mitigate potential events.

During the proposed Mode 4 end state, risk is assessed and managed consistent with 10 CFR 50.65. The NRC staff's review is based on the knowledge of lower RCS pressure in Mode 4, which reduces the severity of a LOCA, and limits any coolant inventory loss in the event of a LOCA.

Finally, TR WCAP-16294 does not address entry into Mode 4 from Mode 5 when the Required Actions are in effect. Such a mode change would be permissible since the revised actions permit continued operation in Mode 4 for an unlimited period of time, and therefore transitioning from Mode 5 to Mode 4 would be permissible using LCO 3.0.4.a of the Standard TS. Therefore, a note is added to each affected LCO which identifies that the provisions of LCO 3.0.4.a are not applicable to Mode 4 entry.

3.1.1 TS 3.3.2 – Engineered Safety Features Actuation System (ESFAS) Instrumentation

The ESFAS instrumentation initiates necessary safety systems, based on the set point for selected unit parameters to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. The ESFAS instrumentation functions are listed in Table 3.3.2-1 of Reference 2.

Function 1.a Safety Injection - Manual Initiation and 1.b. Safety Injection – Automatic Logic and Actuation Relays.

Description: The safety injection (SI) system provides two primary functions: (1) Primary side water addition to ensure maintenance or recovery of reactor vessel water level (covering the active fuel for heat removal, clad integrity, and for limiting the peak clad temperature to < 2200°F), and (2) Boration to ensure recovery and maintenance of shutdown margin (k_{eff} < 1.0). These functions mitigate the effects of high energy line breaks both inside and outside of containment.

Manual initiation causes actuation of all components in the same manner as any of the automatic actuation signals. The automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation. The LCO for both Manual Initiation and Automatic Actuation Logic and Actuation Relays requires that two trains shall be operable.

Proposed Required Actions: For the Manual initiation, the end state for Required Action B.2.2 is revised to be in Mode 4 in 60 hours instead of in Mode 5 in 84 hours and a Note is added stating that LCO 3.0.4.a is not applicable when entering MODE 4.

For automatic logic and actuation relays, the end state for Required Action C.2.2 is revised to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours. A Note is added to Required Action C.2.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one train is inoperable, the other train is available to initiate SI. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via residual heat removal (RHR), there is increased time for operator actions and mitigation strategies, and there is a lower overall risk than proceeding to Mode 5.

If one channel is inoperable, the other channel is available for the operator to initiate SI. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4.

NRC staff accepts the proposed change to revise Required Actions B.2.2 and C.2.2 so that the plant would be allowed to remain in Mode 4, subject to LCO 3.0.4.a being not applicable for entry into Mode 4.

Function 2.a Containment Spray – Manual Initiation, and 2.b. Containment Spray - Automatic Actuation Logic and Actuation Relays

Description: The containment spray (CS) system provides three primary functions: (1) Lowers containment pressure and temperature after a high-energy line break (HELB) in containment, (2) Reduces the amount of radioactive iodine in the containment atmosphere, and (3) Adjusts the pH of the water in the containment recirculation sump after a Loss of Coolant Accident (LOCA). These functions are necessary to ensure the containment structure pressure boundary, limit the radioactive iodine release to the environment in the event of failure of containment structure, and minimize corrosion of the internal containment systems following a LOCA.

The operator can initiate CS by simultaneously actuating two CS actuation switches in the same train. Two switches are used to prevent inadvertent actuation of CS. Simultaneously actuating the two switches in either train will start both trains of CS.

There are two trains for automatic actuation. In Mode 4, adequate time is available to manually actuate required components in the event of a design basis accident. However, because of the large number of components actuated, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

The LCO for manual Initiation requires that two channels per train and two trains shall be operable. The LCO for Automatic Actuation Logic and Actuation Relays requires that two trains shall be operable.

Proposed Required Actions: For the Manual initiation, the end state for Required Action B.2.2 is revised to be in Mode 4 in 60 hours instead of in Mode 5 in 84 hours and a Note is added to Required Action B.2.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

For automatic logic and actuation relays, the end state for Required Action C.2.2 is revised to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours. A Note is added to Required Action C.2.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one channel or train is inoperable, the other train is available for the operator to initiate CS. A cool down to Mode 4 places the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5. In addition, the containment, containment isolation valves, CS system, and containment cooling system are available.

Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, CS System, and containment cooling system are required to be operable in Mode 4. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4.

The staff accepts the proposed amendment to revise Required Actions B.2.2 and C.2.2 so that the plant would be allowed to remain in Mode 4, subject to LCO 3.0.4.a being not applicable for entry into Mode 4.

Function 3.a (1) Containment Isolation, Phase A Isolation, Manual Isolation
Function 3.a (2) Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays
Function 3. b (1) Containment Isolation, Phase B Isolation, Manual Initiation
Function 3. b (2) Containment Isolation, Phase B Isolation, Automatic Actuation Logic and Actuation Relays

Description: Containment Isolation (CI) provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate CI signals, Phase A and Phase B. The Phase A signal isolates all automatically isolatable process lines, except component cooling water (CCW), at a relatively low containment pressure. The Phase A CI is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of CCW, are isolated.

Phase B signal isolates CCW. Manual Phase B CI is accomplished by the same switches that actuate CS. When the two switches in either set are actuated simultaneously, Phase B CI and CS will be actuated in both trains.

The LCO for 3.a (1) requires that two channels be operable. The LCO for 3.a (2) requires that two trains be operable. The LCO for 3.b (1) requires that two channels per train and two trains be operable. And, the LCO for 3.b (2) requires that two trains shall be operable.

Proposed Required Actions: For Functions 3.a (1), 3.b (1), the end state for Required Action B.2.2 is revised to be in Mode 4 in 60 hours instead of in Mode 5 in 84 hours. For Functions 3.a (2) and 3.b (2), the end state for Required Action C.2.2 is revised to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours. A Note is added to Required Actions B.2.2 and C.2.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one channel is inoperable, the other channel is available to the operator to initiate CI. Two trains of automatic actuation logic are also available to actuate the CI equipment. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, CI valves, CS system, and CCW systems are available in Mode 4. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Actions B.2.2 and C.2.2 so that the plant would be allowed to remain in Mode 4.

Function 7.a. Automatic Switchover to Containment Sump, Automatic Actuation Logic and Actuation Relays.

Description: At the end of the injection phase of a LOCA, since the Refueling Water Storage Tank (RWST) is nearly empty, and continued cooling must be provided by the ECCS to remove decay heat, the source of water for the ECCS pumps is automatically switched to the containment recirculation sump. This switchover must occur before the RWST empties to prevent damage to RHR pumps and a loss of cooling capability. Switchover must not occur before there is sufficient water in the containment sump to support Engineered Safety Feature (ESF) pump suction. Also, early switchover must not occur to ensure that sufficient borated water is injected from the RWST.

There are two trains for automatic actuation and the logic and actuation relays consist of the same features and operate in the same manner as described in Function 1.b. The LCO for this function requires that two trains be operable.

Proposed Required Actions: For Function 7.a, the end state for Required Action C.2.2 is revised to be in Mode 4 in 36 hours instead of in Mode 5 in 60 hours, if the inoperable train is not restored to operable status within 24 hours. A Note is added to Required Action C.2.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one train is inoperable, the other train is available to initiate switchover to the containment sump. In addition, the operator can perform the switchover manually. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation.

Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2.2 so that the plant would be allowed to remain in Mode 4.

Function 7.b and 7.c Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Low Coincident With Safety Injection, and RWST Level - Low Low Coincident With Safety Injection and Coincident With Containment Sump Level – High

Description: During the injection phase of a LOCA, automatic switchover from RWST to the containment sump occurs only if the RWST low low level signal is coincident with the SI. This prevents accidental switchover during normal operation.

In some units, additional protection from spurious switchover is provided by requiring a Containment Sump Level - High signal as well as RWST Level - Low Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level - High signal must be present, in addition to the SI signal and the RWST Level - Low Low signal, to transfer the suction of the RIHR pumps to the containment sump.

The RWST has four level transmitters. Units with containment sump level circuitry also have four channels for the sump level instrumentation. The logic requires two out of four channels to initiate the switchover from the RWST to the containment sump. The LCO for this function requires that four channels be operable.

Proposed Required Actions: The end state for Required Action K.2.2 is revised to be in Mode 4 in 18 hours instead of Mode 5 in 42 hours if the inoperable channel is not restored to operable status within 6 hours. A Note is added to Required Action K.2.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one channel is inoperable, the other three channels are available to initiate switchover to the containment sump. The fact that this protection feature is not fully operational would make the operator to be prepared to address a unit transient requiring SI and recirculation knowing that manual initiation of the switchover from RWST to the containment sump may be required. Placement of the unit in Mode 5 does not increase the instrumentation available for event mitigation.

The redundancy is such that a single channel failure and one channel being inoperable will not defeat the initiation of switchover from the RWST to the containment sump. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action K.2.2 so that the plant would be allowed to remain in Mode 4.

3.1.2 TS 3.3.7 - Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

Description: The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Auxiliary Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREFS initiates filtered ventilation and pressurization of the control room.

The actuation instrumentation consists of redundant radiation monitors in the air intakes and control room area. A high radiation signal from any of these detectors will initiate both trains of the CREFS. The control room operator can also initiate the CREFS trains by manual switches in the control room. The CREFS is also actuated by a safety injection (SI) signal. The LCO for this system requires that two channels be operable.

Proposed Required Actions: The end state for Required Action C.2 is to be in Mode 4 in 12 hours instead of in Mode 5 in 36 hours if the Required Action and associated CT for Condition A or B are not met in Mode 1, 2, 3, or 4. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The system design provides redundancy and defense in depth from the multiple channels, trains, and functions available to actuate the CREFS. If one or two channels or trains in one or more functions are inoperable, the Required Actions require one or both CREFS trains to be placed in the emergency radiation protection mode of operation. This places the unit in a conservative mode of operation. In the unlikely event that this is not accomplished and Condition C is entered, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. The system design maintains sufficient defense-in-depth when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2 so that the plant would be allowed to remain in Mode 4.

3.1.3 TS 3.3.8 – Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation

Description: The FBACS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident [involving handling recently irradiated fuel] or a LOCA are filtered and adsorbed prior to exhausting to the environment. The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal (gaseous or particulate) or following an SI signal. Initiation may also be performed manually as needed from the main control room.

Each FBACS train is initiated by high radiation detected by a dedicated channel. Each of the two available channels contains a gaseous and a particulate monitor. High radiation detected by any monitor or an SI signal from the ESFAS initiates fuel building isolation and starts the FBACS. The LCO requirements ensure that instrumentation necessary to initiate the FBACS is operable. The LCO requires that the two trains and [two] channels shall be operable.

Proposed Required Actions: The end state for Required Action D.2 is to be in Mode 4 in 12 hours instead of in Mode 5 in 36 hours if the Required Action and associated CT for Condition A or B are not met in Mode 1, 2, 3, or 4. A Note is added to Required Action D.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one or two channels or trains in one or more functions are inoperable, one or both FBACS trains are to be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. In the unlikely event that this is not accomplished and Condition C is entered, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. The system design maintains sufficient defense-in-depth when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action D.2. so that the plant would be allowed to remain in Mode 4.

3.1.4 TS 3.4.13 – Reactor Coolant System (RCS) Operational Leakage

Description: The safety significance of RCS Leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded.

RCS operational leakage shall be limited to:

a. No pressure boundary leakage,

b. 1 gpm unidentified leakage,

c. 10 gpm identified leakage, and

d. 1 gpm total primary to secondary leakage through all SGs, and

e. [500] gpd primary to secondary leakage through any one SG.

Proposed Required Actions: The end state for Required Action B.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Condition A are not met, or pressure boundary leakage exists or primary to secondary leakage is not within limit. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: RCS leakage that is not large enough to be a small break LOCA should be treated as an event leading to a controlled shutdown which is not modeled in the quantitative risk analysis.

In Mode 4, the RCS pressure is significantly reduced and this reduces the leakage. All LOCA mitigating systems with the exception of the accumulators are available and the RHR serves as the backup to auxiliary feedwater for decay heat removal. If RCS operational leakage is not within limits for reasons other than pressure boundary leakage or primary to secondary leakage the leakage must be reduced to within the limit in 4 hours consistent with Required Action A.1. If operational leakage is not restored to within the limit in 4 hours, in accordance with Required Action A.1, or pressure boundary leakage exists, or primary to secondary leakage is not within the limit, Required Actions B.1 and B.2 become applicable. Required Actions B.1 and B.2 require that the unit be placed in Mode 3 within 6 hours and Mode 4 within 12 hours. Thus, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequence. The staff accepts the proposed change to revise Required Action B.2 so that the plant would be allowed to remain in Mode 4, subject to LCO 3.0.4.a being not applicable for entry into Mode 4.

3.1.5 TS 3.4.14 – RCS Pressure Isolation Valve (PIV) Leakage

Description: 10 CFR 50.2, "Definitions," and 10 CFR 50.55a(c), "Codes and Standards" define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB), which separate the high pressure RCS from an attached low pressure system. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance. A known component of the identified leakage before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational leakage if the other is leaktight.

The main purpose of this specification is to prevent overpressure failure of the low pressure portions of the connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. The failure consequences could be a LOCA outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection.

Proposed Required Actions: The end state for Required Action B.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Condition A are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: This TS limits RCS leakage because of the concern of over-pressurization of a lower pressure system that can lead to an interfacing system LOCA. In Mode 4, the RCS pressure is significantly reduced which reduces the PIV leakage. All LOCA mitigating systems with the exception of the accumulators are available and RHR serves as the backup to auxiliary feedwater for decay heat removal. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action B.2 so that the plant would be allowed to remain in Mode 4.

3.1.6 TS 3.4.15 – RCS Leakage Detection Instrumentation

Description: GDC 30 of Appendix A to 10 CFR 50, "Quality of reactor coolant pressure boundary," requires means for detecting and, to the extent practical, identifying the location of the source of RCS leakage. Leakage detection systems must have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified RCS leakage.

The LCO requires that the following RCS leakage detection instrumentation be operable:

a. One containment sump (level or discharge flow) monitor,

b. One containment atmosphere radioactivity monitor (gaseous or particulate), and

c. One containment air cooler condensate flow rate monitor.

Proposed Required Actions: The end state for Required Action E.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action E.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one function is inoperable, the other functions are available to provide an indication of RCS leakage. In the unlikely event that Condition E occurs, the likelihood of an initiating event is not increased, and placing the unit in Mode 5 does not increase the instrumentation available for detecting RCS leakage. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action E.2 so that the plant would be allowed to remain in Mode 4.

3.1.7 TS 3.5.3 - ECCS - Shutdown

Description: This TS is only applicable in Mode 4. In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and RHR (low head). The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the RWST can be injected into the RCS following an accident. The LCO requires that one ECCS train be operable.

Proposed Required Actions: Condition A is revised from "Required ECCS residual heat removal (RHR) subsystem inoperable" to "Required ECCS train inoperable." Required Action A.1 is revised from "Initiate action to restore required ECCS RHR subsystem to OPERABLE status" to "Initiate action to restore required ECCS train to OPERABLE status." This change allows the unit to remain in Mode 4, rather than transitioning to Mode 5 with an inoperable ECCS high head subsystem.

Assessment: The subsystems addressed by this TS are the ECCS RHR and ECCS High Head subsystems which are both included in the quantitative risk evaluation (Reference 2). The requested change in Action A.1 will enable the unit to remain in a Mode where steam generator cooling is also available for decay heat removal.

Table 3.2.1 of this SE, shows that POS 4 CDP is approximately seven times greater than the POS 3 CDP. Proceeding to Mode 5 does not significantly increase the protection available and additional risk is introduced by switching from AFW cooling to RHR cooling. This supports remaining in Mode 4 for this configuration rather than cooling down to Mode 5.

The proposed change to the Required Action A.1 end state does not change the operability requirement for the ECCS. One train still must be operable in Mode 4. If one train of RHR is inoperable, then remaining in Mode 4 provides core cooling from the AFW pumps with the operable RHR pump as a backup. If both trains of RHR are inoperable, then the unit will remain on AFW cooling while one train is restored. The probability of transients occurring that require the ECCS are less likely in Mode 4 than at-power and the risk associated with transferring to RHR cooling from AFW cooling is eliminated by remaining in Mode 4. Sufficient defense-in-depth is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5.

The staff accepts the proposed change to revise TS 3.5.3 so that the plant would be allowed to remain in Mode 4.

3.1.8 TS 3.5.4 – Refueling Water Storage Tank (RWST)

Description: The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions. The RWST supplies both trains of the ECCS and the Containment Spray System through separate, redundant supply headers during the injection phase of a LOCA recovery.

During normal operation in MODES 1, 2, and 3, the SI and RHR pumps are aligned to take suction from the RWST. This LCO requires that the RWST be operable.

Proposed Required Actions: The end state for Required Action C.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: Since SI and recirculation may not be available due to an inoperable RWST, any loss of inventory events that cannot be isolated can lead to core damage. From Table 3.2.1 of this SE, remaining in Mode 4 (POS 3) instead of cooldown to Mode 5 (POS 4, upper portion of Mode 5) reduces the CDP by more than a factor of 3. The primary accidents such as LOCAs and SLBs are less likely to occur in Mode 4. Since control rods are inserted in Mode 4, the SLB analysis assumption of the highest worth rod stuck is an unlikely scenario. In the lower part of Mode 4 transients progress slower than at power, backup cooling is available via RHS and there is increased time for operator action and mitigation strategies. Proceeding to Mode 5 may add additional risk by switching from AFW cooling to RHR cooling. Based on Table 3.2.1 of this SE, if RWST is inoperable, a shutdown to Mode 4 is appropriate.

In Mode 4, the transient conditions are less severe than at power so that variations in the RWST parameters or other reasons of inoperability are less significant. In addition, if the boron concentration is low, the emergency boration equipment is likely to be available to increase the RCS boron concentration. By changing the end state for Required Action C.2 to Mode 4, the possibility of a loss of inventory event due to switching to RHR cooling is eliminated, reducing the possibility that the RWST inventory would be required. Therefore, sufficient defense-in-depth is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5 with LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise TS 3.5.4 so that the plant would be allowed to remain in Mode 4.

3.1.9 TS 3.6.6A, Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

Description: The CS and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits. The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the CS system during the injection phase of operation. In the recirculation mode of operation, CS pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of service water (SW). Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. Containment Cooling systems are not credited with iodine removal. The LCO requires that two containment spray trains and [two] containment cooling trains shall be operable.

Proposed Required Actions: To revise the end state for Required Action B.2 to be in Mode 4 in 54 hours, and revise the end state for Required Action E.2 to be in Mode 4 in 12 hours. A Note is added to Required Actions B.2 and E.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The CS and containment cooling systems are designed for accident conditions initiated at full power. One train of each system satisfies the assumptions in the safety analyses. One train of CS is required to satisfy assumptions regarding iodine removal. If one train of either CS or containment cooling is inoperable the other train is available to mitigate the accident along with both trains of the other system. If both trains of containment cooling are inoperable, CS can serve as the cooling system and it also serves to remove iodine. Condition F requires that if two CS trains are inoperable or any combination of three or more trains is inoperable the plant must immediately enter LCO 3.0.3. The proposed changes to the TS Bases were also reviewed as input to the TS changes. The requirements of GDC 38 will still be met. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.10 Technical Specification 3.6.6B - Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit not taken for iodine removal by the Containment Spray System)

Description: The CS and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. One train of the CS will cause a reduction of containment pressure, in the event of a design basis accident, to within limits.

The CS system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a CS pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the CS system during the injection phase of operation. In the recirculation mode of operation, CS pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of SW. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. The LCO requires that two CS trains and [two] containment cooling trains shall be operable.

Proposed Required Actions: Revise the end state for Required Action F.2 to be in Mode 4 in 12 hours if the Required Action and associated CT of Condition A, B, C, D, or E not met. A Note is added to Required Action F.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The CS and containment cooling systems are designed for accident conditions initiated at full power. One train of each system satisfies the assumptions for containment cooling in the safety analyses. If one train of either CS or containment cooling is inoperable the other train is available to mitigate the accident along with both trains of the other system. If both trains of containment cooling are inoperable, CS can serve as the cooling system. Condition G requires that if any combination of three or more trains are inoperable the plant must immediately enter LCO 3.0.3. The requirements of GDC 38 will still be met. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.11 TS 3.6.6C, Containment Spray System, (Ice Condenser)

Description: The CS system provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a design basis accident.

Each train includes a CS pump, one CS heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate engineered safety feature (ESF) bus. The RWST supplies borated water to the CS system during the injection phase of operation. In the recirculation mode of operation, CS pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of RHR to additional redundant spray headers completes the CS system heat removal capability. Each RHR train is capable of supplying spray coverage, if required, to supplement the CS system. The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the ECCS is operating in the recirculation mode. The LCO requires that two CS trains shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 54 hours if the Required Action and associated CT not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The CS system is designed for accident conditions initiated at full power. One train satisfies the assumptions in the safety analyses. One train of CS is required to satisfy assumptions regarding iodine removal. If one train of CS is inoperable the other train is available to mitigate the accident. The Ice Condenser is required to be operable and it is designed to handle a heat load in excess of the initial blowdown of a design basis LOCA, or any feedwater or steam line break event inside containment. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The proposed changes to the TS Bases were also reviewed as input to the TS change. The requirements of GDC 38 will still be met. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.12 TS 3.6.6D, Quench Spray (QS) System (Subatmospheric)

Description: The QS system is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS system, operating in conjunction with the recirculation spray (RS) system, is designed to cool and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a design basis accident. Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a design basis accident.

The QS system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the QS system. The QS system is actuated either automatically by a containment High-High pressure signal or manually. Each train of the QS system provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. The LCO requires that two Quench Spray trains shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The QS system is designed for accident conditions initiated at power. One train satisfies the assumptions in the safety analyses. In addition, the containment temperature and pressure limits are set to account for the effects of an energy release during an event at full power operation. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to less severe thermal-hydraulic conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The requirements of GDC 38 will still be met. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.13 TS 3.6.6E, Recirculation Spray (RS) System (Subatmospheric)

Description: The RS system, operating in conjunction with the QS system, is designed to limit the post accident pressure and temperature in the containment to less than the design values and to depressurize the containment structure to a subatmospheric pressure in less than 60 minutes following a design basis accident. The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission product radioactivity from containment to the environment in the event of a design basis accident.

The RS system consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one 50% capacity spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls. The two outside RS subsystems' spray pumps are located outside containment and the two inside RS subsystems' spray pumps are located inside containment. Each RS train (one inside and one outside RS subsystem) is powered from a separate ESF bus. Each train of the RS system provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal. The LCO requires that four RS subsystems [and a casing cooling tank] shall be operable.

Proposed Required Actions: Revise the end state for Required Action F.2 to be in Mode 4 in 54 hours if the Required Action and associated CT not met. A Note is added to Required Action F.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The RS system is designed for accident conditions initiated at power. One train (two subsystems) satisfies the assumptions in the safety analyses. In addition, the containment temperature and pressure limits are set to account for the effects of an energy release during an event at full power operation. One train of RS is required to satisfy assumptions regarding iodine removal. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The proposed changes to the TS Bases were also reviewed as input to the TS change. The requirements of GDC 38 will still be met. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.14 TS 3.6.7, Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

Description: The spray additive system is a subsystem of the CS system that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a design basis accident.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a design basis accident. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms.

For an eductor feed system, the spray additive system consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a CS pump and consists of an eductor for each CS pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line.

For a gravity feed system, the spray additive system consists of one spray additive tank, two parallel redundant motor operated valves in the line between the spray additive tank and the RWST, instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by a balanced gravity feed from the additive tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the containment spray pump suction.

In MODES 1, 2, 3, and 4, a design basis accident (DBA) could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment. The LCO requires that the spray additive system shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 54 hours if the Required Action and associated CT not met. A Note is added to Required Action F.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The TR justifies the proposed change by indicating "[e]vents, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4." The TR also indicates that "proceeding to Mode 5 does not increase the protection available".

CS will still be available to reduce the iodine fission product inventory in the containment. Based on the lower reactor coolant system pressures and temperatures, the ability to maintain the ECCS operation so the criteria of 10 CFR 50.46 are met, the CS systems and containment cooling systems available to depressurize and reduce the airborne radioiodine in containment, the NRC staff finds the proposed change acceptable, with LCO 3.0.4.a not applicable for entry into Mode 4.

3.1.15, Item 6.4.22a Recirculation Fluid pH Control System

Description: Some Westinghouse NSSS plants have replaced the spray additive system with a passive ECCS recirculation fluid pH control system. Although the TS for this system is not contained in NUREG- 1431, the end state is Mode 5 if the system is inoperable, and the Required Action and associated CT are not met. The system consists of baskets in the containment sump with a specified amount of trisodium phosphate in each basket.

The trisodium phosphate dissolves when the containment sump level increases to the level of the baskets. It is highly unlikely that all of the baskets would be empty; therefore, an inoperable recirculation fluid pH control system would still provide some pH control. The justification for changing the end state to Mode 4 for TS 3.6.7, "Spray Additive System," is also applicable to the recirculation fluid pH control system, since they perform the same function.

The recirculation fluid pH control system TS currently requires the unit to be in Mode 3 in 6 hours and Mode 5 in 84 hours if the system is inoperable, and the Required Action and associated CT are not met.

Proposed Required Actions: The Mode 5 end state is proposed to be changed to require the unit to be in Mode 4 in 54 hours if the Required Action and associated CT are not met.

Assessment: The TR justifies the proposed change by indicating "[e]vents, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4." The TR also indicates that "proceeding to Mode 5 does not increase the protection available." The TR also argues that it is highly unlikely that all of the baskets would be empty; therefore, an inoperable recirculation fluid pH control system would still provide some pH control.

CS will still be available to reduce the iodine fission product inventory in the containment. Based on the lower reactor coolant system pressures and temperatures, the ability to maintain the ECCS operation so the criteria of 10 CFR 50.46 is met, the CS systems and containment cooling systems available to depressurize and reduce the airborne radioiodine in containment, the NRC staff finds the proposed change acceptable.

3.1.16 TS 3.6.11 - Iodine Cleanup System (ICS) (Atmospheric and Subatmospheric)

Description: The ICS functions together with the CS and cooling systems following a design basis accident to reduce the potential release of radioactive material, principally iodine, from the containment to the environment.

The ICS consists of two 100% capacity, separate, independent, and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, a demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of radioiodine, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. Each ICS train is powered from a separate ESF bus and is provided with a separate power panel and control panel. During normal operation, the containment cooling system is aligned to bypass the ICS HEPA filters and charcoal adsorbers. For ICS operation following a design basis accident, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers. The LCO requires that two ICS trains shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment:

Function 1: Reduce the concentration of fission products released to the containment atmosphere following a postulated accident.

For Condition A, one ICS train inoperable, seven days are allowed to restore the ICS train to OPERABLE status. The TR requests that if Required Action A.1 cannot be accomplished within the seven day limit, Required Action B.1 be revised to allow the unit to be in Mode 3 in six hours and in Mode 4 in twelve hours. Two trains of containment spray and the second train of ICS will be available for reduction of the concentration of fission products released to the containment following a postulated accident. The requirements of GDC Criterion 41 will be met. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.17 TS 3.6.12, Vacuum Relief Valves (Atmospheric and Ice Condenser)

Description: The purpose of the vacuum relief lines is to protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of containment cooling features, such as the CS system. Multiple equipment failures or human errors are necessary to cause inadvertent actuation of these systems.

The containment pressure vessel contains two 100% vacuum relief lines that protect the containment from excessive external loading. The LCO requires that [two] vacuum relief lines shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment:
Function 1: Protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside containment).

Excessive negative pressure inside containment can occur if there is an inadvertent actuation of containment cooling features, such as the CS system. Excessive negative pressure in a dual containment can cause structural damage to the steel pressure containment. For Condition A, if one vacuum relief line is inoperable, 72 hours are allowed to restore the vacuum relief line to OPERABLE status.

Most containment pressure vessels contain two 100% vacuum relief lines that protect the containment from excessive external loading. This evaluation is applicable only for containment designs with two or more vacuum relief lines. With one vacuum relief line inoperable there will still be one vacuum relief line operable to provide protection for the containment pressure vessel. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change to be acceptable.

3.1.18 TS 3.6.13, Shield Building Air Cleanup System (SBACS) (Dual and Ice Condenser)

Description: The containment has a secondary containment called the shield building, which is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a LOCA. This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The SBACS establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment.

The SBACS consists of two separate and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, moisture separators, a HEPA filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. During normal operation, the shield building cooling system is aligned to bypass the SBACS's HEPA filters and charcoal adsorbers. For SBACS operation following a design basis accident, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers. The LCO requires that two SBACS trains shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment:
Function 1: Collect any containment leakage that leaks into the annular space (annulus) between the shield building and the steel containment vessel following a LOCA.
Function 2: Filter any containment leakage from the annulus before release to the environment.

For Condition A, one SBACS train is inoperable, seven days are allowed to restore the SBACS train to OPERABLE status. For both Function 1 and Function 2, a second train of SBCAS will be available to collect any containment leakage from the annulus and to filter that containment leakage before release to the environment. The energy released to containment from a LOCA will be lower than for the limiting design basis accident. The CS system, containment cooling system (for a dual containment design) and ECCS will be available. The capability of these systems will be well within their design basis should an event occur in Mode 4.

If two trains of SBACS are inoperable, LCO 3.0.3 applies since an associated ACTION for two trains of SBACS inoperable is not provided. On this technical basis, with LCO 3.0.4.a not applicable for entry into Mode 4, the NRC staff finds the proposed change acceptable.

3.1.19 TS 3.6.14, Air Return System (ARS) (Ice Condenser)

Description: The ARS is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a design basis accident. The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting the post accident pressure and temperature in containment to less than design values. The ARS provides post accident hydrogen mixing in selected areas of containment. The ARS also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the containment spray system can cool it.

The ARS consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers with isolation valves. The ARS fans are automatically started and the hydrogen collection header isolation valves are opened by the containment pressure High-High signal 10 minutes after the containment pressure reaches the pressure setpoint. The LCO requires that two ARS trains shall be operable.

Proposed Required Actions: Revise the end state for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment:

Function 1: Recirculation of containment air from the upper containment to lower containment compartment assists in cooling the containment atmosphere and limiting post accident pressure in containment. Containment air cooling for a LOCA or SLB is still required in Mode 4."The CS system will still be available to provide cooling in the upper containment compartment. The energy released into containment from a LOCA or SLB while in Mode 4 is well within the design limits for the containment spray system. CS will be automatically initiated by containment high-high signal or by manual actuation. Adequate cooling will be available to maintain the containment air temperature and containment post accident pressure to within the design limits.

Function 2: Provide mixing in select areas of containment to prevent hydrogen gas accumulation. The ECCS and CS system remain operable. ECCS will be automatically initiated by low pressurizer pressure, high containment pressure, or by manual actuation.

With one train of ARS inoperable the remaining train will be operable to provide air mixing to prevent hydrogen accumulation.

Based on one train of the ARS being operable to help maintain containment cooling and to provide air mixing to prevent hydrogen accumulation along with the CS system to help provide containment cooling in the event of a LOCA or SLB while in MODE 4, with LCO 3.0.4.a not applicable for entry into Mode 4, the change to Required Action B.2 to be in MODE 4 with a CT of 12 hours is acceptable.

3.1.20 TS 3.6.18, Containment Recirculation Drains (Ice Condenser)

Description: The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. [Twenty of the 24] ice condenser bays have a floor drain at the bottom to drain the melted ice into the lower compartment (in the [4] bays that do not have drains, the water drains through the floor drains in the adjacent bays). A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, the check valve opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it drains to the floor and provides a source of borated water at the containment sump for long term use by the ECCS and the CS system during the recirculation mode of operation.

The two refueling canal drains are at low points in the refueling canal. In the event of a design basis accident, the refueling canal drains are the main return path to the lower compartment for containment spray system water sprayed into the upper compartment. The LCO requires that the ice condenser floor drains and the refueling canal drains shall be operable.

Proposed Required Actions: Revise the end state for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated CT are not met. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment:

Function 1: Prevent warm air from entering the ice condenser. For Condition A, one ice condenser floor drain inoperable, one hour is allowed to restore the ice condenser floor drain to OPERABLE status.

Warm air entering the ice condenser can cause sublimation of the ice and/or obstruction of the air passages due to ice buildup. Sublimation of the ice and buildup of ice that could obstruct the air passageways is a slow process. Two trains of containment spray will be available along with the air return system to cool the containment atmosphere to limit the peak containment temperature and pressure in the event of a LOCA or SLB while in Mode 4.

Function 2: Allow water to drain from the ice condenser bays and from the refueling canal to the containment lower compartment to provide a source of borated water at the containment sump for long term use by the ECCS and the CS during the recirculation mode of operation.

For Condition A, one ice condenser floor drain inoperable, one hour is allowed to restore the ice condenser floor drain to OPERABLE status. For Condition B, one refueling canal drain inoperable, one hour is allowed to restore the refueling canal drain to OPERABLE status.

In the event of a LOCA or SLB while in Mode 4, the remaining floor drains will provide sufficient capacity such that water will drain from the ice condenser bays and from the refueling canal to the containment lower compartment to provide a source of borated water at the containment sump for long term use by the ECCS and the CS during the recirculation mode of operation.

Based on the above assessment, with LCO 3.0.4.a not applicable for entry into Mode 4, the ability to provide reasonable assurance that there will be adequate water returned to the containment sump and that there will not be a rapid degradation of the ice condenser due to the in leakage of warm air, the NRC staff finds this proposed change acceptable.

3.1.21 Mode 4 Secondary Side Steam Pressure

TR WCAP-16294-NP, Revision 0, indicated that secondary side steam pressure would be at normal operating pressure. The NRC staff requested that the TR applicant verify the secondary side steam pressure taking into consideration the reduced reactor cooling system average temperature in Mode 4. The TR applicant revised the statement in TR WCAP-16294-NP, Revision 0, to indicate that while in Mode 4, the secondary side steam pressure will be less than normal operating pressure. The TR applicant determined that there will be sufficient pressure available for most plants to operate the turbine driven auxiliary feedwater pumps. This will assure the defense-in-depth will remain available while remaining in Mode 4. The NRC staff finds the revision acceptable.

3.1.22 TS 3.7.7 – Component Cooling Water (CCW) System

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

A typical CCW System is arranged as two independent, full capacity cooling loops, and has isolatable non-safety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of an SI signal, and all nonessential components are isolated.

The principal safety related function of the CCW System is the removal of decay heat from the reactor via the RHR System. This may be during a normal or post accident cooldown and shutdown. The LCO requires that two CCW trains be operable

Proposed Required Actions: The end state for Required Action B.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The CDP values listed in Table 3.2.1 of this SE, from the evaluation for the scenarios, show that there is slightly less risk associated with Mode 4 than there is with a cooldown to Mode 5 when a train of CCW is inoperable.

One CCW train will be operating when the unit enters Mode 4. Each train is designed to handle 100 percent of the heat loads during power operation and accident conditions. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur. Therefore sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action B.2 so that the plant would be allowed to remain in Mode 4.

3.1.23 TS 3.7.8 – Service Water System (SWS)

Description: A typical SWS consists of two separate, 100 percent capacity, safety related, cooling water trains. Each train consists of two 100 percent capacity pumps, one CCW heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA. The pumps aligned to the critical loops are automatically started upon receipt of an SI signal, and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW System and typically is the backup water supply to the AFW system.

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a DBA or transient. During normal operation and a normal shutdown, the SWS also provides this function for various safety related and non-safety related components. The principal safety related function of the SWS is the removal of decay heat from the reactor via the CCW System. The safety related function is covered by this LCO, which requires that two SWS trains shall be operable.

Proposed Required Actions: The end state for Required Action B.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT Condition A are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: Table 3.2.1 of this SE shows that cooldown to Mode 4, rather than cooling down to Mode 5, reduces overall risk of the shutdown process when a train of the SWS is inoperable.

One SWS train will be operating when the unit enters Mode 4. Each train is designed to handle 100 percent of the heat loads during power operation and accident conditions. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action B.2. so that the plant would be allowed to remain in Mode 4.

3.1.24 TS 3.7.9 – Ultimate Heat Sink (UHS)

Description: The UHS provides a heat sink for processing and operating heat from safety related components during a transient or accident, as well as during normal operation. This is done by utilizing the SWS and the CCW System.

The UHS has been defined as the complex of water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the Final Safety Analysis Report. If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of complexes are used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required. This LCO requires that the UHS be operable.

Proposed Required Actions: The end state for Required Action C.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Condition A or B are not met, or the UHS is inoperable [for reasons other than Condition A or B]. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: TS 3.7.9 addresses degradations to the cooling capability of the ultimate heat sink. The most likely scenario for entering Condition C is that the cooling capability of the UHS is only partially degraded. A cooldown to Mode 4 places the unit in a state where heat loads are significantly less than at full power.

The UHS is designed to remove 100% of the heat loads generated during power operation and accident conditions. The heat load will be significantly less in the shutdown modes. Some accidents are less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2. so that the plant would be allowed to remain in Mode 4.

3.1.25 TS 3.7.10 – Control Room Emergency Filtration System (CREFS)

Description: The CREFS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas. The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a pre-filter or demister, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provides backup in case of failure of the main HEPA filter bank.

The CREFS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the system filter trains. The pre-filters or demisters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. This LCO requires that two CREFS trains be operable.

Proposed Required Actions: The end state for Required Action C.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Condition A or B are not met in Mode 1, 2, 3, or 4. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one CREFS train is inoperable, the other train remains available to provide control room filtration. If two CREFS trains are inoperable an independent initiating event and radioactive release must occur for filtration to be required in Modes 4 and 5. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2. so that the plant would be allowed to remain in Mode 4.

3.1.26 TS 3.7.11 – Control Room Emergency Air Temperature Control System (CREATCS)

Description: The CREATCS is an emergency system, parts of which may also operate during normal unit operations. The CREATCS consists of two independent and redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control following isolation of the control room. The LCO requires that two CREATCS trains be operable.

Proposed Required Actions: The end state for Required Action B.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Condition A are not met in Mode 1, 2, 3, and 4. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one CREATCS train is inoperable, the other train remains available to provide control room temperature control. The slower nature of accident event progression in the shutdown modes, and increased time for operator actions and mitigation strategies, limit the severity of accidents in the shutdown modes. The inoperability of equipment does not affect the likelihood of an event occurring and some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action B.2. so that the plant would be allowed to remain in Mode 4.

3.1.27 TS 3.7.12 – ECCS Pump Room Exhaust Air Cleanup System (PREACS)

Description: The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a LOCA. The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower reaches of the Auxiliary Building.

The ECCS PREACS consists of two independent and redundant trains. Each train consists of a heater, a pre-filter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of an SI signal.

The ECCS PREACS is a standby system, aligned to bypass the system HEPA filters and charcoal adsorbers. During emergency operations, the ECCS PREACS dampers are realigned, and fans are started to begin filtration. Upon receipt of the actuating Engineered Safety Feature Actuation System signal(s), normal air discharges from the ECCS pump room isolate, and the stream of ventilation air discharges through the system filter trains. The pre-filters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Proposed Required Actions: The end state for Required Action C.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one ECCS PREACS train is inoperable, the other train remains available to provide pump room air filtration. If two trains are inoperable due to an inoperable ECCS pump room boundary, a LOCA must also occur to require the operation of ECCS PREACS. The severity of the postulated accidents during shutdown modes is limited due to the slower pace of the accident even progression and increased time for operator actions and mitigation strategies. In addition, a LOCA is less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2. so that the plant would be allowed to remain in Mode 4.

3.1.28 TS 3.7.13 – Fuel Building Air Cleanup System (FBACS)

Description: The FBACS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or a LOCA. The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters, functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal.

The FBACS is a standby system, parts of which may also be operated during normal plant operations. Upon receipt of the actuating signal, normal air discharges from the building, the fuel handling building is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters or demisters remove any large particles in the air and any entrained water droplets present to prevent excessive loading of the HEPA filters and charcoal adsorbers. The LCO requires that two FBACS trains be operable.

Proposed Required Actions: The end state for Required Action C.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Condition A or B are not met in Mode 1, 2, 3, or 4 or if two FBACS trains are inoperable in Mode 1, 2, 3, or 4 for reasons other than Condition B. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one FBACS train is inoperable, the other train remains available to provide fuel building air filtration. If two FBACS trains are inoperable, a LOCA or fuel handling accident must also occur to require operation of the FBACs. LOCAs are less likely in Mode 4 and Required

Action E.1 reduces the probability of a fuel handling accident. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2. so that the plant would be allowed to remain in Mode 4.

3.1.29 TS 3.7.14 – Penetration Room Exhaust Air Cleanup System (PREACS)

Description: The PREACS filters air from the penetration area between containment and the Auxiliary Building. The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the air stream, also form part of the system. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of an SI signal.

The PREACS is a standby system, parts of which may also operate during normal unit operations. During emergency operations, the PREACS dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating signal(s), normal air discharges from the penetration room, the penetration room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Proposed Required Actions: The end state for Required Action C.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action C.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: If one PREACS train is declared inoperable, the other train remains available to provide penetration room air filtration. If two PREACS trains are inoperable due to an inoperable penetration room boundary, a LOCA and passive failure in the penetration room must occur to require air filtration. A LOCA is less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action C.2. so that the plant would be allowed to remain in Mode 4.

3.1.30 TS 3.8.1 – [Alternating Current] AC Sources - Operating

Descriptions: The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B diesel generators (DGs)). The AC electrical power system provides independent and redundant source of power to the ESF system.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es). Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System.

After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, [a sequencer/an undervoltage signal] strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

The LCO requires (1) two qualified circuits between the offsite transmission network and the onsite class 1E AC Electrical Power Distribution System, (2) two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and (3) automatic load sequencers for Train A and Train B.

Proposed Required Actions: The end state for Required Action G.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT of Conditions A, B, C, D, E or [F] are not met. A Note is added to Required Action G.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment:

For each of the conditions of this TS LCO, Table 3.2.1 of this SE shows that the CDP decreases slightly when the unit is cooled down to Mode 4 instead of Mode 5.

Two trains of DGs are available if two offsite power circuits are inoperable and two offsite power circuits are available if two diesel generators are inoperable. If an offsite power circuit and/or a diesel generator are inoperable, at least one of each remains available. The slower nature of event progression during shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action G.2. so that the plant would be allowed to remain in Mode 4.

3.1.31 TS 3.8.4 – [Direct Current] DC Sources - Operating

Description: The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). As required by 10 CFR 50, Appendix A, GDC 17, the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure. The DC electrical power system also conforms to the recommendations of Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)."

The typical 125/250 Volts DC (VDC) electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A and Train B). Each subsystem consists of two 125 VDC batteries (each battery 50 percent capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling.

The typical 250 VDC source is obtained by the use of the two 125 VDC batteries connected in series. Additionally there is one spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service. If the spare battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, the 125/250 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

This LCO requires that the Train A and Train B DC electrical power subsystems be operable.

Proposed Required Actions: The end state for Required Action D.2 is revised to be in Mode 4 in 12 hours instead Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action D.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: A quantitative risk evaluation was performed for each condition of this TS by modeling the inoperable equipment. The results are summarized below. The evaluation shows that the CDP decreases slightly when the unit is cooled down to Mode 4 instead of Mode 5.

There are two redundant trains of DC power; so if one is inoperable, the other is available to provide the necessary DC power. Events progress slower during shutdown modes than in the power modes. This provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action D.2. so that the plant would be allowed to remain in Mode 4.

3.1.32 TS 3.8.7 – Inverters - Operating

Description: The function of the inverter is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the Reactor Protection System (RPS) and the ESFAS.

The four (two per train) inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the RPS and ESFAS instrumentation and controls so that the fuel, RCS, and containment design limits are not exceeded. The LCO requires that the Train A and Train B inverters be operable.

Proposed Required Actions: The end state for Required Action B.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action B.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: Results from quantitative risk analysis in Table 3.2.1 of this SE show that the CDP decreases slightly when the unit is cooled down to Mode 4 instead of Mode 5.

There are two redundant trains of inverters; so if one is inoperable, the other train is available to provide the necessary AC power. The slower nature of event progression during shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action B.2. so that the plant would be allowed to remain in Mode 4.

3.1.33 TS 3.8.9 – Distribution Systems - Operating

Description: The onsite Class 1E AC, DC, and AC vital bus electrical power distribution systems are divided by trains into two redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems.

The AC electrical power subsystem for each train consists of a primary Engineered Safety Feature (ESF) 4.16 kV bus and secondary 480 and 120 V buses, distribution panels, motor control centers and load centers. Each 4.16 kV ESF bus has at least one separate and independent offsite source of power as well as a dedicated onsite DG source. Each 4.16 kV ESF bus is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, a transfer to the alternate offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

The secondary AC electrical power distribution subsystem for each train includes the safety related buses, load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1. The 120 VAC vital buses are arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the vital buses are Class 1E constant voltage source transformers powered from the same train as the associated inverter. The DC electrical power distribution subsystem consists of 125 V bus(es) and distribution panel(s).

This LCO requires that Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems to be operable.

Proposed Required Actions: The end state for Required Action D.2 is revised to be in Mode 4 in 12 hours instead of Mode 5 in 36 hours if the Required Action and associated CT are not met. A Note is added to Required Action D.2 stating that LCO 3.0.4.a is not applicable when entering MODE 4.

Assessment: The electrical distribution systems are modeled in the risk evaluation by modeling inoperable equipment. Table 3.2.1 of this SE, shows that the CDP decreases slightly when the unit is cooled down to Mode 4 instead of Mode 5.

The slower nature of event progression during shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur during shutdown modes. Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4 provided LCO 3.0.4.a is not applicable for entry into Mode 4.

The staff accepts the proposed change to revise Required Action D.2 so that the plant would be allowed to remain in Mode 4.

3.2 Risk Evaluation

As stated in Section 2.0 above, the NRC staff reviewed TR WCAP-16294-NP, Revision 0, using SRP Chapters 19.2 and 16.1, and the five key principles of risk-informed decision making presented in RG 1.174 and RG 1.177. SRP 19.2, consistent with RG 1.177, identifies five key safety principles to be met for risk-informed applications, including changes to TS. Each of these principles is addressed by TR WCAP-16294-NP, Revision 0, as discussed below.

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

10 CFR 50.36(c) provides that TSs will include LCOs which are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When an LCO of a nuclear reactor is not met, the licensee will shut down the reactor or follow any remedial action permitted by the TSs until the condition can be met. TR WCAP-16294-NP, Revision 0, proposes only to change the final end state condition applicable when the LCO is not met and the reactor is to be shut down. The LCOs themselves would remain unchanged. Therefore, the proposed changes for the required shutdown end states are consistent with current regulations, and satisfy the first key safety principle of RG 1.177.

2. The proposed change is consistent with the defense-in-depth philosophy.

Consistency with the defense-in-depth philosophy is maintained if:

- A reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design is avoided.
- System redundancy, independence and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).
- Defenses against potential common cause failures are preserved, and the potential for the introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are preserved.
- The intent of the general design criteria in 10 CFR Part 50, Appendix A, are maintained.

The NRC staff assessment of the proposed changes with respect to defense-in-depth was performed in Section 3.1, above.

3. The proposed change maintains sufficient safety margins.

The NRC staff assessment of the proposed changes with respect to maintenance of safety margins was performed in Section 3.1, above.

4. <u>When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement</u>.

TR WCAP-16294-NP, Revision 0, concludes that the proposed change in end state from Mode 5 to Mode 4 results in a reduction in CDF. The NRC staff reviewed the qualitative and quantitative analyses documented in TR WCAP-16294-NP, Revision 0, in order to assess the validity of this conclusion. The NRC staff notes that a reduction in CDF achieved by remaining in hot shutdown versus transitioning to and from cold shutdown is consistent with previously reviewed and approved precedents (References 4, 5, and 6). The NRC staff's review of the risk assessment methods, analyses, and conclusions are presented below.

5. <u>The impact of the proposed change should be monitored using performance measurement strategies.</u>

Plant shutdowns due to not meeting the TS Required Action and associated CTs are infrequent events.. The proposed changes would permit a plant to remain in Mode 4 and complete repairs rather than requiring the plant to incur the risk of transition to and from Mode 5, and conducting repairs in Mode 5, which involves higher risk. The impact of the proposed change is therefore a reduction in plant risk. The change has no impact on the operation or performance of plant components, and performance measurement strategies are therefore considered unnecessary.

3.2.1 Risk Approach

The TR's generic qualitative risk assessment shows that the proposed TS end state changes result in an increase in defense-in-depth for the expected initiating events. This is achieved by performing qualitative risk comparisons between Mode 5, the current end state, and the proposed end state, Mode 4 on SG cooling.

The generic quantitative risk assessment (1) substantiates the conclusion of the qualitative risk assessment by providing numerical results for a representative plant, and (2) investigates the robustness of the results to uncertainties in data and modeling assumptions through sensitivity analyses.

In addition, specific assessments for each proposed TS end state change were also performed, using risk insights from the qualitative and, if applicable, quantitative risk assessments, to ensure that the specific condition causing the LCO does not increase the risk when the proposed new end state is implemented. Finally, risk insights are used in the implementation of the proposed change to identify risk-significant plant configurations and compensatory measures.

The NRC staff finds that the risk assessment approach is comprehensive and follows NRC staff guidance as documented in RGs 1.174 and 1.177.

3.2.2 Assumptions

The risk impact of the mode changes was evaluated based on the following assumptions:

- All equipment is assumed to be available, unless operating procedures direct that equipment be isolated or locked out, or if the equipment is associated with the specific TS LCO being evaluated. Of specific relevance to this analysis is the availability of the turbine-driven AFW pump in Mode 4.

- Entry into the shutdown mode under consideration (Mode 4 or Mode 5) is for the primary intent to repair a non-functional component and return the plant to power as soon as is practical.
- Shutdown and cooldown of the plant terminates when the end state mode is entered, and no further cooldown occurs.
- For the Mode 4 end state, SDC is not aligned, and the plant remains on SG cooling, avoiding transition risks associated with SDC alignment.
- For the Mode 5 end state, the RCS remains at its nominal inventory and the RCS boundary strength is not compromised (e.g., via installation of nozzle dams).

These assumptions are consistent with typical entries into the shutdown modes for short repairs, which are the intended uses of the TS end state change, and are reflected in the Limitations and Conditions in Section 4 of this safety evaluation.

3.2.3 Quality of the PRA Analysis

The quality of both the qualitative and quantitative risk assessments, in terms of scope, level of detail, and technical adequacy, must support the application and the role of the PRA results in the integrated decision process. Per RG 1.174, emphasis on PRA quality can be reduced if the proposed changes result in very small risk increases or in risk decreases. The proposed change to TS end states is evaluated to result in a risk decrease, and so, consistent with RG 1.174, reduced emphasis on PRA quality may be acceptable.

Typically, the quality of the PRA analyses that support plant-specific licensing actions are evaluated based on completion of industry peer reviews of the relevant at-power, plant-specific PRA model, and resolution of the findings from the reviews, along with a focused scope review by the NRC staff of those elements of particular relevance to the application. In this case, the TR generically evaluates the difference in risk between two plant shutdown conditions and, therefore, a plant-specific, at-power PRA model is not relevant to the application.

Although TR WCAP-16294-NP, Revision 0, provides detailed information regarding plant shutdown and mode changes from at-power conditions until cold shutdown (Mode 5), and the heatup and return to power, the only portions relevant to this application cover plant operation at the upper temperature range of Mode 4 and transition to the upper temperature range of Mode 5, and then back to Mode 4. This is because, regardless of the TS end state, a plant must always transition from Mode 1 to Mode 4, and then ultimately return from Mode 4 to Mode 1. Therefore, the NRC staff review did not consider the risk analyses for the common mode transitions (i.e., Mode 1 to upper end of Mode 4 and back to Mode 1), which are applicable to both end states.

The quality of the risk assessments was evaluated by considering 1) the scope and treatment of initiating events based on the plant conditions, 2) the availability of mitigating systems considering TS requirements, operating practices and procedures relevant to the operating mode, and 3) treatment of plant alignment changes required to achieve the particular plant conditions during the plant shutdown and cooldown. In addition for the quantitative risk assessment, the modeling, data sources, and the treatment of human error dependencies, were evaluated, including a review of the specific cutsets generated for each end state.

Initiating events applicable to at-power conditions were considered and modified based on the reduced operating temperatures and pressures which exist in Modes 4 and 5. Specifically, high energy line breaks such as LOCAs, steam generator tube ruptures (SGTRs), and steamline breaks were considered applicable to Mode 4 at a reduced frequency compared to at-power frequencies, and were not considered credible in Mode 5, due to lower temperatures and pressures in the RCS and secondary. Initiating events involving anticipated transient without scram were eliminated for both modes based on the operating restriction requiring the reactor trip breakers to be open. Loss of offsite power events were retained at the same frequency for both modes, based on a review of available data sources applicable to shutdown conditions. Initiating events related to interruption of main feedwater were excluded from both modes, since the system would be out of service once AFW was aligned during the initial plant shutdown. For Mode 5, additional initiating events associated with loss of SDC and inventory loss during alignment of SDC and transition away from and returning to SG cooling were considered, along with events challenging low temperature overpressure protection of the reactor vessel.

The analyses did not address external initiating events, including internal fires. There are no unique mitigation requirements for these events in either Mode 4 or Mode 5, and so the general risk assessment insights and conclusions would be applicable to such initiating events, and the conclusions of the risk analyses would not be affected.

The NRC staff review of the initiating events found the scope of initiators considered in the risk analyses, as well as the treatment of initiators based on differences in plant conditions for the two end states, to be reasonable and comprehensive. The risk analyses considered differences in the availability of mitigating equipment between the two end state modes, based on TS requirements and operating practices. Although the TSs do not require many safety systems to be available in Mode 5, the analyses reasonably assume that, for brief shutdowns required to effect equipment repairs and return the unit promptly to service, those systems would remain available unless required to be isolated or locked out by TS or operating procedures. Since the intent of the analyses is to justify a Mode 4 end state, the availability of safety systems in Mode 5 would conservatively bias the results in favor of a Mode 5 end state, and therefore this represents a conservative assumption. The NRC staff review of the mitigating systems found the assumptions to be reasonable, and the scope of systems evaluated to be appropriate to support this application.

During its review of the equipment availability, the NRC staff questioned the assumption that the turbine-driven AFW pump would be available in Mode 4 when the standard Westinghouse TS only require one motor-driven AFW pump. At the upper temperature range of Mode 4, which is the desired end state, the turbine-driven AFW pump is physically capable of operation. The risk analyses identified the availability of this pump as a key factor in support of the application, and so appropriate controls on its availability must be addressed, as reflected in the Limitations and Conditions in Section 4 of this safety evaluation.

The risk analyses considered the plant alignments necessary to establish SDC during the transition from Mode 4 to Mode 5, and to secure SDC when returning the plant to operation. These alignments are implicitly incorporated as additional initiating events for loss of decay heat removal and loss of inventory applicable to the Mode 5 end state only. No unique alignments were considered for maintaining either Mode 4 or Mode 5 conditions once successfully established. The NRC staff review found that the treatment of transition risk was reasonable and supported the application.

For the quantitative analysis, system models and data were identified as being based on a typical Westinghouse PWR at-power PRA model which had been peer reviewed, and for which the significant review comments had been resolved. The mitigating systems credited in the analysis were reviewed by the NRC staff, and it was concluded that the failure modes and probabilities would not be significantly changed during shutdown operations. Therefore, the models and failure data are of adequate quality and are acceptable for use in this application.

For the shutdown risk assessment applicable to Mode 5 on RHR cooling, mitigation of accidents and transients typically requires significant operator actions, since the plant design does not provide for automatic actuations of required plant components. For the quantitative risk analyses, it is essential to account for potential dependency between multiple human errors in order to properly assess risk. The NRC staff review identified that a dependency evaluation was performed, but that it appeared to be conservative (i.e., resulted in higher human error probabilities than a more realistic evaluation). For purpose of this application, a conservative dependency evaluation non-conservatively biases the risk analysis conclusions in favor of the proposed Mode 4 end state. In response to an NRC question, the dependency evaluation was reviewed and updated, and sensitivity analyses conducted which validated the analysis. Based upon this updated evaluation, the NRC staff finds that the treatment of human error dependencies for the quantitative risk analysis is acceptable to support this application.

3.2.4 Risk Assessment Results

3.2.4.1 <u>Qualitative Risk Assessment</u>

The qualitative risk assessment compares 1) operation in the upper temperature range of Mode 4 while remaining on SG cooling, and 2) alignment of SDC and continued cooldown through Mode 4 to the upper temperature range of Mode 5, maintaining operation in Mode 5 during equipment repairs, and then heating up to the upper temperature range of Mode 4 and returning to SG cooling. This comparison, which assesses the systems and components available to maintain critical safety functions for postulated initiating events, considered the following:

- Differences in the likelihood and consequences of initiating events due to changes in plant temperatures and pressures,
- Differences in the availability of plant equipment, including containment isolation and cooling capability, and
- Transition risks associated with plant equipment alignment changes for initiating and terminating SDC.

The TR identifies, for the two operational modes, key plant parameters (Tables 6-1, 6-3, 6-4, and 6-5), status of systems (Tables 6-2, 6-4, and 6-5), and plant activities (Tables 6-4 and 6-5). The initiating events applicable to each operating mode are presented and evaluated as to their likelihood (Table 6-6).

Since Mode 4 is the preferred end state, assumptions which increase Mode 4 risk or decrease Mode 5 risk are conservative for this application. The key differences between the preferred end state, Mode 4 on SG cooling, compared with the current required transition to Mode 5 on RHR cooling and ultimate return to Mode 4 after equipment repairs, were identified as:

- Reduced RCS and secondary side pressure and temperature in Mode 5.

  In both end states, RCS and secondary side pressure and temperature are significantly reduced compared to nominal at-power conditions, and this reduces the likelihood of LOCAs and high energy line breaks, including steam generator tube ruptures. In Mode 4, the frequency of these events is reduced compared to at-power conditions, and in

Mode 5 the analysis eliminates these events from consideration. This conservatively biases the risk analysis in favor of Mode 5 as the preferred end state.

- Increased likelihood of loss of decay heat removal during transfer from SG cooling to SDC.

  The frequency of loss of decay heat removal is greater with Mode 5 as the end state compared to Mode 4 due to the system re-alignments required. Loss of decay heat removal events in Mode 4 can be mitigated with AFW or the RHR system following depressurization of the RCS. In Mode 5, the turbine-driven auxiliary feedwater (TDAFW) pump would be unavailable due to the lack of steam pressure in the secondary. Thus there are additional options for decay heat removal available in Mode 4 that are not available in Mode 5.

- Increased likelihood of loss of inventory during transfer from SG cooling to SDC.

  The frequencies of loss of inventory events can be greater with Mode 5 as the end state compared to Mode 4 due to the system re-alignments required. Loss of inventory events in Mode 4 can be mitigated with the available train of ECCS. In Mode 5, one full train of ECCS is not required to be available, the SI pumps are out of service and inventory control is dependent upon the charging system. Thus, there are additional options for inventory control available in Mode 4 that are not available in Mode 5.

- Potential for cold overpressurization events in Mode 5.

  Although not identified as a significant source of risk, these events are considered in Mode 5.

- Mitigation of loss of offsite power events.

  Mitigation of loss of offsite power/station blackout events in Mode 4 can be provided by the AFW system including the TDAFW pump. Availability of the TDAFW pump is particularly important in case the event degrades to a station blackout, since the pump is not dependent upon the availability of AC power from the diesel generators. In Mode 5, the turbine-driven AFW pump is unavailable, and the plant will be dependent on restoring electric power to the RHR system. Reactor coolant pump seal LOCAs are not postulated to occur during loss of seal cooling due to station blackout in either mode based on reduced temperatures and pressures in the RCS. Thus there are additional options for mitigation of loss of offsite power events available in Mode 4 that are not available in Mode 5.

- Redundancy and diversity of mitigating and support systems in Mode 4.

  In Mode 4, there are additional TS requirements governing the availability of mitigating systems and support systems, including sources of electric power, compared to Mode 5. Further, there are more potential testing and maintenance activities which take place in Mode 5 compared to Mode 4 which could result in loss of offsite power. Finally, TS requirements associated with containment isolation and cooling are applicable in Mode 4 and not in Mode 5.

Important insights derived from the qualitative assessment are listed below:

- The frequency of loss of decay heat removal is greater with Mode 5 as the end state compared to Mode 4 due to the system re-alignments required. Loss of decay heat removal events in Mode 4 can be mitigated with AFW (all pumps available), or the RHR system following depressurization of the RCS. In Mode 5, the TDAFW pump will not be available to provide mitigation of similar events, and the automatic AFW start signal would not be available. Therefore, additional options are available for decay heat removal in Mode 4 that are not available in Mode 5.

- The frequency of loss of inventory events is greater with a Mode 5 as the end state compared to Mode 4 due to the system realignments required. Loss of inventory events in Mode 4 can be mitigated with the available train of ECCS. In Mode 5, a full train of ECCS is not required to be available, and SI pumps would be out of service, so inventory control is dependent on the charging system. Therefore, additional options are available for inventory control in Mode 4 compared to Mode 5.
- Mitigation of loss of offsite power/station blackout events in Mode 4 can be provided by the AFW system, including the TDAFW pump. In Mode 5, the TDAFW pump will not be available, and the plant will be dependent upon restoring electric power to the RHR system. Therefore, additional options are available for decay heat removal in Mode 4 compared to Mode 5 in the event of a loss of offsite power or station blackout.
- Risk in the shutdown modes, including Mode 5, is very dependent on electric power availability. There are more required independent sources of electric power in Mode 4 compared to Mode 5, and there are more potential activities in Mode 5 that could cause a loss of offsite power.
- In Mode 4, there is more redundancy and diversity of mitigating and support systems required to be available than there is in Mode 5.

The NRC staff finds the qualitative risk assessment adequate to support an assessment of the differences in the plant status which impact configuration risk. The analysis supports a conclusion that the proposed TS end state changes from Mode 5 to Mode 4 do not increase risk or decrease defense-in-depth based on the following:

- Reasonable and conservative assumptions with regards to the likelihood of initiating events occurring at reduced temperature and pressure conditions in both Modes 4 and 5;
- Availability of additional mitigation capabilities in Mode 4 compared to Mode 5 for postulated initiating events;
- Availability of mitigation capability for loss of offsite power in Mode 4 via the TDAFW pump which is not dependent upon on-site AC sources or restoration of AC power; and
- Avoidance of transition risks associated with SDC alignments when remaining in Mode 4.

3.2.4.2 <u>Quantitative Risk Assessment</u>

The NRC staff reviewed the quantitative risk assessment to ensure that:

- Initiating events, accident sequences, and failures found to be significant contributors to shutdown risk in previous studies have been addressed;
- Important assumptions and data used are reasonable;
- Important uncertainties in data and modeling assumptions were identified and sensitivity studies were performed to provide confidence in the conclusions regarding the proposed TS end states; and

- Design and operational differences among Westinghouse plants were identified and appropriate sensitivity studies were performed which show that the conclusions of the quantitative risk assessment apply to all Westinghouse plants.

A quantitative risk assessment was performed using a generic plant PRA model for a Westinghouse PWR to assess the core damage risk of the two end states. As in the qualitative risk assessment, these end states are 1) operation in the upper temperature range of Mode 4 while remaining on SG cooling, and 2) alignment of SDC and continued cooldown through Mode 4 to the upper temperature range of Mode 5, maintaining operation in Mode 5 during equipment repairs, and then heating up to the upper temperature range of Mode 4 and returning to SG cooling.

3.2.4.2.1 Mitigating Systems models

The model is a generic transition risk model representative of Westinghouse plants addressing internal events. It is based on the information identified in the qualitative risk insights, and is representative of a single unit site with standard two train systems. The generic model employs system configurations common to many Westinghouse plants. The key characteristics of the models of mitigating systems were identified as follows:

- AFW System – two motor-driven pumps and one turbine-driven pump. This design is similar to many Westinghouse plants. Some plant designs include a diesel-driven pump which would provide additional diverse mitigation for loss of offsite power events in either Mode 4 or Mode 5. Such designs would enhance the mitigation of loss of offsite power events, which are larger contributors to the risk profile in Mode 4 compared to Mode 5. Therefore, the analyzed configuration conservatively biases the risk results in favor of a Mode 5 end state.
- ECCS – two train system, each train includes a high pressure and low pressure subsystem. The centrifugal charging pumps are used for high pressure safety injection. Low pressure injection and recirculation is performed by the RHR pumps and heat exchangers. This design is similar to many Westinghouse plants.

  Some plant designs include separate intermediate head safety injection pumps. The assumption of combined charging and high pressure safety injection pumps conservatively bounds this application because it assures that at least one SI pump is available in Mode 5, while the separate pumps design has no requirement for availability in Mode 5, and in some cases requires the SI pumps to be disabled due to potential RCS overpressure concerns. Therefore, the assumption of combined charging and high pressure safety injection pumps conservatively biases the results in favor of a Mode 5 end state.

  Some plant designs include separate low pressure safety injection pumps and RHR pumps. The availability of separate low pressure injection and recirculation pumps would reduce risk in Mode 5, which is offset by not crediting RHR cooling as a recovery option in Mode 4. Therefore, the overall conclusions of the report would not be impacted by this assumption.
- RPS – solid state protection system. Some plant designs use relay protection systems; however, the reliability of the two systems was identified as not significantly different, and so the generic model is acceptable as representative of the RPS of all Westinghouse plants.

- CCW System – two trains with one pump per train and a spare pump which can be aligned to either train. This is a common design among Westinghouse plants. The availability of a spare pump is more significant in Mode 5, where the CCW system is required to operate to maintain decay heat removal. Plants with a common CCW header would be expected to have a higher failure probability for the CCW system. This would directly impact decay heat removal in Mode 5, whereas failure of the CCW system in Mode 4 would not directly contribute to a loss of AFW and SG cooling. Therefore, the assumption of separate headers and availability of a spare pump conservatively biases the results in favor of a Mode 5 end state.
- SWS – two trains with one pump per train and a spare pump which can be aligned to either train. This is a common design among Westinghouse plants. The effects of the availability of the spare pump and a common header are similar to these effects on the CCW system previously discussed.
- Electrical Power System – two trains with one emergency diesel generator per train. This is a common design among Westinghouse plants. Some plants have more redundancy in their design, including shared diesels between units. Such designs would enhance the mitigation of loss of offsite power events, which are larger contributors to the risk profile in Mode 4 compared to Mode 5. Therefore, the analyzed configuration conservatively biases the risk results in favor of a Mode 5 end state.

The NRC staff reviewed the assumed mitigating system configurations, and the consideration of alternative designs, and finds them to be reasonable and appropriately justified by the evaluations provided as bounding for all Westinghouse plants.

3.2.4.2.2 Plant Response models

Distinct plant response models are used for Mode 4 and Mode 5, based on the differences in applicable initiating events and the required actions for mitigation of those events in the two modes.

Mode 4

- LOCAs – mitigation requirements are the same as the at-power PRA model, except for the availability of accumulators.
- SGTR – mitigation requirements are the same as the at-power PRA model.
- Secondary Side Breaks – mitigation requirements are the same as the at-power PRA model
- Loss of Decay Heat Removal – mitigation requirements are the same as the at-power PRA model for loss of main feedwater, except that the operating AFW or startup feedwater pump initially in operation is assumed to be failed and unavailable.
- Loss of Offsite Power – mitigation requirements are the same as the at-power PRA model, except that at reduced RCS temperatures, reactor coolant pump seal LOCAs are not assumed to occur.

Mode 5

- Loss of Inventory – this initiator is unique to Mode 5 and is not addressed in the at-power PRA model, as it is postulated to occur due to improper alignment of the RHR system during transition to SDC. The loss of inventory event is mitigated by leak isolation, makeup, and restoration of decay heat removal, and it may include alignment of the recirculation mode of cooling for unisolated leaks.

- Loss of Decay Heat Removal – this initiator is unique to Mode 5 and is postulated to occur due to improper alignment of the RHR system during transition to SDC or due to failures during SDC operation. The loss of decay heat removal event is mitigated either by a motor-driven AFW pump and restoration of SG cooling or by once through cooling (feed-and-bleed) including high pressure recirculation.
- Loss of Offsite Power – this initiator is assumed to interrupt SDC in Mode 5. Mitigation of a loss of offsite power event is similar to a loss of decay heat removal, except that SDC may be restored if power is available either by the emergency diesel generators or by recovery of offsite power.
- Cold Overpressurization – this initiator is mitigated by operator action to control charging and letdown, or if this action is unsuccessful, by relief valve operation. The potential for a failure to close of the relief valve is also considered, and if this occurs, it is assumed that the event is not mitigated. While this conservative treatment biases the results in favor of a Mode 4 end state, this initiator is not a significant contributor to the risk analyses results.
- Boron Dilution – this initiator is mitigated by operator action to terminate the dilution or to initiate boration. While this conservative treatment biases the results in favor of a Mode 4 end state, the contribution of this initiating event is insignificant to the final results, and is, therefore, acceptable.

The NRC staff reviewed the scope and treatment of initiating events in the two end states and considered similar information available in NUREG/CR-6144 Vol. 2 (Reference 18), and finds that the initiating event scope is appropriate and the assumptions regarding mitigation of those events reasonable and conservative for this application.

3.2.4.2.3 Time in End States

In order to employ the models of the two end states including transition between Modes 4 and 5, the estimated time anticipated to be in each mode must be determined. These times are used in the calculation of initiating event probabilities applicable during operation in a particular mode or plant condition. The TR based these times on the time limits of the TS (applicable during the shutdown), and on information from several Westinghouse plants for startup times following a forced non-refueling outage.

The NRC staff reviewed the assumptions and bases for the times anticipated to be in Mode 4 and for a transition to Mode 5 and return to Mode 4, following a forced non-refueling outage and finds them to be reasonable and appropriate for this application.

3.2.4.2.4 Initiating Event Probabilities

The probability of an initiating event is the product of its frequency and the time spent in the particular end state being evaluated. Section 3.2.4.2.3 discussed the basis for the time spent in each end state. The frequencies of some initiators are also varied based on the plant conditions.

LOCAs – The frequency of LOCAs in Mode 4 is assumed to be reduced by a factor of 20 from the typical at-power frequency based on NUREG/CR-6144 (Reference 13). In Mode 5, LOCAs not considered credible due to the reduced temperatures and pressures of the RCS.

Interfacing Systems LOCA – The nominal at-power frequency is maintained for Mode 4, and the events are not considered credible in Mode 5 due to the reduced temperatures and pressures of the RCS. This assumption is conservative for this application, since it increases the risk of Mode 4 compared to Mode 5.

Reactor Vessel Rupture – The nominal at-power frequency is maintained for Mode 4, and the event is not considered credible in Mode 5 due to the reduced temperatures and pressures of the RCS. This assumption is conservative for this application, since it increases the risk of Mode 4 compared to Mode 5.

SGTR – The nominal at-power frequency is maintained for Mode 4, and the event is not considered credible in Mode 5 due to the reduced temperatures and pressures of the RCS. This assumption is conservative for this application, since it increases the risk of Mode 4 compared to Mode 5.

Secondary Side Breaks – The nominal at-power frequency is maintained for Mode 4, and the event is not considered credible in Mode 5 due to the reduced temperatures and pressures of the RCS. This assumption is conservative for this application, since it increases the risk of Mode 4 compared to Mode 5.

Loss of Inventory – This event is only applicable in the transition from Mode 4 to Mode 5, and while in Mode 5. The frequency used is based on a similar treatment found in Reference 14. The total frequency is a combination of operator error resulting in an inadvertent transfer of reactor coolant out of the RCS, events which initiate in systems connected to the RCS, and events resulting from maintenance activities.

Loss of Offsite Power – The nominal at-power frequency is maintained for Modes 4 and 5. This is based on a review of loss of offsite power events in EPRI Technical Report 10029987 (Reference 19) to determine a revised frequency for shutdown conditions. The result was similar to the at-power frequency, and therefore the at-power frequency was retained.

Cold Overpressurization – This event only applies to Mode 5. The frequency is based on WCAP-11737 (Reference 20), which is a higher frequency than Reference 14. Although a higher frequency is not conservative for this application, the contribution of this initiating event is insignificant and, therefore, the frequency used is acceptable.

Boron Dilution – This event is not considered credible in Mode 4, but is considered credible in Mode 5 with a frequency based on Reference 14. Although this is non-conservative for this application, by reducing the risk in Mode 4 compared to Mode 5, the contribution of this initiating event is insignificant and, therefore, the frequency used is acceptable.

3.2.4.2.5 Quantitative Results

The models were quantified to obtain the risk in terms of core damage probability (CDP) for a plant shutdown and restart using a Mode 4 end state compared with a Mode 5 end state, with the following results:

- CDP for Mode 4 end state: 6.02E-6
- CDP for Mode 5 end state: 9.52E-6

The increase in CDP for a Mode 5 end state was attributed to the additional risk incurred during transition from SG cooling to shutdown (RHR) cooling. The quantitative results support a Mode 4 end state over a Mode 5 end state.

The results were further reviewed to obtain qualitative insights of the risk profile for the two end states. Loss of decay heat removal, loss of offsite power, and loss of inventory were the dominant initiating events in both end states. The Mode 5 unique initiating events, cold overpressurization and boron dilution, were not significant contributors and did not influence the results. These insights, and the lower CDP in Mode 4 compared to Mode 5, are consistent with the qualitative risk analysis and insights.

3.2.4.2.6 Containment Considerations

In Mode 4, containment systems, including the structure, isolation valves, and cooling and spray systems, are required to be operable since a design basis accident could cause a release of radioactive material to containment and an increase in containment pressure and temperature that would require the use of these systems for accident mitigation. In Mode 5, the probability and consequences of these events are lesser due to the reduced pressure and temperature of the RCS and secondary systems. However, the risk analyses consider more than the design basis events, and such events occurring in Mode 5 may lead to core damage and cause a challenge to containment.

For the proposed changes to the TSs not involving containment systems, an end state of Mode 4 compared to Mode 5 means that containment and the containment safeguards systems are required to be operable and, therefore, are available to mitigate the consequences of any event. For changes to the TSs for containment safeguards systems, however, a Mode 4 end state maintains the plant in a condition where design basis accidents may occur which would require the availability of the containment boundary. Although the probability and the consequences of these events are significantly reduced in Mode 4, the NRC staff considers that the function of containment as a fission product barrier, which is achieved by compliance with LCOs 3.6.1, 3.6.2, and 3.6.3 for containment integrity and isolation, must be available in Mode 4. Since the scope of the TR, as revised in response to the NRC staff RAI, excludes LCOs 3.6.1, 3.6.2, and 3.6.3, the availability of the containment boundary is assured while a plant is in a Mode 4 end state, and this is acceptable.

3.2.4.2.7 External Events

The quantitative risk analysis does not include consideration of external events, including internal fires and floods. The qualitative risk analysis demonstrated increased redundancy and diversity of available mitigation systems for decay heat removal, inventory control, and electric power. With more options available in Mode 4 compared to Mode 5, any external initiating event would be less likely to result in a loss of all available equipment needed for mitigation. In either mode, the reduced RCS temperatures and pressures eliminate the requirement to maintain seal cooling for the reactor coolant pumps, and loss of seal cooling is one significant impact for internal floods and fires. This reduces the impact of these events in Modes 4 or 5.

With regards to internal flooding, in Mode 4, main feedwater system operation has been terminated and secondary cooling is provided by AFW. Therefore, feedwater system flooding events are not applicable. The flooding events of interest in Modes 4 and 5 involve the CCW and SW systems. The quantitative results show that the risk in Mode 5 is greater when a train of these systems is inoperable than the risk in Mode 4. Thus, the risk impact of an internal flood event involving these systems would exhibit a similar difference in risk and, therefore, inclusion of these events would not change the overall conclusions of the risk analyses.

With regards to internal fire events, the operator response to fires occurring in Mode 4 would be similar to the response to fires occurring in Mode 5. The additional safety equipment available in Mode 4 compared to Mode 5 provides additional assurance of successful mitigation of fires. Therefore, inclusion of fire initiators would not be expected to change the overall conclusions of the risk analyses.

With regards to seismic risk, it is reasonable to assume that the trains of a seismically qualified system will respond to a seismic event in a similar manner regardless of the plant operating mode. Once again, the availability of additional safety-related, seismically qualified systems in Mode 4 compared to Mode 5 provides additional assurance of successful mitigation of seismic events. Further, a seismic event is likely to lead to a loss of offsite power, and there are more independent sources of power required to be available in Mode 4 compared to Mode 5, including the TDAFW pump which can provide decay heat removal without AC power. Therefore, inclusion of seismic events would not change the overall conclusions of the risk analyses.

The NRC staff concludes that the TR WCAP-16294-NP, Revision 0, adequately justifies that external events risk would not impact the overall conclusion of the report favoring a Mode 4 end state over Mode 5.

3.2.4.2.8 TS LCO Specific Quantitative Analyses

The baseline PRA model for transition risk assumes availability of equipment consistent with the assumptions discussed.

For those TS LCOs within the scope of the TR which include equipment modeled in the PRA, additional quantitative analyses were performed assuming unavailability of the equipment as per the TS conditions. The results are shown in the following table.

Table 3.2.1 Core Damage Probability for Various TS LCOs

TS LCO	Condition Modeled	CDP – Mode 4 End State	CDP – Mode 5 End State
3.5.3	ECCS – Shutdown, no available high head SI subsystem	1.32E-5	9.52E-5
3.5.4	Refueling Water Storage Tank, no available tank	3.08E-5	1.13E-4
3.6.6A and 3.6.6B	Containment Spray and Cooling, one train Containment Spray unavailable	6.02E-6	9.52E-6
3.6.6A and 3.6.6B	Containment Spray and Cooling, two trains Containment Spray unavailable	6.02E-6	9.52E-6
3.6.6A and 3.6.6B	Containment Spray and Cooling, one train Containment Cooling unavailable	6.03E-6	9.53E-6
3.6.6A and 3.6.6B	Containment Spray and Cooling, two trains Containment Cooling unavailable	6.36E-6	9.77E-6
3.6.6A and 3.6.6B	Containment Spray and Cooling, one train Containment Spray and one train Containment Cooling unavailable	6.03E-6	9.53E-6
3.7.7	Component Cooling Water, one train unavailable, backup pump unavailable	1.01E-5	2.46E-5
3.7.7	Component Cooling Water, one train unavailable, backup pump available	8.35E-6	2.22E-5
3.7.8	Service Water, one train unavailable, backup pump unavailable	2.41E-5	4.38E-5
3.7.8	Service Water, one train unavailable, backup pump available	8.09E-6	2.71E-5
3.8.1	AC Power – Operating, one offsite circuit unavailable	6.79E-6	1.25E-5
3.8.1	AC Power – Operating, two offsite circuits unavailable	1.12E-3	6.02E-3
3.8.1	AC Power – Operating, one diesel generator unavailable	7.29E-6	1.62E-5
3.8.1	AC Power – Operating, two diesel generators unavailable	2.63E-5	1.15E-4
3.8.1	AC Power – Operating, one offsite circuit and one diesel generator unavailable	1.67E-5	5.11E-5
3.8.1	AC Power – Operating, one load sequencer unavailable	7.48E-6	1.67E-5
3.8.4	DC Power – Operating, one battery charger unavailable	6.02E-6	9.52E-6
3.8.4	DC Power – Operating, one battery unavailable	7.44E-6	1.68E-5

3.8.4	DC Power – Operating, one DC subsystem unavailable	1.31E-4	2.19E-4
3.8.7	Inverters – Operating, train with one inverter unavailable	7.50E-6	1.67E-5
3.8.7	Inverters – Operating, train with two inverters unavailable	7.40E-6	1.62E-5
3.8.9	AC Power Distribution, one subsystem unavailable	3.68E-5	6.64E-5
3.8.9	AC Power Distribution, one vital bus panel unavailable	7.50E-6	1.67E-5
3.8.9	AC Power Distribution, three vital bus panels unavailable	3.22E-5	1.19E-4

The quantitative risk analyses results demonstrate that for each TS LCO condition evaluated, the risk of remaining in Mode 4 is lower than proceeding to Mode 5, which confirms the insights from the qualitative risk assessment.

3.2.4.2.9 Sensitivity Studies

The report investigated important assumptions to determine if their uncertainty would impact the overall conclusion of the analyses.

- Time in Mode 5 Sensitivity

  The time spent in Mode 5, including transition to SDC in Mode 4, was assumed as 70 hours based on TS requirements and on operational data from Westinghouse plants. Since this is a higher risk configuration, a sensitivity study was done using 24 hours as the assumed time. This time was chosen based on the time to complete the transition from Mode 4 to Mode 5. While the quantitative results show that there is a dependency on the time spent in Mode 5 and in transition, the conclusion of less risk by remaining in Mode 4 is not changed.

- SGTR Initiating Event Frequency Sensitivity

  Westinghouse plants include various types of steam generators. The SGTR initiating event frequency was increased by a factor of four from the base case value to investigate the sensitivity of the results. The TR states that the risk increases only about 14% for the Mode 4 case (SGTR is not modeled in Mode 5); therefore, the model is not sensitive to SGTR frequency in either mode, and the conclusion of less risk by remaining in Mode 4 is not changed.

- Probability of Offsite Power Recovery Sensitivity

  The probability of not recovering offsite power within two hours (required in Mode 5) is 0.5, based on Reference 14. A sensitivity case was run assuming the probability of non-recovery is 0.1, a factor of 5 reduction. The revised case reduced the CDP in Mode 5 by only about 14%; therefore, the model is not sensitive to offsite power recovery probability, and the conclusion of less risk by remaining in Mode 4 is not changed.

Based upon these results, the NRC staff concludes that the risk assessment and associated sensitivity studies are adequate to show that the risk changes associated with the proposed changes are either negative or risk neutral for all Westinghouse plants.

3.2.5 Tier 2 and Tier 3 Assessment

The TR identifies the availability of the TDAFW pump as a general tier 2 requirement in order to remain in Mode 4. This will be addressed by revising the bases of each TS action to which a revised end state is applied. Each individual assessment of TS action included an evaluation of tier 2 requirements; no TS LCO action-specific tier 2 restrictions were identified.

Licensees have programs in place to comply with 10 CFR 50.65(a)(4) to assess and manage the risk from proposed maintenance activities. These programs can support licensee decision making regarding the appropriate actions to control risk due to emergent equipment unavailabilities occurring in Mode 4.

3.2.6 Risk Assessment Summary

The NRC staff finds that the Westinghouse risk assessment approach is comprehensive and follows NRC staff guidance as documented in RGs 1.174 and 1.177. In addition, the analyses show that the criteria of the three-tiered approach for allowing TS changes are met as explained below:

- Risk Impact of the Proposed Change (Tier 1). The risk changes associated with the proposed TS changes, in terms of mean yearly increases in CDF and LERF, are risk neutral or risk beneficial. In addition, there are no time intervals associated with the implementation of the proposed TS end state changes during which there is an increase in the probability of core damage or early release with respect to the current end states.
- Avoidance of Risk-Significant Configurations (Tier 2). The availability of the TDAFW pump is addressed as a general tier 2 restriction. No specific restrictions on equipment availability or other controls were identified for preventing risk-significant plant configurations.
- Configuration Risk Management (Tier 3). Licensees have programs in place to comply with 10 CFR 50.65(a)(4) to assess and manage the risk from proposed maintenance activities. These programs can support licensee decision making regarding the appropriate actions to control risk due to emergent equipment unavailabilities occurring in Mode 4.

In addition, the generic risk impact of the proposed end state mode change was evaluated subject to the following:

- Entry into the shutdown mode under consideration (Mode 4 or Mode 5) is for the primary intent of repairing a non-functional component and returning the plant to power as soon as practical.
- While in Mode 4, SG cooling via the TDAFW pump is available in the event of an interruption of decay heat removal, upon operator action to start the pump. This requires availability of the TDAFW pump, a minimum available pressure in the secondary, an available water supply to the pump, and no maintenance or other activities which would delay establishing SG cooling.

3.3 Technical Evaluation Summary

This section is limited to the NRC staff's evaluation of proposed changes associated with cooldown to Mode 4 instead of Mode 5 for specific TS Required Actions in TR WCAP-16294 based on traditional engineering considerations such as defense-in-depth and safety margins. The proposed changes to the TS end state are consistent with standard practices used in setting the allowed outage times and surveillance test intervals. The reduction in outage time in Mode 4 might provide operational flexibility; the change might allow an increased allocation of the plant personnel's time to more safety-significant aspects of plant operation.

The impact of the proposed changes to the TS end state for the Required Actions is evaluated to determine that the changes are consistent with the defense-in-depth philosophy. Risk insights from both qualitative and quantitative risk assessments were used in the TS change-specific justifications. In addition to the risk arguments, defense-in-depth arguments are used to justify each system-specific TS change, in accordance with the "integrated decision-making" process of Regulatory Guides 1.174 and 1.177.

The codes and standards approved for use by the NRC continue to be met. The proposed revisions provide sufficient margin to account for the analysis and data uncertainties. The proposed TS end state and CT change does not adversely affect any assumptions or inputs to the safety analysis. The proposed changes do not allow plant operation in a configuration outside the design basis. For the proposed changes Mode 4 offers additional defense-in-depth and a lower risk level. Therefore there is no impact on safety margins.

Based on technical and regulatory evaluation of the proposed changes to TS by the NRC staff, it is concluded that the proposed changes meet the current regulations.

4.0 LIMITATIONS AND CONDITIONS

Licensees requesting the TS changes to operate their plants in accordance with TR WCAP-16294-NP must include the following in the TS Bases to ensure that the implementation of this TR will be consistent with the NRC staff's evaluation:

1. The primary purpose for entry into Mode 4 end states is to accomplish short duration repairs to restore inoperable equipment.
2. The availability of the TDAFW pump is assured while the plant remains in Mode 4 in accordance with the assumption of the TR.
3. Operational procedures have been established to ensure long-term decay heat removal, should the SG cooling be lost while operating in Mode 4.
4. Include the NRC approved-TR as a reference in the TS Bases.

5.0 CONCLUSION

The NRC staff's evaluation approves only the end state changes specifically identified in Section 3.1 of this SE. The NRC staff finds that the TR used realistic assumptions regarding the plant conditions and the availability of various mitigating systems in analyzing the risks and considering the defense-in-depth and safety margins. Thus the NRC staff concludes that the TR uses realistic assumptions to justify the change in the end state. The NRC staff's review is based on the knowledge of lower RCS pressure in Mode 4, which reduces the severity of a

LOCA, and limits any coolant inventory loss in the event of a LOCA. The staff reiterates that there are high risk plant evolutions and configurations in Mode 4 and that plant operators should be in a heightened state of awareness during Mode 4 status.

In addition, as shown in the November 20, 2009 letter (Reference 21), the NRC staff reviewed the proposed TR WCAP-16294, Rev. 0 changes to NUREG-1431, Revision 3, "Standard Technical Specifications Westinghouse Plants," TS and Bases, as well as additional markups of TR WCAP-16294-NP. The NRC staff found the proposed revisions to the STS and Bases, as described in the November 20, 2009 letter, acceptable, subject to the limitations and conditions shown in Section 4.0. The Technical Specifications Task Force will submit the STS and Bases changes justified in TR WCAP-16294, Rev. 0, (and as shown in Reference 21) as a TSTF Traveler, for NRC staff review and approval.

6.0 REFERENCES

1. Letter from A. Pietrangelo (NEI) to T. Tjader (NRC), " September 9, 2005 (ADAMS Accession Number ML052620374).
2. WCAP-16294-NP, "Risk Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," Westinghouse, August 2005. (ADAMS Accession No. ML052620374).
3. Letter from T. Mensah (NRC) to B. Bradley (NEI), "Request for Additional Information (RAI) Regarding Nuclear Energy Institute Topical Report (TR) WCAP-16294-NP, Rev. 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," June 13, 2007 (ADAMS Accession No. ML071550116).
4. Letter from T. Mensah (NRC) to B. Bradley (NEI), "Request for Additional Information (RAI) Regarding Nuclear Energy Institute Topical Report (TR) WCAP-16294-NP, Rev. 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," October 23, 2008 (ADAMS Accession No. ML082740382).
5. Letter from B. Bradley (NEI) to J. Thompson (NRC), "Response to NRC Request for Additional Information Regarding PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134), December 12, 2007 (ADAMS Accession No. ML080500143).
6. Letter from B. Bradley (NEI) to J. Thompson (NRC), "Response to NRC Request for Additional Information Regarding PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134), November 26, 2008 (ADAMS Accession No. ML090080156).
7. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," US NRC, November 2002.
8. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk Informed Decision Making: Technical Specifications," US NRC, August 1998.
9. NUREG 1431, Vol. 1, Rev. 3.0, "Standard Technical Specifications Westinghouse Plants," US NRC, June 2004.
10. USNRC to R. Bernier, Safety Evaluation of CE NPSD-1186, Rev. 00, "Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs," July 17, 2001 (ADAMS Accession Number ML011980047).
11. USNRC to J. Gray, Safety Evaluation of Topical Report NEDC-32988, Rev. 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants," September 27, 2002 (ADAMS Accession Number ML022700603).

12. USNRC to G. Bischoff, Final Safety Evaluation for BAW-2441, Revision 2, "Risk-Informed Justification for LCO End-State Changes," August 25, 2006 (ADAMS Accession Number ML062130286).

13. Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants, USNRC, May 2000.

14. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

15. NUREG-0800, Standard Review Plan 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007.

16. NUREG-0800, Standard Review Plan 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3, June 2007.

17. USNRC, NUREG-0800, Standard Review Plan 16.1, "Risk-Informed Decision Making: Technical Specifications," Revision 1, March 2007.

18. NUREG/CR-6144, Vol. 2, Parts 1A and 1B, "Evaluation of Potential Severe Accidents during Low Power and shutdown Operations at Surry, Unit 1," June 1994.

19. EPRI, Technical Report 10029987, "Losses of Offsite Power at U. S. Nuclear Power Plants through 2001," April 2002.

20. WCAP-11737, "Low Temperature Overpressurization," March 1989.

21. Letter from B. Bradley (NEI) to T. Mensah (NRC), "Response to A Request for Additional Information (RAI) On Technical Specifications and Bases Associated with PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action End states for Westinghouse NSSS PWRs" (MD5134), November 20, 2009.

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Date:

RESOLUTION OF NUCLEAR ENERGY INSTITUTE (NEI) COMMENTS ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-16294 (TITLE)
NUCLEAR ENERGY INSTITUTE
PROJECT NO. 689

By letter dated December 10, 2009, an NRC draft safety evaluation (SE) regarding the U.S. Nuclear Regulatory Commission's (NRC's) approval of WCAP-16294-NP, was provided for NEI review and comment. By letter dated January 12, 2010, the NEI submitted its comments on the Draft SE. The NRC staff's disposition of the NEI comments on the Draft SE are provided below.

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
1A	P1/L39	Replace: "certain" with "specific TS"	Editorial	Agree. Change made.
1B	P1/L40-L43	Delete the sentence: "The proposed change to the TSs would allow time to perform short-duration repairs which would otherwise necessitate exiting the original Mode of operation. Short duration repairs are on the order of 2-to-3 days, but not more than a week."	WCAP-16294 justified Mode 4 as an acceptable end state. WCAP-16294 did not include any limitations on the period of time that a unit can remain in Mode 4.	Agree with alternate resolution. The NRC staff deleted the following sentence: *Short duration repairs are on the order of 2-to-3 days, but not more than a week.* Basis: The intent of the sentence is to state the NRC staff's expectation for timely licensee action to resolve the need to enter Mode 4. It should be stated that in order to operate the plant in the safest way possible with minimum operator intervention, entry into the shutdown MODE approved in this SE (MODE 4) shall be for the primary purpose of accomplishing inoperable equipment repairs. If the repairs are not completed within a reasonable time period, then the plant may voluntarily transition to cold shutdown.
1C	P1/L43	Delete the sentence: "The Mode 4 TS	10CFR50.65 is applicable during all conditions of plant operation, including	Agree with alternate resolution. Risk assessment and management per

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		endstate is applied, if risk is assessed and managed."	normal shutdown operations. 10CFR50.65(a)(4) requires licensees to assess and manage the increase in risk that may result from proposed maintenance activities. Therefore compliance with this regulation ensures that risk will be assessed and managed during the proposed endstate.	10CFR50.65 is a regulatory requirement, and it is not necessary to restate this as a requirement. This sentence was revised to the following: "The Mode 4 TS end state is applied, and risk is assessed and managed in accordance with 10 CFR 50.65."
2	P1/L44-L46	Delete the following: "(1) entry into the shutdown mode is for a short interval," Renumber Item 2 to 1 and Item 3 to 2.	WCAP-16294 justified Mode 4 as an acceptable end state. WCAP-16294 did not include any limitations on the period of time that a unit can remain in Mode 4.	Disagree. Change not made. Basis: The intent of the sentence is to state the NRC staff's expectation for timely licensee action to resolve the need to enter Mode 4. It should be stated that in order to operate the plant in the safest way possible with minimum operator intervention, entry into the shutdown MODE approved in this SE (MODE 4) shall be for the primary purpose of accomplishing inoperable equipment repairs. If the repairs are not completed within a reasonable time period, then the plant may voluntarily transition to cold shutdown.
3	P2 Table	Include complete titles for TS 3.6.6A, 3.6.6.B, and 3.6.6C.	The revision makes the TS titles complete and consistent with NUREG-1431.	Agree. Change made.
4	P4/L12	Revise "TSs" to "TS".	Editorial	Agree. Change made.
5	P5/L32	Revise: "...assessing proposed permanent TS	The revision is a more accurate description of the	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		changes in AOT." To: "...assessing proposed permanent AOT and STI TS changes."	TS changes discussed in RG 1.77.	Also corrected P5/L35 as a result of NRC staff review: Change 1.777 to 1.177.
6	P5/L45	Revise: "The objective of Tier 2 evaluation..." To: "The objective of the Tier 2 evaluation..."	Editorial	Agree. Change made.
7	P7/L19, L21, L23, L24, L27-L29 and L31. P8/L1, L2, and L3.	Revise TS Mode definitions as follows: Mode 1 - Revise "rated power" to "rated thermal power" Mode 2 – Revise "rated power" to "rated thermal power" and delete the sentence: "The reactor vessel head closure bolts are fully tensioned." Mode 3 – Delete the sentence: "The reactor vessel head closure bolts are fully tensioned." Mode 4 - Delete the sentence: "The RCS pressure would typically be low enough to permit operation of the shutdown cooling (SDC)	The revisions make the Mode definitions in the draft SE consistent with the corresponding TS Mode definitions in NUREG-1431. The operational considerations discussed for Mode 4 and 5 are not part of the Mode definitions in NUREG-1431. The Mode 6 change is editorial	Agree. Changes made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		system, although low pressure is not a requirement of the mode and heat removal may be via the steam generators (SGs)." Mode 5 – Delete the sentence: "The RCS pressure permits operation of the SDC system, since heat removal via the SGs cannot maintain RCS temperature below 200°F." Mode 6 – Revise: "The reactor mode is shutdown..." To: "The reactor in this mode is shutdown..."		
8	P8/L5 and L6	Revise: "End states for unit conditions are prescribed in TS..." To: "End states for unit conditions are prescribed in the TS..." Revise: "The current TS actions require placing a plant in..." To: "The current TS actions require placing the unit in..."	Change is for editorial (the) and consistency (unit) reasons.	Agree. Changes made.
9	P8/L12	Revise: "operations" To: "operation"	Editorial	Agree. Changes made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
10	P8/L14	Revise: "...and actions were implemented..." To: "and guidance was issued..."	The revision clarifies that guidance (NUMARC 91-06) was developed to address shutdown operation. The current wording implies some Action (e.g., TS changes) was implemented.	Agree. Change made.
11	P8/L18	Revise: " accident mitigation in Modes 4 and 5..." To: " accident mitigation (with the remaining operable train of equipment) in Modes 4 and 5..."	The revision clarifies that the assessment of the availability of plant equipment performed in WCAP-16294 for accident mitigation discussed that the remaining operable train of equipment was available.	Disagree. Change not made. Basis: The TR includes a general section describing the plant decay heat removal and accident mitigation in modes 4 and 5. Individual LCO actions are then also evaluated as described in the comment, considering the remaining available equipment. This is addressed in the Page 8, Lines 41-44 of the draft SE. In addition, the resolution of Comment #13, shown below, addresses this.
12	P8/L21	Delete: "to support the return to service"	This change improves the accuracy of statements referring to the assessments performed in WCAP-16294. Other than discussing the Mode change required by the TS, WCAP-16294 does not address specific reasons for the Mode changes.	Disagree. Change not made. Basis: The assessment in the TR for transitioning mode 4 -mode 5 - mode 4 is specifically part of the return-to-service evaluation, and this should be retained. There would be no reason for a plant to enter mode 4 from mode 5 unless it was starting up.
13	P8/L40 and L41	Revise: "...the comparative risk of completing repairs in Mode 4 or proceeding	This change improves the accuracy of statements referring to the assessments performed in	Disagree. Basis: The TR basis is shutdowns required by TS when equipment is not

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
		to Mode 5 for repairs and then returning to Mode 4 for plant startup." To: "...the comparative risk of remaining in Mode 4 *or proceeding* to Mode 5 and then returning to Mode 4 for plant startup.	WCAP-16294. Other than discussing the Mode change required by the TS, WCAP-16294 does not address specific reasons for the Mode changes.	repaired on line. So the identification of repairs is appropriate. To address Comment #11, the following was added to the end off the last sentence, "considering the available equipment for accident mitigation."
14	P9/L16 and L21	Revise: "allowances" To: "allowance"	Editorial	Agree. Change made.
15	P9/L17 and 18	Delete: "(i.e., unanalyzed)"	The term "unanalyzed" implies that a FSAR Chapter 15 safety analysis is associated with this change. The addition of the Note ensures the correct usage of the TS. The Note being added to the affected TS Required Action merely assures that LCO 3.0.4a is applied as intended.	Agree. Change made. It is not necessary to identify conditions outside TS as unanalyzed.
16	P9/L22	Revise: "...therefore, appropriate operational limits are applied with the addition of Notes to affected TS..." To: "...therefore, an appropriate limitation is applied by the addition of Notes to the affected TS..."	The use of "operational" in this context is unnecessary and potentially confusing given that the change only assures the proper use of the existing TS requirement. The other changes are editorial.	Agree. Change made. The proposed note is not an operational limit, so the comment is valid.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
17	P9/L26	Revise: "…applied to corresponding…" To: "applied to the corresponding…"	Editorial	Agree. Change made.
18	P9/L33	Revise: "…mitigate several events." To: "…mitigate potential events."	The affected sentence refers to manual SI actuation capability in Mode 4. The existing wording implies that there are analyzed events for which SI is required in Mode 4. There are no specific events analyzed in Mode 4 for which SI is required.	Agree. Change made. The word several is not essential to the meaning of the sentence.
19	P9/L34-L36	Delete the following sentence: "During the proposed end state (MODE 4), the event mitigating systems, structures and their components such as pumps, and valves must be assessed at predetermined, periodic interval to ensure safety."	WCAP-16294 does not include any requirements or assumptions regarding a periodic assessment of the mitigating systems, structures and their components while in Mode 4. This sentence adds a requirement for periodic monitoring that is not part of the assessments performed in WCAP-16294.	Agree with alternate resolution. This sentence was revised to the following: During the proposed Mode 4 end state, risk is assessed and managed consistent with 10 CFR 50.65.
20	P9/L36-L39 P15/L23-L25 P16/L15-18	P9 Delete the following sentence: "The NRC staff's review was based, in part, on the assumption that while remaining at Mode 4, that plants	Except for providing information regarding the range of RCS pressures that are indicative of operation in Mode 4, WCAP-16294 does not contain any guidance,	Agree with alternate resolution. P9: The sentence was revised to the following: The NRC staff's review is based on the knowledge of lower RCS pressure in Mode 4, which reduces the severity of a LOCA, and limits any

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		would be at the lower end of RCS pressure range so that it will reduce the potential for LOCA and will limit coolant inventory loss in the event of a LOCA, as described further in Sections 3.1.4 and 3.1.5 of this SE." P15 Delete the following: "Though lower RCS pressure in Mode 5 compared to Mode 4 enables the unit to reduce RCS leakage to lower amounts, RCS pressure in Mode 4 must be maintained at the lower end of the pressure range at Mode 4." P16 Delete the following: "Though PIV leakage can be reduced to a lower level in Mode 5 compared to Mode 4 because of lower pressure, the unit should be brought to lower part of Mode 4 where the RCS pressure can be maintained significantly	requirements, or assumptions regarding what RCS pressure should be maintained while a plant is in Mode 4. The WCAP did not provide a basis for this NRC staff assumption (page 9) and requirements (pages 15 and 16).	coolant inventory loss in the event of a LOCA. Basis: The NRC staff notes that current TS Bases state that "the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE *and its potential consequences." The* TS Bases further state that this action "... reduces the factors that tend to degrade the pressure boundary..." P15: Agree. Change made. This sentence was deleted. P16: Agree. Change made. This sentence was deleted.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		lower than at power which will reduce the effects of the PIV leakage."		
21	P9/L41	Revise: "...allowable value of..." To: "...setpoint for..."	The change improves the accuracy of the statement as the setpoint (not allowable value) provides the protection being discussed.	Agree. Change made.
22	P10/L4	Revise: "...coverage of..." To: "...covering..."	Editorial	Agree. Change made.
23	P10/L16, L20, and L31	Delete "...from MODE 5"	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree with alternate resolution. Deletion of "from Mode 5" maintains consistency with the actual marked up TS, and does not change the intent of the NRC staff evaluation. However, this same conclusion is repeated in every 3.1 subsection, without any up front discussion as to its basis. The following paragraph was added in Section 3.1, identifying the fact that the TR does not address entry into Mode 4 from Mode 5, and that this is precluded by a note in each LCO identifying that 3.0.4a is not applicable for Mode 4 entry. "The TR does not address entry into Mode 4 from Mode 5 when the Required Actions are in effect. Such a mode change would be permissible since the revised actions permit continued operation in Mode 4 for an unlimited period of time, and therefore

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
				transitioning from Mode 5 to Mode 4 would be permissible using LCO 3.0.4a of the Standard TS. Therefore, a note is added to each affected LCO which identifies that the provisions of LCO 3.0.4a are not applicable to Mode 4 entry."
24	P10/L29	Revise: "...proposed amendment to delete Required Actions..." To. "...proposed change to revise Required Actions..."	The reference to an amendment and the deletion of Required Actions is incorrect. NUREG-1431 is a generic document.	Agree. Change made.
25	P10/L43	Revise: "...train to prevent inadvertent actuation of CS." To: "...train. Two switches are used to prevent inadvertent actuation of CS.	The revision improves the clarity of the description.	Agree. Change made.
26	P10/L44	Revise the word "set" to "train"	The revision improves the clarity and consistency in the context of this description.	Agree. Change made.
27	P11/L25	Revise: "...are required to be operational." To: "are required to be operable in Mode 4."	The revision is more consistent with the applicable TS terminology for systems required to be operable. The addition of Mode 4 is consistent with the proposed change.	Agree. Change made.
28	P11/L11,L12, L16, and L29	Delete "...from MODE 5"	The description of the Note being added to the affected Action Conditions	Agree with alternate proposal. Refer to the NRC resolution of Comment 23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
29	P11/L29	In the paragraphs addressing the staff's acceptance of each proposed change, the reference to the limitations and conditions of Section 4.0 is inconsistent.	Section 4.0 is applicable to the entire SE, it does not need to be individually referenced for each change. The reference to Section 4.0 is used in some acceptance paragraphs (e.g., P11/L29) and not in others (e.g., P12/L20)	Agree. The reference to the limitations and conditions of Section 4.0 were removed from the individual subsections for each change, since the limitations and conditions are applicable to the entire SE.
30	P12/L12, L18, L37, and L43	Delete "...from MODE 5"	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
			from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
31	P12/L16	Revise; "In addition, CI valves, CS system, and CCW systems are available." To: "In addition, CI valves, CS system, and CCW systems are available in Mode 4.	The revision provides additional clarity and is consistent with the proposed change.	Agree. Change made.
32	P12/L19	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
33	P13/L1	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
34	P13/L4	Add "<u>Coincident With Safety Injection and</u>" after RWST Level – Low Low in the title of Function 7c.	The change makes the title of this instrument Function consistent with the title used in NUREG-1431.	Agree. Change made.
35	P13/L22 and L32	Delete: "...from Mode 5."	The description of the Note being added to the	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
36	P13/L33	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
37	P14/L6, L15, L34, and L42	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
			"from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
38	P14/L16 and L43	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
39	P15/L19	Add: "or primary to secondary leakage is not within limit" after "or pressure boundary leakage exists"	This change makes the Conditions addressed by Required Action B.2 complete and consistent with NUREG-1431.	Agree. Change made.
40	P15/L20 and L34	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
41	P15/L32	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
42	P15/L21	Revise: "A RCS leakage that is considered to be not large enough…" To: "RCS leakage that is not large enough…"	Editorial	Agree. Change made.
43	P15/L28-31	Revise the following sentence: "If any pressure boundary leakage leakage exists, or if unidentified leakage, identified leakage, or primary to secondary leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences." To: "If RCS operational leakage is not within limits for reasons other than pressure boundary leakage or primary to secondary leakage the leakage must be reduced to within the	The revised description of the applicable RCS operational leakage requirements is more consistent with the actual TS requirements specified in NUREG-1431.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		limit in 4 hours consistent with Required Action A.1. If operational leakage is not restored to within the limit in 4 hours, in accordance with Required Action A.1, or pressure boundary leakage exists, or primary to secondary leakage is not within the limit, Required Actions B.1 and B.2 become applicable. Required Actions B.1 and B.2 require that the unit be placed in Mode 3 within 6 hours and Mode 4 within 12 hours. Thus, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequence."		
44	P16/L13, L23, and L40	Delete: "…from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
			of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
45	P16/L24	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
46	P16/L41	Add the word "an" before the word "indication"	Editorial	Agree. Change made.
47	P17/L4,	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			(e.g., B&W and CE).	
48	P17/L5 and L35	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
49	P17/L14, L16, and L18	L14 – Capitalize "a" in word "action" L16 – Delete "RHR" and capitalize the word "operable" L18 – Add "sub" to word system.	The revisions make the text consistent with the NUREG-1431 TS being discussed.	Agree. Change made.
50	P17/L27	Revise "C.1" to "A.1"	The Required Action being revised by WCAP-16294 is A.1 not C.1.	Agree. Change made.
51	P18/L5	Revise "E.2" to 'C.2"	The Required Action being discussed in this text is C.2 not E.2.	Agree. Change made.
52	P18/L6 and L25	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
53	P18/L7	Revise: "Since SI and recirculation are not available due to inoperable RWST..." To: "Since SI and recirculation may not be available due to an inoperable RWST..."	The revision clarifies that variations in level or boron concentration outside the limits of the RWST TS would not necessarily make SI and recirculation unavailable, although the capability of the systems to perform their intended safety function may be degraded.	Agree. Change made.
54	P18/L12	Add the word "the" before the word "lower".	Editorial	Agree. Change made.
55	P18/L16	Delete the following: "for reasons other than boron concentration or temperature"	The revision simplifies the sentence and eliminates an incomplete list of reasons why the RWST may be inoperable. If the deleted text is retained, boron concentration and temperature not within limits must be added to this sentence to list all the reasons why the RWST may be inoperable.	Agree. Change made.
56	P18/L26	Revise: "amendment" To: "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
57	P19/L5	Revise: "60 hours" to	This change is consistent	Agree. Change made.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		"54 hours"	with the TS changes previously transmitted to the NRC in NEI Letter dated 9/20/09 (B. Bradley (NEI) to T. M. Mensah (NRC)). Note this letter was actually sent by email to the NRC on 11/20/09	
58	P19/L7, L18, and L44	Delete: "...from Mode 5"	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree with alternate resolution. Refer to NRC resolution of Comment #23.
59	P19/L13	Revise: "Action F" to "Condition F"	Reflects the correct TS terminology.	Agree. Change made.
60	P19/L16	Capitalize "b" in word "bases" Revise: "technical specification" to "TS"	Reflects consistent TS terminology and use of acronyms.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
61	P19/L18	Delete "applicants"	This word is not required in the SE for a TR. It is not used in other similar sentences in this SE.	Agree. Change made.
62	P20/L2 and L3	Delete the following: "...and it also serves to remove iodine."	No credit is taken for iodine removal in TS 3.6.6.B of NUREG-1431.	Agree. Change made.
63	P20/L3	Revise: "Action G" to "Condition G"	Reflects the correct TS terminology.	Agree. Change made.
64	P20/L3 and L4	L3 - Delete: "...two CS trains are inoperable or..." L4 – Revise "...more trains inoperable..." To: "...more trains are inoperable	The revision simplifies the L3 statement consistent with the TS wording and makes an editorial change in L4.	Agree. Change made.
65	P20/L5	Delete: "The proposed changes to the TS bases were also reviewed as input to the TS change."	This sentence is applicable to the TS and Bases revised consistent with the changes included in NEI Letter dated 9/20/09 but actually transmitted by email dated 11/20/09 from B. Bradley (NEI) to T. M. Mensah (NRC). TS 3.6.6B, where this sentence was applied in P20/L5 is not addressed in the NEI letter.	Agree. Change made.
66	P20/L7,L32, and L43	Delete: "...from Mode 5..."	The description of the Note being added to the affected Action Conditions	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
67	P20/L9	Add the word "System" to the TS title.	This change makes the TS title consistent with NUREG-1431.	Agree. Change made.
68	P20/L30	Revise: "60 hours" to "54 hours"	This change is consistent with the TS changes previously transmitted to the NRC in NEI Letter dated 9/20/09 (B. Bradley (NEI) to T. M. Mensah (NRC)). Note this letter was actually sent by email to the NRC on 11/20/09	Agree. Change made.
69	P20/L41	Capitalize "b" in word "bases"	Reflects consistent TS terminology	Agree. Change made.
70	P20/L41	Add: "TS" in front of the word "change"	Clarification	Agree. Change made.
71	P21/L16, and	Delete: "...from Mode	The description of the	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
	L24-25	5…"	Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
72	P21/L23 and L24	Delete: "The proposed changes to the TS bases were also reviewed as input to the TS change."	This sentence is applicable to the TS and Bases revised consistent with the changes included in NEI Letter dated 9/20/09 but actually transmitted by email dated 11/20/09 from B. Bradley (NEI) to T. M. Mensah (NRC). TS 3.6.6D, where this sentence was applied (in P21/L23&24) is not addressed in the NEI letter.	Agree. Change made.
73	P21/L20	Revise: "…in full…" to "…at full…"	Editorial	Agree. Change made.

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
74	P22/L2, and L46	Revise: "60 hours" to "54 hours"	This change is consistent with the TS changes previously transmitted to the NRC in NEI Letter dated 9/20/09 (B. Bradley (NEI) to T. M. Mensah (NRC)). Note this letter was actually sent by email to the NRC on 11/20/09	Agree. Change made.
75	P22/L4, L13, and L48	Delete: "...from Mode 5..."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree with alternate resolution. Refer to NRC resolution of Comment #23.
76	P22/L9	Revise: "...in full power." To: "...at full power."	Editorial	Agree. Change made.
77	P22/L11	Capitalize "b" in word "bases"	Reflects consistent TS terminology	Agree. Change made.

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
78	P22/L38 and L39	L38 - Add the word "spray" in front of "additive tank". L39 – Add the word "containment" in front of "spray pump suction."	Revises the system names to be consistent with the TS terminology used in NUREG-1431.	Agree. Change made.
79	P23/L2, L5,L35, L38 and L40	Delete the word "applicant"	This word is not required in the SE for a TR. It is not used in other similar sentences in this SE.	Agree. Change made.
80	P23/L2 and L35	Delete brackets around "e" in word "events"	Editorial	Disagree. No change made. The word "events" is capitalized in the TR as the first word of a sentence, so technically making it lower case and bracketing it is appropriate for the SE.
81	P23/L9 and L44	Revise word "cooling" to "coolant"	Correct system terminology.	Agree. Change made.
82	P23/L32	Add the word "The" before "Mode 5"	Editorial	Agree. Change made.
83	P23/L33	Revise "60 hours" to "54 hours"	This change is consistent with the TS changes previously transmitted to the NRC in NEI Letter dated 9/20/09 (B. Bradley (NEI) to T. M. Mensah (NRC)). Note this letter was actually sent by email to the NRC on 11/20/09	Agree. Change made.
84	P24/L19 and L33	Delete: "...from Mode 5..."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
			NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
85	P24/L27	Delete the word "applicant"	This word is not required in the SE for a TR. It is not used in other similar sentences in this SE.	Agree. Change made.
86	P24/L26-L28	Revise: "Action A states that with one ICS train inoperable the licensee has seven days to restore the ICS train to OPERABLE status. The TR requests that if Action A cannot be accomplished within the seven day limit, Required Action B.1 *requires the unit to be in* Mode 3 in six hours and in Mode 4 in twelve hours."	Minor corrections, clarifications, and editorial changes to improve understanding and clarity.	Agree. Change made.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		To: "For Condition A, one ICS train inoperable, seven days are allowed to restore the ICS train to OPERABLE status. The TR requests that if Required Action A.1 cannot be accomplished within the seven day limit, Required Action B.1 be revised to allow the unit to be in Mode 3 in six hours and in Mode 4 in twelve hours."		
87	P24/L31 and L32	Delete: "The proposed changes to the TS bases were also reviewed as input to the TS change."	This sentence is applicable to the TS and Bases revised consistent with the changes included in NEI Letter dated 9/20/09 but actually transmitted by email dated 11/20/09 from B. Bradley (NEI) to T. M. Mensah (NRC). TS 3.6.11, where this sentence was applied (in P24/L31&32) is not addressed in the NEI letter.	Agree. Change made.
88	P25/L3, L20, and L44	Delete: "...from Mode 5..."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
		-	transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
89	P25/L11 and L12	Revise: "Action A states that if one vacuum relief line is *inoperable, the licensee* has 72 hours to restore vacuum relief line to OPERABLE status. To: "For Condition A, if one vacuum relief line is inoperable, 72 hours are allowed to restore the vacuum relief line to OPERABLE status."	Editorial clarifications	Agree. Change made.
90	P25/L47 and L48	Revise: "Action A states that if one SBACS train is inoperable the licensee has seven days to	Editorial clarifications	Agree. Change made.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		restore SBACS train to OPERABLE status." To: "For Condition A, one SBACS train is inoperable, seven days are allowed to restore the SBACS train to OPERABLE status."		
91	P26/ L1-L4	Revise: "The energy released to containment from a LOCA will be lower than for the maximum design basis accident. CS systems and containment cooling systems, and ECCS will be available. The loads on these systems will be well within their design basis." To: "The energy released to containment from a LOCA will be lower than for the limiting design basis accident. The CS system, containment cooling system (for a dual containment design) and ECCS will be available. The	Clarifications, and editorial changes to improve understanding.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		capability of these systems will be well within their design basis should an event occur in Mode 4."		
92	P26/L12	Delete: "The proposed changes to the TS bases were also reviewed."	This sentence is applicable to the TS and Bases revised consistent with the changes included in NEI Letter dated 9/20/09 but actually transmitted by email dated 11/20/09 from B. Bradley (NEI) to T. M. Mensah (NRC). TS 3.6.11, where this sentence was applied (in P24/L31&32) is not addressed in the NEI letter.	Agree. Change made.
93	P26/L13, L14, and L35	Delete: "...from Mode 5..."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
			conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
94	P26/L39, L40 and L41	Revise: "Recirculation of containment air from upper containment to lower containment compartment to assist cooling the containment atmosphere and limiting post accident pressure in containment. Containment air cooling for a LOCA is still required for operation in Mode 4." To: "Recirculation of containment air from the upper containment to lower containment compartment assists in cooling the containment atmosphere and limiting post accident pressure in containment. Containment air cooling for a LOCA or SLB is still required in Mode 4."	Clarifications, and editorial changes to improve understanding.	Agree. Change made.
95	P26/L42 and L43	Revise: "The CS system will still be operational to	Clarifications, and editorial changes to improve understanding.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		provide cooling in the upper containment compartment. The energy released into containment from a LOCA while in Mode 4…" To: The CS system will still be available to provide cooling in the upper containment compartment. The energy released into containment from a LOCA or SLB while in Mode 4…"		
96	P26/L46	Add the word "to" before the word "within"	Editorial	Agree. Change made.
97	P27/L3-L5 and L7 – L11	Delete the following: L3-L5 "The ECCSs are designed in accordance with 10CFR50, Appendix A, GDC 35, 36, "Inspection of emergency core cooling system," and 37, "Testing of emergency core cooling system."" L7-L11 "Criteria 36 and 37 address system inspection and testing.	The reference to the ECCS GDCs are correct, however, the system design criteria are not relevant to the proposed changes in WCAP-16294. In addition the discussion on P27 pertains to the ARS not the ECCS.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		Criterion 35 states: A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts."		
98	P27/L3	Revise: "…or manual actuation." to "…or by manual actuation."	Editorial	Agree. Change made.
99	P27/L13	Revise: "With one train of containment air return inoperable…" To: "With one train of ARS *inoperable…*"	Consistent terminology for the system.	Agree. Change made.
100	P27/L14-L28	Delete the following: LCO 3.6.14 requires "[t]wo ARS [air return systems] trains shall be OPERABLE." With two trains inoperable there will not be any forced ventilation to provide air mixing to prevent	The discussion of two trains inoperable and the requirements of LCO 3.0.3 is correct, however, it is not relevant to the changes proposed in WCAP-16294, which is not applicable to LCO 3.0.3.	Agree. Change made.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		hydrogen accumulation. According to 10 CFR 50.36(c)(2) *Limiting conditions for operation*, (i) "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met." NUREG-1431, Standard Technical Specifications Westinghouse Plants, LCO 3.0.3, states "[w]hen an LCO is not met and the associated ACTIONS *are not met*, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not		

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours, b. MODE 4 within 13 hours, and c. MODE 5 within 37 hours.".		
101	P27/L30	Delete: "The proposed changes to the TS bases were also reviewed."	This sentence is applicable to the TS and Bases revised consistent with the changes included in NEI Letter dated 9/20/09 but actually transmitted by email dated 11/20/09 from B. Bradley (NEI) to T. M. Mensah (NRC). TS 3.6.14, where this sentence was applied (in P27/L30) is not addressed in the NEI letter.	Agree. Change made.
102	P27/L33 and L35	L33 Revise: "...CS systems..." To: "...CS system..." L35 Revise: "...REQUIRED ACTION..." To "...Required Action..."	Consistent terminology.	Agree. Change made.
103	P27/L35	Delete: "...from Mode 5..."	The description of the Note being added to the	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
			affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
104	P27/L34	Revise: "..."in the event of a LOCA..." To: "...in the event of a LOCA or SLB..."	To be complete. The system is designed for both DBAs.	Agree. Change made.
105	P28/L9	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
106	P27/L13-15	Revise: "Action A states that if one ice condenser floor drain is inoperable, the licensee has one hour to restore the ice condenser floor drain to OPERABLE status." To: For Condition A, one ice condenser floor drain inoperable, one hour is allowed to restore the ice condenser floor drain to OPERABLE status."	Correct terminology and for consistency.	Agree. Change made.
107	P27/L15-24	Delete the following: "If more than one drain is inoperable LCO 3.0.3, states "[w]hen an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the	The discussion of two drains inoperable and the requirements of LCO 3.0.3 are correct but it is not relevant to the changes proposed in WCAP-16294, which is not applicable to LCO 3.0.3.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours, b. MODE 4 within 13 hours, and c. MODE 5 within 37 hours." Since there is no Action for more than one floor drain inoperable LCO 3.0.3 is applicable."		
108	P27/L30	Revise: "...in the event of a LOCA..." To: "...in the event of a LOCA or SLB..."	To be complete. The system is designed for both DBAs.	Agree. Change made.
109	P27/L36-L39	Revise: "Action A states that if one ice condenser floor drain is inoperable, the licensee has one hour to restore ice condenser floor drain to	Correct terminology and for consistency.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		OPERABLE status. Action B states if one refueling canal drain is inoperable there is a one hour limit to restore refueling canal drain to OPERABLE status." To: "For Condition A, one ice condenser floor drain inoperable, one hour is allowed to restore the ice condenser floor drain to OPERABLE status. For Condition B, one refueling canal drain inoperable, one hour is allowed to restore the refueling canal drain to OPERABLE status."		
110	P28/L39-L48	Delete the following: "If more than one drain is inoperable LCO 3.0.3, states "[w]hen an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not	The discussion of two drains inoperable and the requirements of LCO 3.0.3 are correct but it is not relevant to the changes proposed in WCAP-16294, which is not applicable to LCO 3.0.3.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in: a. MODE 3 within 7 hours, b. MODE 4 within 13 hours, and c. MODE 5 within 37 hours." Since there is no Action for more than one floor drain inoperable LCO 3.0.3 is applicable."		
111	P29/L1	Revise: "In the event of a LOCA..." To: "In the event of a LOCA or SLB..."	To be complete. The system is designed for both DBAs.	Agree. Change made.
112	P29/L6	Delete: "The proposed changes to the TS bases were also reviewed as input to the TS change."	This sentence is applicable to the TS and Bases revised consistent with the changes included in NEI Letter dated 9/20/09 but actually transmitted by email dated 11/20/09 from B. Bradley (NEI) to T. M. Mensah (NRC). TS 3.6.14, where this	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			sentence was applied (in P27/L30) is not addressed in the NEI letter.	
113	P29/L8 and L9	Revise: "Based on the above assessment, with LCO 3.0.4.a not applicable for entry into Mode 4 from Mode 5, of the ability to provide a reasonable assurance that there will be adequate water..." To: "Based on the above assessment, with LCO 3.0.4.a not applicable for entry into Mode 4, the ability to provide reasonable assurance that there will be adequate water..."	Editorial changes and: The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree. Change made.
114	P29/L13-L23	Delete the following: "3.1.21 Mode 4 Secondary Side Steam Pressure TR WCAP-16294-NP, Revision 0, indicated that secondary side steam pressure would be at normal operating pressure. The NRC staff requested that the TR applicant verify the	This SE Section does not involve the approval of a TS change. The Section addresses a change to the TR which is a statement of fact regarding the design and operation of typical Westinghouse PWRs that will be incorporated into the approved version of the TR.	Disagree. Change not accepted. Basis: Since the NRC staff's question will result in a revision to the statement in TR WCAP-16294, these words in the SE will remain.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		secondary side steam pressure taking into consideration the reduced reactor cooling system average temperature in Mode 4. The TR applicant revised the statement in TR WCAP-16294-NP, Revision 0, to indicate that while in Mode 4, the secondary side steam pressure will be less than normal operating pressure. The TR applicant determined that there will be sufficient pressure available for most plants to operate the turbine driven auxiliary feedwater pumps. This will assure the defense-in-depth will remain available while remaining in Mode 4. The NRC staff finds the revision acceptable." Note: The Deletion of this Section requires re-numbering subsequent Sections.		
115	P29/L45	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
116	P29/L2	Revise: "…associated with Mode 4…" To: "…associated with a cooldown to Mode 4…"	Consistency with WCAP-16294.	Disagree. Change not accepted. The risk assessment of Table 3.2.1 evaluates remaining in mode 4 compared to cooling down to mode 5 and then returning to mode 4. Therefore, adding this phrase is not appropriate.
117	P29/L8,L28, and L35	Delete: "…from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
			include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
118	P29/L9 and L36	Revise: the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
119	P29/L12	Revise: "The SWS..." to "A typical SWS..."	The plant specific design of the SWS can vary somewhat from the typical design described in NUREG-1431.	Agree. Change made.
120	P29/L30	Add the word "the" before the word "shutdown"	Editorial	Agree. Change made.
121	P30/L10 and L11	L10 - Add the word "of" before the word "Condition" L11 – Add the word "Condition" in before the letter "A"	Editorial	Agree. Change made.
122	P30/L12, L13, L22 and L44	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
123	P30/L20	Revise: "...during the shutdown mode." To: "...in Mode 4."	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Change partially accepted. The context of the phrase "during the shutdown mode" refers to both Modes 4 and 5. This was changed to "**during shutdown modes**", but should not be Mode 4 exclusively.
124	P30/L23	Revise: the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
125	P30/L42	Add the word "of" before the word "Condition"	Editorial	Agree. Change made.
126	P31/L3	Delete the word "both"	Editorial	Agree. Change made.
127	P31/L5, L18, L25 and L26	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T.	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
128	P31/L6 and L27	Revise: the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
129	P32L9/	Delete: "...of Condition A..."	The shutdown Required Actions of Condition C (being discussed in the affected text) are applicable to both Conditions A and B. Therefore, to be accurate, both Conditions A and B should be included in the discussion or simply delete Condition A.	Agree. Change made.
130	P32/L11 and L19	Delete "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
131	P32/L15 and Line17	Revise references to "shutdown modes" to "Mode 4"	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Change partially accepted. The context of the phrase "in shutdown mode" appears to refer to both Modes 4 and 5, and so should be changed to "**during shutdown modes**", but should not be Mode 4 exclusively. In addition, the text should also be revised to read "**postulated** accidents".
132	P32/L20	Revise the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
133	P33/L2, L8, L32, and L38	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
134	P33/L5 and L7	L5 Revise: "…Condition E…" To: "…Required Action E.1…" L7 Revise: "…form Mode 5…" To: "…from Mode 5…"	Compliance with the Required Action is what makes the probability of a fuel handling accident lower. Editorial	Agree. Change made.
135	P33/L9 and L39	Revise the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
136	P33/L36	Revise: "…shutdown modes." To: "…Mode 4"	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Refer to resolution of Comment #123.
137	P34/L33 and L35	Revise: "shutdown modes" To: "Mode 4"	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Refer to resolution of Comment #123.
138	P34/L37	Delete: "…from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	***Draft SE Page & Line Number***	***NEI Comment***	***NEI Justification***	***NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)***
			to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
139	P34/L38	Revise the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
140	P35/L23 and L32	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
141	P35/L24	Revise: "Quantitative risk evaluation..." To: "A quantitative risk evaluation..."	Editorial	Agree. Change made.
142	P35/L25	Revise "Results are summarized below." To: "The results are summarized below."	Editorial	Agree. Change made.
143	P35/L25	Revise: "The analysis show..." To: "The evaluation shows..."	A quantitative risk evaluation is being discussed, not an analysis.	Agree. Change made.
144	P35/L28 and L30	Revise the references to "shutdown modes" to: "Mode 4"	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Refer to resolution of Comment #123.
145	P35/L33	Revise the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
146	P36/L4, L12, and L40	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of	Agree with alternate resolution. Refer to NRC resolution of Comment #23.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			"from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	
147	P36/L8, L9, and L10	Revise the references to "shutdown modes" to: "Mode 4"	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Refer to resolution of Comment #123.
148	P36/L13	Revise the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
149	P36/L29 and L30	Revise: "...Table B on TS page 3.8.9-1." To: "Table B 3.8.9-1."	This is a Bases Table. The Table is labeled B 3.8.9-1	Agree. Change made.
150	P36/L36	Add the word "to" before the word "be"	Editorial	Agree. Change made.
151	P37/L1-L5	Delete the following: "If the inoperable distribution subsystem cannot be restored to operable status within the required CT, the unit must be brought to a MODE in which the LCO does not apply. The proposed allowed CTs are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly	This statement is no longer applicable. See the proposed Required Actions on P37/L37-L40. The unit is not brought to a Mode in which the LCO does not apply. The unit can remain in Mode 4 with the proposed change.	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		manner and without challenging plant systems."		
152	P37/L6 and L8	Revise the references to "shutdown modes" to "Mode 4"	Mode 4 is the applicable shutdown Mode being discussed in this case.	Disagree. Refer to resolution of Comment #123.
153	P37/L10	Delete: "...from Mode 5."	The description of the Note being added to the affected Action Conditions is inconsistent with the TS and Bases markups transmitted to the NRC in NEI letter dated 9/20/09 (but actually sent by email to the NRC on 11/20/09 from B. Bradley (NEI) to T. Mensah (NRC)). The text of the Note does not include "from MODE 5". The change (deletion of "from Mode 5" from the Note) was made to conform to the similar Notes used in the other NSSS Vendors TSTFs (e.g., B&W and CE).	Agree with alternate resolution. Refer to NRC resolution of Comment #23.
154	P37/L11	Revise the word "amendment" to "change"	In this case the word amendment is not applicable. NUREG-1431 is a generic document.	Agree. Change made.
155	P37/L11	Revise "Required Action B.2" to "Required Action D.2"	To be consistent with the Required Actions discussed on P37/L37-L40	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
156	P38/L19, L20 and L21	L19 – Revise: "Plant shutdowns due to exceeding TS CTs are infrequent events." To: "Plant shutdowns due to not meeting the TS Required Action and associated CTs are infrequent events." L20 - Delete "…and complete repairs…" L21 - Delete "…and conducting repairs in Mode 5,…"	L19 – Reflects correct TS terminology and usage. L20&L21 - This change improves the accuracy of statements referring to the assessments performed in WCAP-16294. Other than discussing the Mode change required by the TS, WCAP-16294 does not address specific reasons for the Mode change.	L19: Agree. Change made. L20&L21: Disagree. Change not made. Refer to NRC resolution of Comment #12.
157	P39/L2, L7, L8, L16,	L-2 - Revise "…subject to…" to "…based on…" L7 – Revise "…the shutdown mode under consideration…" to "…Mode 4…" L-7 – Delete"…a short interval with"	L-2 - Editorial L7 - Mode 4 is the applicable shutdown Mode being discussed in this case. L-7 - It should be noted that, currently, plants can remain in Mode 5 indefinitely since the TS do not limit the time a plant can stay in Mode 5. A sensitivity study that is discussed in WCAP-16294 (Section 6.5.1, "POS 4 Time Sensitivity") demonstrates that even	L-2: Agree. Change made. L7: Disagree. The assumption applies equally to the evaluations of mode 4 and mode 5, so the original wording is correct. Made clarification as noted: "…under consideration (Mode 4 or Mode 5)". L-7: Agree. Change made. Deleting reference to "short interval" does not change the assumption, so no objection.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			when the time spent in Mode 4 is twice the time in Mode 5, the risk of being in Mode 4 is still less than being in Mode 5. Therefore, to state that a plant can only stay in Mode 4 for a short time is not consistent with the assessments discussed in WCAP-16294 and limiting the time in Mode 4 could force a plant to transition to Mode 5 and introduce more plant risk than if the plant remained in Mode 4. The time a plant spends in Mode 4 will depend on the time it takes to resolve the issue that caused the plant to enter Mode 4 and should not be otherwise limited by statements to the effect that only a short time in Mode 4 is acceptable. It should also be noted that plants would not remain in Mode 4 longer than necessary to resolve the issue(s) that caused entry into Mode 4 due to commercial considerations.	
		L8 – Delete "…of that entry being…"	L8 – Editorial due to deletion of "short interval" in L7.	L8 – Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		L16 - add "the" before "shutdown modes" L16 – Delete "…short duration…"	L16 – Editorial L16 – Deletion of "short interval." See second L7 comment above for the justification.	L16- Agree. Change made. L16- Agree. Change made. Deleting reference to "short duration" does not change the assumption, so no objection.
158	P41/L9	Revise: the word "probability" to "probabilities"	Editorial	Agree. Change made.
159	P41/ L27 and L28	Delete the following: "…during equipment repairs…"	This change improves the accuracy of statements referring to the assessments performed in WCAP-16294. Other than discussing the Mode change required by the TS, WCAP-16294 does not address specific reasons for the Mode change.	Disagree. Change not made. The assessment in the TR for transitioning mode 4 -mode 5 - mode 4 is specifically part of the return-to-service evaluation, and this should be retained. There would be no reason for a plant to enter mode 4 from mode 5 unless it was starting up.
160	P41/L38	Revise: "…(Table 6-5)." To: "…(Tables 6-4 and 6-5)."	More consistent with WCAP-16294	Agree. Change made.
161	P42/L2	Delete the following: "…after equipment repairs…"	This change improves the accuracy of statements referring to the assessments performed in WCAP-16294. Other than discussing the Mode change required by the TS, WCAP-16294 does	Disagree. Change not made. The assessment in the TR for transitioning mode 4 -mode 5 - mode 4 is specifically part of the return-to-service evaluation, and this should be retained. There would be no reason for a plant to enter mode 4 from mode 5 unless it was starting up.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
			not address specific reasons for the Mode change.	
162	P42/L23	Revise: "ECCS is not available..." To: "ECCS is not required to be available..." Delete: the word "as"	Improve the accuracy of the statement and consistent with a similar statement on SE P44/L15. Editorial	Agree. Change made.
163	P44/L16 and L17	Delete: "...during equipment repairs...."	This change improves the accuracy of statements referring to the assessments performed in WCAP-16294. Other than discussing the Mode change required by the TS, WCAP-16294 does not address specific reasons for the Mode change.	Disagree. Change not made. The assessment in the TR for transitioning mode 4 -mode 5 - mode 4 is specifically part of the return-to-service evaluation, and this should be retained. There would be no reason for a plant to enter mode 4 from mode 5 unless it was starting up.
164	P46/L34 and L37	Revise: "In order to employ the models of the two end states including transition between Modes 4 and 5, the expected time spent in each mode must be determined." To: "In order to employ the models of the two end states including transition between Modes 4 and 5, the	Edited to be consistent with WCAP-16294, Section 6.3.1.2, "Time in Each Plant Operating State."	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		estimated time anticipated to be in each mode must be determined."		
165	P46/L-L	Revise:" The NRC staff reviewed the assumptions and bases for the times spent in Mode 4 and for a transition to Mode 5 and return to Mode 4 after completion of repairs and finds them to be reasonable and appropriate for this application." To: "The NRC staff reviewed the assumptions and bases for the times anticipated to be in Mode 4 and for a transition to Mode 5 and return to Mode 4, following a forced non-refueling outage and finds them to be reasonable and appropriate for this application."	Edited to be consistent with WCAP-16294, Section 6.3.1.2, "Time in Each Plant Operating State."	Agree. Change made.
166	P52/L9 and L10	Revise: "This will be addressed by revising the bases of each TS action to which a revised end state is	This change makes the SE text consistent with the TS markups submitted with WCAP-16294.	Disagree. Change not made. Basis: The Bases for each LCO should identify Mode 4 as acceptable contingent on availability of the TDAFW pump, as originally specified.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		applied." To: "This will be addressed by revising the Bases of TS 3.7.5, "AFW System," to identify this Tier 2 requirement."		
167	P52/L37	Revise: "...the shutdown mode under consideration..." To: "...Mode 4..."	Mode 4 is the applicable shutdown Mode being discussed in this case. It should be noted that, currently, plants can remain in Mode 5 indefinitely since the TS do not limit the time a plant can stay in Mode 5. A sensitivity study that is discussed in WCAP-16294 (Section 6.5.1, "POS 4 Time Sensitivity") demonstrates that even when the time spent in Mode 4 is twice the time in Mode 5, the risk of being in Mode 4 is still less than being in Mode 5. Therefore, to state that a plant can only stay in Mode 4 for a short time is not consistent with the assessments discussed in WCAP-16294 and limiting the time in Mode 4 could force a plant to transition to Mode 5 and introduce more plant risk than if the	Disagree. The assumption applies equally to the evaluations of mode 4 and mode 5, so the original wording is correct. Made clarification as noted: "...under consideration (Mode 4 or Mode 5).

Comment Number	*Draft SE Page & Line Number*	*NEI Comment*	*NEI Justification*	*NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)*
		Delete: "…a short interval with…"	plant remained in Mode 4. The time a plant spends in Mode 4 will depend on the time it takes to resolve the issue that caused the plant to enter Mode 4 and should not be otherwise limited by statements to the effect that only a short time in Mode 4 is acceptable. It should also be noted that plants would not remain in Mode 4 longer than necessary to resolve the issue(s) that caused entry into Mode 4 due to commercial considerations.	Agree. Change made. Deleting reference to "short duration" does not change the assumption, so no objection.
168	P53/L18	Revise: "The proposed TS AOT or end state change…" To: "The proposed TS end state and CT change…"	More accurately reflects the changes proposed in WCAP-16294.	Agree. Change made.
169	P53/L49-L51	Revise: "Licensees requesting the TS changes to operate their plants in accordance with TR WCAP-16294-NP must	This statement is revised to be consistent with the changes made to the list of commitments that follow the statement.	Agree with alternate resolution. This sentence was revised to the following: Licensees requesting the TS changes to operate their plants in accordance with TR WCAP-16294-NP must include the following in the TS

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		commit to the following requirements in the TSs or its associated bases. These commitments assure that the implementation of this TR will be consistent with the NRC staff's evaluation: To: "Licensees requesting the TS changes justified in TR WCAP-16294-NP must commit to the following requirements in the TS Bases. This commitment assures that the implementation of this TR will be consistent with the NRC staff's evaluation:"		Bases to ensure that the implementation of this TR will be consistent with the NRC staff's evaluation:
170	P53/L36-L39	Delete the following: "Entry into the shutdown modes approved in this SE shall be for the primary purpose of accomplishing <u>short time</u> repairs to restore inoperable equipment. Appropriate procedures and exemptions should be secured from the NRC if the plants are likely to stay more than the duration stipulated by the amended TS	A sensitivity study that is discussed in WCAP-16294 (Section 6.5.1, "POS 4 Time Sensitivity") demonstrates that even when the time spent in Mode 4 is twice the time in Mode 5, the risk of being in Mode 4 is still less than being in Mode 5. Therefore, to state that a plant can only stay in Mode 4 for a short time is not consistent with the assessments discussed in	Agree with alternate resolution. This sentence was revised to the following: The primary purpose for entry into Mode 4 endstates is to accomplish short duration repairs to restore inoperable equipment. Basis: The intent of these sentences is to state the staff's expectation for timely licensee action to resolve the need to enter Mode 4. It should be stated that in order to operate the plant in the safest way possible with minimum operator intervention, entry into the

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		duration."	WCAP-16294 and limiting the time in Mode 4 could force a plant to transition to Mode 5 and introduce more plant risk than if the plant remained in Mode 4. The time a plant spends in Mode 4 will depend on the time it takes to resolve the issue that caused the plant to enter Mode 4 and should not be otherwise limited by statements to the effect that only a short time in Mode 4 is acceptable. It should also be noted that plants would not remain in Mode 4 longer than necessary to resolve the issue(s) that caused entry into Mode 4 due to commercial considerations. The NRC does not issue procedures to licensees. There are no exceptions required to any regulations. No limitation on remaining in Mode 4 is specified.	shutdown MODE approved in this SE (MODE 4) shall be for the primary purpose of accomplishing inoperable equipment repairs. If the repairs are not completed within a reasonable time period, then the plant may voluntarily transition to cold shutdown.
171	P53/L40-L43	Delete: "The requested end state changes do not prohibit licensee from entering cold shutdown, if they wish to do so for	This is not a commitment that needs to be made, and the TS do not prohibit going to a mode that is lower than the endstate required by the TS. The current TS allowance to	Agree. Change made.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		operational reasons or maintenance requirements. In such cases, the specific requirements associated with the requested end state changes do not apply."	enter lower Modes is not impacted by this change.	
172	P54/L1 and L2	Delete: "Appropriate plant procedures and administrative controls shall be used when the plant is operated in the proposed end state."	The only requirement to remain in the Mode is that the TDAFW pump be available, which is addressed by an L&C in comment 174 below.	Agree. Change made.
173	P54/L3-L5	Delete: "Entry in to the proposed end states shall be in accordance with the requirements of 10 CFR 50.65. The licensee should assess and manage the increase in risk that may result from the proposed maintenance activities."	10CFR50.65 is applicable during all conditions of plant operation, including normal shutdown operations. 10CFR50.65(a)(4) requires licensees to assess and manage the increase in risk that may result from proposed maintenance activities. Compliance with this regulation ensures that risk will be assessed and managed during the proposed endstate, and no commitment is required to comply with 10CFR50.65.	Agree. Change made.
174	P54/L6 and L7	Revise:	Editorial	Agree with alternate resolution.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		"The availability of the TDAFW pump must be assured while the plant is operating in Mode 4 in accordance with the assumptions of the TR." To: "The availability of the TDAFW pump must be assured while the plant remains in Mode 4 in accordance with the assumption of the TR."		The following sentence was revised to the following: The availability of the TDAFW pump is assured while the plant remains in Mode 4 in accordance with the assumption of the TR.
175	P54/L10-L13	Delete: "Any plant-specific license amendment request (LAR) which deviates from the NRC-approved TR, will require additional NRC review. The licensee should document its basis for deviating from the approved TR in its LAR to support a detailed review by the NRC technical branches involved in the TR review."	This is not a commitment, and is more appropriate to be included in CLIIP Notice of Availability for these proposed changes.	Agree. Change made.
176	P54/L17	Revise: "The NRC staff's evaluation approves only the operation as described in References 5 and 6."	This is a more accurate statement of the changes approved in the SE.	Agree with alternate resolution. This sentence was revised to the following: The NRC staff's evaluation approves only the endstate changes specifically identified in Section 3.1 of this SE.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		To: The NRC staff's evaluation approves only the endstate changes described in Section 3.1 of this SE.		
177	P54/L21-L29	Revise: "The NRC staff's review was based, in part, on the assumption that while remaining at Mode 4, that plants would be at the lower end of RCS pressure range so that it will reduce the potential for LOCA and will limit coolant inventory loss in the event of a LOCA, as described further in Sections 3.1.4 and 3.1.5 of this SE. For mitigation of several events during potential high risk plant configuration in Mode 4, there should be heightened awareness among plant operators that some protection features are not fully operational. In addition, during potential high risk configuration in Mode 4 where Safety Injection is not	Except for providing information regarding the range of RCS pressures in Mode 4, WCAP-16294 does not include any guidance, requirements, or assumptions regarding what RCS pressure should maintained while a plant is in Mode 4. The risk evaluations performed in WCAP-16294 did not include requirements for heightened operator awareness, additional equipment monitoring, additional administrative controls (except for the TDAFW AFW pump), new or different procedures or training. Mode 4 operation is not new or different because of the changes proposed in WCAP-16294. Appropriate procedures and operator training currently exist for Mode 4 operation.	Agree with alternate resolution. These sentences have been revised to the following: The NRC staff's review is based on the knowledge of lower RCS pressure in Mode 4, which reduces the severity of a LOCA, and limits any coolant inventory loss in the event of a LOCA. The staff reiterates that there are high risk plant evolutions and configurations in Mode 4 and that plant operators should be in a heighten state of awareness during Mode 4 status.

Comment Number	Draft SE Page & Line Number	NEI Comment	NEI Justification	NRC Resolution (Agree, Agree with Alternate Resolution, Disagree)
		available, the plant should be closely monitored using appropriate administrative controls, guidance, appropriate plant procedures, and proper staff training." To: "The NRC staff's review was based, in part, on the assumption that while remaining in Mode 4, the TDAFW pump would be available."		
178	P54/L31	Revise: "...changes to the NUREG-1431..." To: "...changes to NUREG-1431..."	Editorial	Agree. Change made.
179	P54/L34	Delete the word "industry"	The TSTF does not need to be identified as the industry TSTF.	Agree. Change made.
180	P54/L36 and L37	Delete: "...for incorporating the TS and Bases changes in the next revision of NUREG-1431."	It has not been determined whether these changes will be included in Rev. 4 of NUREG-1431.	Agree. Change made.

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PWR Owners Group

Member Participation* for MUHP-3015 and PA-LSC-0194

Utility Member	Plant Site(s)	Participant	
		Yes	**No**
AmerenUE	Callaway (W)	X	
American Electric Power	D.C. Cook 1&2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		X
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)		X
Constellation Energy Group	Ginna (W)	X	
Dominion Connecticut	Millstone 2 (CE)		X
Dominion Connecticut	Millstone 3 (W)	X	
Dominion Kewaunee	Kewaunee (W)	X	
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)	X	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W)	X	
Duke Energy	Oconee 1, 2, 3 (B&W)		X
Entergy	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 2, Waterford 3 (CE), Arkansas 1 (B&W)		X
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W)	X	
	TMI 1 (B&W)		X
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W)	X	
	Davis-Besse (B&W)		X
Florida Power & Light Group	St. Lucie 1 & 2 (CE)		X
	Turkey Point 3 & 4, Seabrook (W), Pt. Beach 1&2 (W)	X	
Luminant Power	Comanche Peak 1 & 2 (W)	X	
Xcel Energy	Prairie Island 1&2 (W)	X	
Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
Progress Energy	Robinson 2, Shearon Harris (W),	X	
	Crystal River 3 (B&W)		X
PSEG – Nuclear	Salem 1 & 2 (W)	X	
Southern California Edison	SONGS 2 & 3 (CE)		X
South Carolina Electric & Gas	V.C. Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	X	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	

***Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.**

PWR Owners Group

International Member Participation* for MUHP-3015 and PA-LSC-0194

Utility Member	Plant Site(s)	Participant Yes	Participant No
British Energy	Sizewell B	X	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	X	
Hokkaido	Tomari 1 & 2 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 1, Ohi 1 & 2, Takahama 1 (W)	X	
	Mihama 2 & 3, Ohi 3 & 4, Takahama 2, 3 & 4 (MHI)		X
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)	X	
	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 , 5 & 6 (CE)		X
Kyushu	Genkai 1, 2, 3 & 4, Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)	X	
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)	X	
Ringhals AB	Ringhals 2, 3 & 4 (W)	X	
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2 (W)	X	
Taiwan Power Co.	Maanshan 1 & 2 (W)	X	
Electricite de France	54 Units	X	

***This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending documents to participants not listed above.**

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LIST OF ACRONYMS

AFW	Auxiliary Feedwater
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDP	Core Damage Probability
CI	Containment Isolation
CREATCS	Control Room Emergency Air Temperature Control System
CREFS	Control Room Emergency Filtration System
CRMP	Configuration Risk Management Program
CS	Containment Spray
CVCS	Chemical and Volume Control System
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators
EFW	Emergency Feedwater
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FBACS	Fuel Building Air Cleanup System
FW	Feedwater
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
ICS	Iodine Cleanup System
IE	Initiating Event
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LERP	Large Early Release Probability
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
MD	Motor-driven
MFW	Main Feedwater
NEI	Nuclear Energy Institute
NSSS	Nuclear Steam Supply System
PIV	Pressure Isolation Valve
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PREACS	Pump Room Exhaust Air Cleanup System (or Penetration Room Exhaust Air Cleanup System)
PWROG	Pressurized Water Reactor Owners Group
QS	Quench Spray
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RS	Recirculation Spray

LIST OF ACRONYMS (cont.)

RWST	Refueling Water Storage Tank
SBACS	Shield Building Air Cleanup System
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SSPS	Solid State Protection System
SW	Service Water
SWS	Service Water System
TD	Turbine-driven
UHS	Ultimate Heat Sink
WOG	Westinghouse Owners Group

IDENTIFICATION OF REVISIONS

This WCAP revision (Revision 1) incorporates PWROG responses provided to the NRC as documented in the following Nuclear Energy Institute letters (see Appendix C through Appendix F for copies):

- NEI letter, Subject: Response to NRC Request for Additional Information Regarding PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134), dated December 12, 2007.

- NEI letter, Subject: Response to NRC Request for Additional Information Regarding PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134), dated November 26, 2008.

- NEI letter, Subject: Draft Revisions to PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134), dated March 13. 2009.

- NEI letter, Subject: Response to a Request for Additional Information (RAI) on Technical Specifications and Bases Associated with PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134), dated September 20, 2009 (note that the date on the letter contained a typographical error and should have been November 20, 2009, the date of the actual transmittal).

The following provides a summary of the revisions to the WCAP.

IDENTIFICATION OF REVISIONS		
Revision	**Location**	**Description**
0	NA	Original Issue
1		Added SE from NRC.
1	Executive Summary	Updated Owners Group name.
1	Section 1	Updated Owners Group name.
1	Table 2-1	Deleted Technical Specifications 3.6.1-B, 3.6.2-D, 3.6.3-F, 3.6.4A-B, 3.6.4B-B, 3.6.5A-B, 3.6.5B-B, 3.6.5C-B, 3.6.8-B, 3.6.15-B, 3.6.16-D, 3.16.17-C.
1	Section 3	Changed transition time for Mode 4 to Mode 5 during shutdown and for Mode 5 to Mode 4 during startup from 70 hours to 24 hours.
1	Table 6-1	Changed Secondary Side Pressure for TS Mode 4 to Mode 3 from "Normal operating pressure" to "Less than normal operating pressure".
1	Table 6-2	Changed Auxiliary Feedwater System Status for Mode 6 and Mode 5 from "Out of service" to "Not required".

IDENTIFICATION OF REVISIONS		
Revision	**Location**	**Description**
1	Table 6-3	Changed Secondary Side Status for Target Conditions for Mode 4 from "Normal operating pressure" to "Less than normal operating pressure".
1	Table 6-4	Changed Secondary Side status for POS 3 from "Normal operating pressure" to "Less than normal operating pressure".
1	Table 6-5	Changed Secondary Side status for POS 5 from "Normal operating pressure" to "Less than normal operating pressure".
1	Section 6.2.2	Changed text to indicate that in POS 3, the secondary side will be less than normal operating pressure rather than near the normal operating pressure.
1	Section 6.3.1	Transition Risk Model – Rewrote the model description.
1	Section 6.3.1.1	For POS 3 and 5 Plant Response Model, changed descriptions for the following initiating events: LLOCA, MLOCA, SLOCA, LOSP, SGTR, and Secondary Side Breaks. For POS 4 Plant Response Model, changed descriptions for the following initiating events: Loss of Decay Heat Removal, LOSP, Cold Overpressurization.
1	Table 6-7	Separated POS 3 into two categories: (1) Shutdown to Mode 5 and (2) Shutdown and Repair in Mode 4; provided justification for the new category.
1	Table 6-8	Added new column with data for new POS 3 category (Shutdown and Repair in Mode 4).
1	Section 6.3.2	Text and Tables 6-9 and 6-10 revised to reflect updated analysis.
1	Section 6.4	Changed text describing the risk calculation results and deleted text indicating that the base case results are included.
1	Section 6.4.1	Changed text in Defense-In-Depth Considerations sections for Function 1.a., 1.b, 2.a, 2.b, 3.a (1), and 3.a (2).
1	Section 6.4.2	Changed text in the Basis for Proposed Change section to clarify what must occur for radiation in the control room to be a concern.
1	Section 6.4.7	Text and Table 6-11 revised to reflect updated analysis. Base case results removed.
1	Section 6.4.8	Text and Table 6-12 revised to reflect updated analysis. Base case results removed.
1	Sections 6.4.9 through 6.4.16, 6.4.23, 6.4.28 through 6.4.30	Deleted text within these sections; section headings maintained to preserve section numbering.
1	Section 6.4.17	Text and Table 6-13 revised to reflect updated analysis. Base case results removed.
	Section 6.4.19	Under Defense-in-Depth Considerations, changed text to indicate that ice condenser is available rather than required to be operable.
1	Sections 6.4.24, 6.4.26, 6.4.27	Under Defense-in-Depth Considerations, changed text to indicate that two trains of containment spray are available rather than required to be operable.

IDENTIFICATION OF REVISIONS		
Revision	**Location**	**Description**
1	Section 6.4.32	Text and Table 6-14 revised to reflect updated analysis. Base case results removed.
1	Section 6.4.33	Text and Table 6-15 revised to reflect updated analysis. Base case results removed.
1	Section 6.4.35	Updated Basis for Proposed Change and Defense-in-Depth Considerations for CREFS.
1	Section 6.4.40	Text and Tables 6-16 to 6-19 revised to reflect updated analysis. Base case results removed.
1	Section 6.4.41	Text and Tables 6-20 to 6-22 revised to reflect updated analysis. Base case results removed.
1	Section 6.4.42	Text and Table 6-23 revised to reflect updated analysis. Base case results removed.
1	Section 6.4.43	Text and Tables 6-24 and 6-25 revised to reflect updated analysis. Base case results removed.
1	Section 6.5.1, Table 6-26	Updated CDP values in table for POS 4 and Totals. Also deleted final row which was TOTAL Excluding POS 4. Revised paragraph following table.
1	Section 6.5.2, Table 6-27	Updated CDP values in Table 6-27. Also deleted final row which was Total Excluding POS 4. Revised paragraph following table.
1	Section 6.5.3	Added new Section 6.5.3, Probability of Power Recovery Sensitivity.
1	Section 6.6	Provided justification for adding a Tier 2 requirement.
1	Section 6.7	Changed text to indicate that more equipment options are available for POS 3 rather than required.
1	Section 7	Added a statement indicating that the sensitivity cases in Section 6.5 also examine a change in the probability of not recovering power for POS 4.
1	Section 8	Updated the section based on the changes made throughout the WCAP.
1	Appendix A	Text and cutsets revised to reflect the updated analyses.

EXECUTIVE SUMMARY

The Pressurized Water Reactor Owners Group (PWROG) has undertaken a risk-informed program to evaluate the endstates that the Technical Specification Actions require the unit to be placed in if the Required Action and associated Completion Time are not met. The Technical Specifications contained in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3, were reviewed to determine the Actions for which changes to the endstates are proposed. The endstates are currently defined based on transitioning the unit to a Mode or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. Mode 5 is the current endstate for LCOs that are applicable in Modes 1 through 4.

The risk of the transition from Mode 1 to Mode 4 or Mode 5 depends on the equipment that is operable. For example, the transition from Mode 4 to Mode 5 can introduce additional risk since it is required to re-align the unit from steam generator cooling to residual heat removal, or shutdown, cooling. During this re-alignment, there is an increased potential for loss of shutdown cooling and loss of inventory events. In addition, decay heat removal following a loss of offsite power event in Mode 5 is dependent on AC power for shutdown cooling, whereas, in Mode 4 the turbine-driven auxiliary feedwater pump will be available. Therefore, transitioning to Mode 5 may not always be the appropriate endstate from a risk perspective.

The purpose of this program is to evaluate and identify the appropriate endstate for a number of Technical Specification Required Actions based on the risk of transitioning the unit from Mode 1 to the lower Modes. Mode 4 is justified as an acceptable alternate endstate to Mode 5.

The proposed changes to the Technical Specification endstates will also reduce the amount of time a unit is shutdown to restore inoperable equipment. The time reduction comes from not requiring a cooldown to Mode 5 and subsequent heat up from Mode 5.

A risk-informed approach, consistent with Regulatory Guides 1.174 and 1.177 (References 1 and 2, respectively) is used in this evaluation. The risk associated with the transition from Mode 1 to Modes 4 and 5, and then returning to Mode 1 operation, is assessed both qualitatively and quantitatively. In addition to assessing the risk impact, the impacts on defense-in-depth and safety margins are also considered.

The Required Actions in the Technical Specifications listed in Table 2-1 are been evaluated for a change in endstate from Mode 5 to Mode 4. The general qualitative evaluation of the plant operating states concludes that there are advantages in risk and in defense-in-depth when the unit remains in Mode 4 rather than continuing to cool down to Mode 5. Technical Specification-specific evaluations demonstrate that there is less risk for the unit if the endstates for these Technical Specifications are changed from Mode 5 to Mode 4. The probabilistic risk assessment model used for the quantitative evaluations is described and shown to be representative of all Westinghouse NSSS units. The results of sensitivity cases support the conclusion that there is less risk associated with a cooldown to Mode 4 than there is for a cooldown to Mode 5. These conclusions are also supported by the Tier 2 assessment and the qualitative assessment of external events.

The evaluations presented and their associated conclusions support changing the endstate from Mode 5 to Mode 4 for the Technical Specifications listed in Table 2-1. Markups of the changes to the NUREG-1431 Technical Specifications and Bases are presented in Appendix B.

1 INTRODUCTION

The Pressurized Water Reactor Owners Group (PWROG) has undertaken a risk-informed program to evaluate the endstates that the Technical Specification Actions require the unit to be placed in if the Required Action and associated Completion Time are not met. The Technical Specifications contained in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3, were reviewed to determine the Actions for which changes to the endstates are proposed. The endstates are currently defined based on transitioning the unit to a Mode or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. Mode 5 is the current endstate for LCOs that are applicable in Modes 1 through 4.

The risk of the transition from Mode 1 to Mode 4 or Mode 5 depends on the equipment that is operable. For example, the transition from Mode 4 to Mode 5 can introduce additional risk since it is required to re-align the unit from steam generator cooling to residual heat removal, or shutdown, cooling. During this re-alignment, there is an increased potential for loss of shutdown cooling and loss of inventory events. In addition, decay heat removal following a loss of offsite power event in Mode 5 is dependent on AC power for shutdown cooling, whereas, in Mode 4 the turbine-driven auxiliary feedwater pump will be available. Therefore, transitioning to Mode 5 may not always be the appropriate endstate from a risk perspective.

The purpose of this program is to evaluate and identify the appropriate endstate for a number of Technical Specification Required Actions based on the risk of transitioning the unit from Mode 1 to the lower Modes. Mode 4 is justified as an acceptable alternate endstate to Mode 5.

The proposed changes to the Technical Specification endstates will also reduce the amount of time a unit is shutdown to restore inoperable equipment. The time reduction comes from not requiring a cooldown to Mode 5 and subsequent heat up from Mode 5.

A risk-informed approach, consistent with Regulatory Guides 1.174 and 1.177 (References 1 and 2, respectively) is used in this evaluation. The risk associated with the transition from Mode 1 to Modes 4 and 5, and then returning to Mode 1 operation, is assessed both qualitatively and quantitatively. In addition to assessing the risk impact, the impacts on defense-in-depth and safety margins are also considered.

2 TECHNICAL SPECIFICATIONS AND CHANGE REQUEST

The Technical Specification Required Action endstates evaluated for the endstate change are contained in NUREG-1431, "Standard Technical Specifications for Westinghouse Plants" (Reference 3). Technical Specification number, title, Condition, current endstate, and the proposed endstate are provided in Table 2-1.

Table 2-1 Proposed Changes to Endstates

Technical Specification/ Condition	Title	Current Endstate	Proposed Endstate
3.3.2-B	ESFAS Instrumentation	5	4
3.3.2-C	ESFAS Instrumentation	5	4
3.3.2-K	ESFAS Instrumentation	5	4
3.3.7-C	Control Room Emergency Filtration System Actuation Instrumentation	5	4
3.3.8-D	Fuel Building Air Cleanup System Actuation Instrumentation	5	4
3.4.13-B	RCS Operational Leakage	5	4
3.4.14-B	RCS Pressure Isolation Valve Leakage	5	4
3.4.15-E	RCS Leakage Detection Instrumentation	5	4
3.5.3-C	Emergency Core Cooling System - Shutdown	5	4
3.5.4-C	Refueling Water Storage Tank	5	4
3.6.6A-B	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6A-E	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6B-F	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6C-B	Containment Spray System (Ice Condenser)	5	4
3.6.6D-B	Quench Spray System (Subatmospheric)	5	4
3.6.6E-F	Recirculation Spray System (subatmospheric)	5	4
3.6.7-B	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)	5	4
3.6.11-B	Iodine Cleanup System (Atmospheric and Subatmospheric)	5	4
3.6.12-B	Vacuum Relief Valves (Atmospheric and Ice Condenser)	5	4

Table 2-1 (cont.)	Proposed Changes to Endstates		
Technical Specification/ Condition	**Title**	**Current Endstate**	**Proposed Endstate**
3.6.13-B	Shield Building Air Cleanup System (Dual and Ice Condenser)	5	4
3.6.14-B	Air Return System (Ice Condenser)	5	4
3.6.18-C	Containment Recirculation Drains (Ice Condenser)	5	4
3.7.7-B	Component Cooling Water System	5	4
3.7.8-B	Service Water System	5	4
3.7.9-C	Ultimate Heat Sink	5	4
3.7.10-C	Control Room Emergency Filtration System	5	4
3.7.11-B	Control Room Emergency Air Temperature Control System	5	4
3.7.12-C	ECCS Pump Room Exhaust Air Cleanup System	5	4
3.7.13-C	Fuel Building Air Cleanup System	5	4
3.7.14-C	Penetration Room Exhaust Air Cleanup System	5	4
3.8.1-G	AC Sources – Operating	5	4
3.8.4-D	DC Sources – Operating	5	4
3.8.7-B	Inverters – Operating	5	4
3.8.9-D	Distribution Systems – Operating	5	4

3 NEED FOR TECHNICAL SPECIFICATION CHANGE

As discussed in Regulatory Guide 1.177 acceptable reasons for requesting Technical Specification changes fall into one or more of the following categories:

- Improvement to Operational Safety: A change to the Technical Specifications can be made due to reductions in the plant risk or a reduction in the occupational exposure of plant personnel in complying with the Technical Specification requirements.

- Consistency with Risk Basis in Regulatory Requirements: Technical Specification requirements can be changed to reflect improved design features in a plant or to reflect equipment reliability improvements that make a previous requirement unnecessarily stringent or ineffective. Technical Specifications may be changed to establish consistently based requirements across the industry or across an industry group.

- Reduce Unnecessary Burdens: The change may be requested to reduce unnecessary burdens in complying with current Technical Specification requirements, based on operating history of the plant or industry in general. This includes extending completion times 1) that are too short to complete repairs when components fail with the plant at-power, 2) to complete additional maintenance activities at-power to reduce plant down time, and 3) to provide increased flexibility to plant operators.

The benefits of changing the Technical Specification Required Action endstates are related primarily to the first two categories.

With regard to operational safety, the risk of the transition from Mode 1 to Mode 4 is lower than the risk of the transition from Mode 1 to Mode 5. The additional mode transition (Mode 4 to Mode 5) involves re-aligning the unit from steam generator cooling to shutdown cooling (residual heat removal (RHR)). This activity requires system alignment changes that can lead to loss of inventory events and loss of shutdown cooling. In addition, in Mode 4, as opposed to Mode 5, additional systems are available for event mitigation that provide a reduced risk once the unit has transitioned to the required endstate. As an example, for a loss of offsite power/station blackout (LOSP/SBO) event, the turbine driven auxiliary feedwater pumps will be available for decay heat removal in Mode 4. In Mode 5, this capability is not available.

Changing the required endstate will also result in increasing unit availability by decreasing the time shutdown. The additional time required to transition to Mode 5 from Mode 4 when shutting down and also to Mode 4 from Mode 5 when restarting can be eliminated with the endstate change. As noted in Section 6.3.1.2, a typical time for the transition from Mode 4 to Mode 5 during shutdown and from Mode 5 to 4 during startup is 24 hours. This change will allow a time reduction of 24 hours.

4 DESIGN BASES REQUIREMENTS AND IMPACT

The requested change to the Technical Specification Required Action endstate does not impact the design basis for any unit. As discussed in Sections 5.1 and 5.2 defense-in-depth and safety margins will not be impacted. The FSAR accident analyses, events considered and input assumptions will not be changed.

The current Required Action for the unit to be in Mode 5 in 36 hours if inoperable equipment is not restored to operable status, or the Required Actions and associated Completion Times are not met, is being revised to be in Mode 4 in 12 hours.

5 IMPACT ON DEFENSE-IN-DEPTH AND SAFETY MARGINS

In addition to discussing the impact of the changes on plant risk, as presented in Section 6, the traditional engineering considerations also need to be addressed. These include defense-in-depth and safety margins. The fundamental safety principles on which the plant design is based cannot be compromised. Design basis accidents are used to develop the plant design. These are a combination of postulated challenges and failure events that are used in the plant design to demonstrate a safe plant response. Defense-in-depth and adequate safety margins may be impacted by the proposed change and consideration needs to be given to these elements.

5.1 DEFENSE-IN-DEPTH

The proposed change needs to meet the defense-in-depth principle that consists of a number of elements. These elements follow:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design.
- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.
- Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are maintained.
- The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

Operation in Mode 4 offers additional system availability over Mode 5. The additional systems, that offer increased defense-in-depth, are available to mitigate events that can occur. For example:

- <u>Core Cooling</u>: In Mode 4, core cooling can be maintained by the SGs in conjunction with AFW, and the RHR system can provide a backup function. In Mode 5, core cooling is dependent on the RHR system and AFW is not required to be operable.
- <u>Inventory Makeup</u>: In Mode 4, one train of ECCS is available to mitigate loss of coolant events. The ECCS is not required to be operable in Mode 5. Mitigation of loss of coolant events is dependent on the availability of a high head injection pump/train.
- <u>Electrical Power Sources</u>: In Mode 4, all emergency DGs are operable, while in Mode 5 not all are required to be operable, depending on the other equipment required to be operable.

In addition, the endstate changes eliminate the need to transition from Mode 4 to Mode 5. This transition requires re-alignment of systems to transfer to shutdown cooling, which is cooling provided by the RHR system for the Westinghouse NSSS plants. This transition can lead to an increased probability of loss of shutdown cooling or loss of inventory events. Remaining in Mode 4 eliminates the need to make this transition.

Therefore, the system redundancy, dependence, and diversity are increased by remaining in Mode 4. The reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is not impacted since there is no change in plant configuration, design, or operation in the Modes of interest. In addition, the other elements of defense-in-depth are not impacted by this change since the plant configuration (systems, structures, and components) and operation within each Mode remains the same.

The impact on defense-in-depth is further discussed in each individual requested endstate change in Section 6.

5.2 IMPACT ON SAFETY MARGINS

The safety analysis acceptance criteria in the licensing basis is not impacted by this change. The codes and standards approved for use by the NRC continue to be met. Availability of redundant and diverse systems will be maintained or improved with the proposed endstate changes. The proposed changes will not allow plant operation in a configuration outside the design basis. For the proposed changes, Mode 4 offers additional defense-in-depth and a lower risk level, for the requested changes, than Mode 5. There is no impact on safety margins.

6 ASSESSMENT OF THE IMPACT OF THE ENDSTATE CHANGE

This section provides the discussion of the impact of each proposed endstate change on defense-in-depth, safety margins, and risk. The risk impact is discussed qualitatively and quantitatively. The quantitative risk discussion is primarily directed at the impact on core damage frequency. The impact with respect to large early release frequency is provided qualitatively. Also included in this section is a discussion of the transition risk model used in the quantitative evaluations.

The evaluation was completed consistent with the requirements in Regulatory Guides 1.174 and 1.177.

6.1 APPROACH TO THE EVALUATION

The evaluation supporting these Technical Specification changes is divided into three parts: defense-in-depth, safety margins, and risk assessment. The impact on defense-in-depth is discussed, in general terms, in Section 5.1. The impact on safety margins is discussed, in general terms, in Section 5.2. The specific impact of each proposed change is discussed, with regard to defense-in-depth, in Section 6.4.

The risk assessment supporting these changes is divided into two parts. The first part discusses the impact of the changes qualitatively and the second assesses the impact from the quantitative perspective. The qualitative and quantitative risk impacts are discussed in general terms in Sections 6.2 and 6.3. The specific impact of each proposed change is discussed in Section 6.4.

The Technical Specification change being considered is the Technical Specification Required Action endstate. A change from a Mode 5 endstate to a Mode 4 endstate is justified for a number of Technical Specifications. Each proposed endstate change listed in Table 2-1 is evaluated and discussed separately in Section 6.4.

The analysis approach identifies the appropriate endstate from a risk perspective, that is, which endstate is associated with the lowest risk. The evaluation considers the risk associated with:

- The transition from operating in Mode 1 at 100% power to Mode 4 or Mode 5 with the inoperable equipment assumed to be unavailable.
- Operation in the endstate of interest with the inoperable equipment assumed to be unavailable.
- The transition from Mode 4 or Mode 5 to Mode 1 at 100% power with all equipment operable.

To perform the quantitative evaluation a transition risk model is used. This model has the capability to evaluate the risk of changing plant operating modes during plant shutdowns and startups. Transitions between modes is specifically modeled. The risk analysis considers more than design basis events. The model is discussed in Section 6.3. Core damage probability (CDP) is the risk metric used in this model. Large early release probability (LERP) is not addressed quantitatively. A qualitative evaluation is performed for LERP.

6.2 QUALITATIVE RISK ASSESSMENT

The qualitative risk assessment is directed at the risk associated with plant operation in Mode 4 and Mode 5, and transitioning to and from these modes. Consistent with the Technical Specification mode definitions, this evaluation assumes that the unit has entered the mode of interest when the conditions associated with the upper end of mode are met. The following sections provide a qualitative assessment of the risk associated with transitioning a plant to, from, and operating in Modes 4 and 5. All equipment is assumed to be available, unless operating procedures direct that equipment be isolated or locked-out. In Section 6.4, the impact of inoperable equipment is addressed.

6.2.1 Mode Transition Risk Assessment

The following provides the general steps to shutting the plant down (to Mode 5). Plants that are shutting down as required by the Technical Specification or for a refueling or other forced outage will generally be starting at or near 100% power (Mode 1).

- The power reduction will start and eventually the feedwater supply will be switched from the main feedwater system to the auxiliary or startup feedwater system, and the turbine will be tripped. These plant configuration changes provide for an increased probability of loss of feedwater flow to the steam generators. After transitioning to AFW or the startup feedwater system and reducing the power level to less than 5%, the plant will be in Mode 2.

- The plant power level will continue to be reduced to 0% with AFW used to remove decay heat. The reactor coolant system (RCS) temperature remains at ~557°F, or 547°F depending on the no load T_{avg}, and the RCS pressure is at ~2235 psig with the secondary side at normal operating pressure. Mode 3 (hot standby) is achieved when the power level is 0%.

- The next step is to begin RCS cooldown. The shutdown margin (boration) is established and the reactor trip breakers are opened. The number of operating reactor coolant pumps (RCPs) may be reduced. Cooldown is initiated via the steam dump system or SG atmospheric relief valves. The transition to Mode 4 (hot shutdown) occurs when the RCS temperature is less than 350°F.

- The cooldown continues and the safety injection pumps are disabled. RCS cooling can be switched to the RHR system when the RCS temperature is less than 340°F and the RCS pressure is less than 365 psig. The number of operating RCPs can be reduced to one. The switch to Mode 5 (cold shutdown) occurs when the RCS temperature is less than 200°F.

To return to power, similar steps are required in the reverse order. Table 6-1 provides a summary of the important RCS parameters for the different mode transitions. This table also provides the Technical Specification temperatures and power levels specified for the different Modes.

Table 6-1 Key Plant Parameters by Technical Specification Mode					
Parameter	**Mode 5 to Mode 4**	**Mode 4 to Mode 3**	**Mode 3 to Mode 2**	**Mode 2 to Mode 1**	**Mode 1**
Technical Specification RCS Average Temperature	≤200°F (Mode 5, Cold Shutdown)	>200°F to <350°F (Mode 4, Hot Shutdown)	≥350°F (Mode 3, Hot Standby)	NA (Mode 2, Startup)	NA (Mode 1, Power)
Technical Specification Reactor Power Level	NA (Mode 5)	NA (Mode 4)	NA (Mode 3)	≤5% (Mode 2)	>5% (Mode 1)
RCS Average Temperature	~185°F to ~330°F	~330°F to ~557°F	~557°F	~557°F	~557°F
RCS Pressure	~340 psig	~340 psig to ~2235 psig	~2235 psig	~2235 psig	~2235 psig
Pressurizer Status	Water solid to bubble	Bubble	Bubble	Bubble	Bubble
Secondary Side Pressure	0 psig	Less than normal operating pressure	Normal operating pressure	Normal operating pressure	Normal operating pressure

The following discussion centers around several plant operating states (POS). The POS approach is used as the basis for the qualitative discussion to be consistent with the quantitative transition PRA model, that is discussed in Section 6.3.1. A POS is a unique plant configuration defined by a set of parameters. For each POS, a unique set of initiating events, plant conditions, and systems available for event mitigation can be identified. Very often a plant will be in a POS for a relatively short period of time, because switching the plant configuration (system re-alignments) is required to reach the desired endstate. To identify the POSs, an understanding of the typical key activities that are in progress as the plant shuts down and restarts is required. The following provides a general summary of these activities:

Modes 1-2

- Decrease power (Mode 1)
- Take the turbine off-line (Mode 1)
- Transfer from MFW (main feedwater) to AFW (note that some plants may continue on MFW depending on their MFW design and approach to plant shutdown) (Mode 1)
- Mode 2 (startup) when power level is ≤ 5%

Modes 2-3

- Reduce power to 0% (Mode 2 to Mode 3)
- Insert control banks (Mode 2)
- Mode 3 (hot standby) when power level is 0% (T_{ave} = 557°F, pressurizer pressure = 2235 psig, pressurizer level = 25% to 40%, SG levels = 65%)

Modes 3-4 (upper part of Mode 4 on AFW, then transfer to RHR)

- Borate the RCS to the required boron concentration to satisfy the shutdown margin requirement (Mode 3)
- If running and not required to feed the SGs, stop the turbine-driven AFW pump (Mode 3)
- If running, shut down the MFW pumps (Mode 3)
- Maintain SG water level (Mode 3)
- If withdrawn, insert the shutdown banks (Mode 3)
- Open the reactor trip breakers (Mode 3)
- Reduce the number of operating RCPs (Mode 3)
- At approximately 2000 psig, block safety injection and steamline isolation (Mode 3)
- At approximately 950 psig, isolate the accumulators (Mode 3)
- If the secondary side is to be cooled down, start "Secondary Plant Shutdown" (Mode 3)
- Place cold overpressure protection in service prior to reaching 350°F (or plant-specific temperature) (Mode 3)
- RCS cooldown is controlled by the condenser steam dump valve and SG atmospheric relief valves
- Decrease the RCS temperature from ~557°F to ~330°F (transition to Mode 4 occurs when the RCS temperature drops below 350°F)
- Decrease the RCS pressure from ~2235 psig to ~340 psig
- When the RCS pressure is less than 365 psig and the RCS temperature is less than 340°F, place one RHR loop in operation (Mode 4)

Modes 4-5 (on RHR system)

- Disable the safety injection pumps (Mode 4)
- Defeat the AFW actuation system (Mode 4)
- Reduce the operating RCPs to one (Mode 4)
- Cooldown the RCS using the RHR system
- Maintain the pressurizer pressure at 250 psig
- At 200°F, Mode 5 (Cold Shutdown) is entered
- Continue to cooldown to 190°F to 170°F
- End state: A bubble in the pressurizer, the RCS temperature is between 190°F and 170°F, the RCS pressure is 250 psig
- Centrifugal charging pumps (CCP) are in standby (Mode 5) (one CCP operating if a RCP is running)
- Solid state protection system (SSPS) is in service only for certain functions depending on whether the control rods are capable of withdrawing (Mode 5)

Modes 5-4 (on RHR system)

- CCPs are in standby (Mode 5) (one CCP operating if a RCP is running)
- SSPS is in service (Mode 5)
- Increase the RCS temperature from ~185°F to ~330°F (transition to Mode 4 occurs when RCS temperature exceeds 200°F)
- Verify that AFW is aligned for startup (Mode 4)
- Maintain the RCS pressure at ~340 psig
- RCS cooling by RHR (Mode 5 and lower end of Mode 4)
- Cold overpressure protection in service (Mode 5 and lower end of Mode 4)

Modes 4-3 (lower end of Mode 4 on RHR, then transfer to AFW)

- Prepare the SG for startup (Mode 4)
- AFW actuation signals and AFW components are available for automatic actuation (Mode 4)
- Place the RHR system in standby (lower end of Mode 4)
- Block the cold overpressure protection system (Mode 4)
- Initiate AFW (note that at some plants, a startup feedwater pump or condensate pumps and MFW may be used for startup instead of AFW) (Mode 4)
- Increase the RCS temperature from ~330°F to ~557°F (transition to Mode 3 occurs when the RCS temperature is ≥ 350°F)
- Increase the RCS pressure from ~340 psig to ~2235 psig
- Start the remaining RCPs (Mode 3)
- Verify pressurizer (PZR) pressure safety injection (SI) and steamline pressure SI and steamline isolation (SLI) auto reset (Mode 3)
- RCS heatup is controlled by the condenser steam dump valves and the SG atmospheric relief valves

Modes 3-2

- Close the reactor trip breakers (Mode 3)
- Withdraw shutdown and control banks (Mode 3)
- Raise power to less than 5% (Mode 3 to Mode 2)

Modes 2-1

- Transfer from AFW to MFW (note that some plants may already be on MFW depending on their MFW design and approach to plant startup) (Mode 1)
- Increase power (Mode 1)
- Bring turbine on-line (Mode 1)

Table 6-2 provides a summary of system status by the Technical Specification mode. Table 6-3 provides the target plant conditions when shutting down as required by the Technical Specifications. The target conditions of interest are for Modes 4 and 5. As noted previously, it is assumed that the plant will stop the shutdown after entering the mode required by the Technical Specifications.

Each POS is defined based on the plant state, available equipment, and potential initiating events. Table 6-4 defines the POSs for the transition from power operation to cold shutdown (Mode 5). Table 6-5 defines the POSs for the transition from cold shutdown (Mode 5) to power operation. These plant operating states are based on the previously discussed information and each POS represents a unique plant configuration that is defined by the plant conditions and parameters provided on these tables.

Table 6-2 System Status by Technical Specification Mode

System	Mode 6	Mode 5	Mode 4	Mode 3	Mode 2	Mode 1
RCS Charging and Letdown[1]	Establish function	In service	In service	In service	In service	In service
Reactor Coolant Pumps	None running	As needed for plant heatup	As needed for plant heatup	All RCPs running	All RCPs running	All RCPs running
Reactor Trip Breakers	Open	Open	Open	Open/Closed	Closed	Closed
Residual Heat Removal	In service	In service	In service or in standby	Standby	Standby	Standby
Auxiliary Feedwater	Not required	Not required	Aligned for startup or in service	In service	In service	In service and then standby after switch to MFW
High Head Injection[1]	Pull to lock	Pull to lock when water solid, standby with bubble	Standby	Standby	Standby	Standby
Cold Overpressure Protection	Establish function	In service	In service[2]	Not required	Not required	Not required
High Flux at Shutdown Alarm (HFASA)	In service	In service	In service	In service	Not required	Not required
Source Range	Two channels in service	Two channels in service	Two channels in service	Two channels in service	Two channels in service below P-6	Not required
Intermediate Range	Not required	Not required	Not required	Not required	Two channels in service	Two channels in service below P-10
Power Range	Not required	Not required	Not required	Not required	Required	Required
Solid State Protection System	Not required	Not required	In service	In service	In service	In service
Emergency Diesel Generators	Less than full complement[3]	Less than full complement[3]	Full complement	Full complement	Full complement	Full complement

Notes:

1. One charging pump is operating to provide RCS charging in Modes 1-6.
2. Cold overpressurization is required in the lower part of Mode 4.
3. Depending on equipment required to be operable.

Table 6-3 Important Parameters for Mode Target Conditions

Parameter	Mode 1	Mode 2	Mode 3	Mode 4	Mode 5
RCS Temperature	557°F	557°F	557°F	~330°F	170°F to 190°F
RCS Pressure	2235 psig	2235 psig	2235 psig	340 psig	250 psig
Secondary Side Status	Normal operating pressure	Normal operating pressure	Normal operating pressure	Less than normal operating pressure	Low pressure
PZR Status	Bubble	Bubble	Bubble	Bubble	Bubble
(Decay) Heat Removal Mode	MFW	AFW	AFW	AFW	RHR
Power Level	100%	5%	0%	0%	0%

Table 6-4 Plant Operating States (Power Operation to Cold Shutdown)

State	POS 1	POS 2	POS 3	POS 4
Plant Mode	1 (transition only) 2 3 (upper part)	3 (middle part)	3 (lower part) 4 (upper part)	4 (lower part) 5 (upper part)
RCS Temperature	557°F	557°F to XX°F[1]	XX°F[1] to 340°F	340°F to 180°F
RCS Pressure	2235 psig	2235 psig to 950 psig	950 psig to 365 psig	365 psig to 250 psig
Pressurizer	Bubble	Bubble	Bubble	Bubble
Secondary Side	Normal operating pressure	Normal operating pressure	Less than normal operating pressure	Low pressure (shutdown)
Activities	• AFW for decay heat removal • Reduce power • Switch from MFW to AFW • Borate • Insert control rods • Take turbine off-line	• AFW for decay heat removal and cooldown • Open trip breakers • Reduced operating RCPs • Block SI and SLI • RCS cooldown	• AFW for decay heat removal and cooldown • Reduced operating RCPs • Isolate accumulators • RCS cooldown • Start secondary side cooldown	• RHR for decay heat removal and cooldown • Switch to RHR cooling • Disable SI pumps • Defeat AFW start signals • Cold overpressure protection (COP) in service
System Status	• AFW operating • All systems available	• AFW operating • All systems available • SI and SLI signals blocked • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Accumulators isolated • Reactor trip breakers open	• RHR operating • SI, SLI, and AFW signals blocked • Accumulators isolated • SI pumps disabled • COP in service • Reactor trip breakers open

Note:

1. A defined temperature is not important to this analysis.

Table 6-5 Plant Operating States (Cold Shutdown to Power Operation)

State	POS 4	POS 5	POS 6	POS 7
Plant Mode	4 (lower part) 5 (upper part)	3 (lower part) 4 (upper part)	3 (middle part)	1 (transition only) 2 3 (upper part)
RCS Temperature	180°F to 340°F	340°F to XX°F[1]	XX°F[1] to 557°F	557°F
RCS Pressure	250 psig to 365 psig	365 psig to 950 psig	950 psig to 2235 psig	2235 psig
Pressurizer	Bubble	Bubble	Bubble	Bubble
Secondary Side	Low pressure (shutdown)	Less than normal operating pressure	Normal operating pressure	Normal operating pressure
Activities	• RHR for decay heat removal • Switch to AFW cooling • Establish AFW actuation signals • RCS heatup	• AFW for decay heat removal • One RCP running[2] • RCS heatup • Start secondary side heatup	• AFW for decay heat removal • One RCP running[2] • RCS heatup • Establish SI and SLI signals • Un-isolate accumulators	• AFW for decay heat removal • Switch from AFW to MFW • Withdraw shutdown and control rods • Bring turbine on-line • Close trip breakers • All RCPs running • Increase power
System Status	• RHR operating • SI, SLI, and AFW signals blocked • Accumulators isolated • SI pumps disabled • COP in service • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Accumulators isolated • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Reactor trip breakers open	• AFW to MFW • All systems available

Notes:

1. A defined temperature is not important to this analysis.
2. If the rods are not capable of withdrawal.

Table 6-6 lists the possible internal initiating events for each POS. The following discusses this information. These notes are also provided with Table 6-6.

1. RCS pressure is much lower in POS 3, 4, and 5 than in POS 1, 2, 6, and 7. Large and medium LOCAs are considered in POS 3 and 5, but at a reduced frequency. Large and medium LOCAs are not considered in POS 4.

2. Small LOCAs in POS 1, 2, 6, and 7 are due to random pipe breaks and random RCP seal failures. Small LOCAs in POS 3 and 5 are significantly reduced due to reduced RCS pressure and temperature. Small consequential LOCAs can also occur in POS 1 and 7 due to transient events that lead to the opening of PZR PORVs with failure of the PORVs to reseat. These are not considered in any other POS due to the low plant power level. In POS 4, LOCAs are due to alignment issues and open valves, not pipe breaks or random failures of RCP seals, and are referred to as loss of inventory events.

3. SGTRs are considered in POS 3 and 5, assuming the at-power frequency based on NUREG/CR-6144 (Reference 4), Section 4.7. SGTRs are not considered in POS 4 because the pressure difference across the tubes is much lower and the steam generators are not being used for RCS cooling. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.

4. Secondary side breaks are not considered in POS 4 since the secondary side pressure is much lower than in POS 1, 2, 3, 5, 6, and 7.

5. RCP seal LOCAs due to loss of seal cooling are not considered in POS 3, 4, and 5 since the RCS pressure and temperature are much lower than in POS 1, 2, 6, and 7. In addition, for POS 3, 4, and 5, there is a minimum of 50°F subcooling, which means that a RCP seal pop-open event is not an issue. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.

6. Rod withdrawal is not considered in POS 2, 3, 4, 5, and 6 since the reactor trip breakers are open.

7. Loss of decay heat removal is due to a loss of AFW in POS 1, 2, 3, 5, 6, and 7 and a loss of RHR in POS 4.

8. With regard to loss of feedwater control, there is an increased probability for loss of feedwater due to feedwater control problems related to the switch from MFW to AFW during the transition down in power and related to the switch from AFW to MFW during the transition up in power. This is only applicable in POS 1 and 7.

Table 6-6 Initiating Events by Plant Operating State

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 5	POS 6	POS 7
Large LOCA[1]	X	X	X		X	X	X
Medium LOCA[1]	X	X	X		X	X	X
Small LOCA[2]	X	X	X		X	X	X
Loss of Inventory[2]				X			
RCP Seal LOCAs (Loss of Seal Cooling)[5]	X	X				X	X
Loss of Feedwater Control[8]	X						X
Loss of Decay Heat Removal[7]	X	X	X	X	X	X	X
Loss of Offsite Power	X	X	X	X	X	X	X
Cold Overpressurization				X			
SG Tube Rupture[3]	X	X	X		X	X	X
Secondary Side Breaks[4]	X	X	X		X	X	X
ATWS							
Boron Dilution	X			X			X
Rod Withdrawal[6]	X						X

Notes:

1. RCS pressure is much lower in POS 3, 4, and 5 than in POS 1, 2, 6, and 7. Large and medium LOCAs are considered in POS 3 and 5, but at a reduced frequency. Large and medium LOCAs are not considered in POS 4.
2. Small LOCAs in POS 1, 2, 6, and 7 are due to random pipe breaks and random RCP seal failures. Small LOCAs in POS 3 and 5 are significantly reduced due to reduced RCS pressure and temperature. Small consequential LOCAs can also occur in POS 1 and 7 due to transient events that lead to the opening of PZR PORVs with failure of the PORVs to reseat. These are not considered in any other POS due to the low plant power level. In POS 4, LOCAs are due to alignment issues and open valves, not pipe breaks or random failures of RCP seals, and are referred to as loss of inventory events.
3. SGTRs are considered in POS 1, 2, 3, 5, 6, and 7. Even though in POS 3 and POS 5 the delta P across the tubes (P_{RCS} - $P_{secondary\ side}$) is much lower than in POS 1, 2, 6, and 7, the frequency was not reduced. SGTRs are not considered in POS 4 because the pressure difference across the tubes is much lower and the steam generators are not being used for RCS cooling. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.
4. Secondary side breaks are not considered in POS 4 since the secondary side pressure is much lower than in POS 1, 2, 3, 5, 6, and 7.
5. RCP seal LOCAs due to loss of seal cooling are not considered in POS 3, 4, and 5 since the RCS pressure and temperature are much lower than in POS 1, 2, 6, and 7. In addition, for POS 3, 4, and 5, there is a minimum of 50°F subcooling, which means that a RCP seal pop-open event is not an issue. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.
6. Rod withdrawal is not considered in POS 2, 3, 4, 5, and 6 since the reactor trip breakers are open.
7. Loss of decay heat removal is due to a loss of AFW in POS 1, 2, 3, 5, 6, and 7 and a loss of RHR in POS 4.
8. With regard to loss of feedwater control, there is an increased probability for loss of feedwater due to feedwater control problems related to the switch from MFW to AFW during the transition down in power and related to the switch from AFW to MFW during the transition up in power. This is only applicable in POS 1 and 7.
9. Boron dilution events have not been found to be important to plant risk, and are excluded from further consideration in POS 2, 3, 5, and 6.

POS 1 and POS 7

This state is defined as Technical Specification Mode 1 (only for transitioning from power operation to Mode 2), Mode 2, and the upper part of Mode 3. In Mode 1, only the additional risk associated with the transition from power operation to Mode 2 is included in POS 1. This additional risk is due to potential transients cause by feedwater control issues when shutting the plant down. Other risks associated with Mode 1 operation are not considered part of the risk of transitioning to lower modes, but are considered part of the at-power risk.

Mode 2 and the upper part of Mode 3 are very similar except for the power level (5% vs. 0%). All other key plant parameters are the same. The same initiating events are applicable in Modes 2 and the upper part of 3. The ATWS event is not considered since the plant is at very low power and the high RCS pressures cannot be reached. Most transients cannot occur due to equipment status, for example, the turbine and generator are not operating. Since the plant will be shutting down from 100% power, decay heat levels will be high.

The key difference in POS 1 and POS 7 is the decay heat level. With the lower decay heat levels when returning to power, due to the time the reactor was shut down, operators have a longer time to respond to events. In addition, feedwater control concerns and issues bringing the turbine on-line can lead to a different risk level when starting up than when shutting down.

In these POSs, all systems are available for event mitigation, that is, none have been removed from service related to the mode changes.

POS 2 and POS 6

This state is defined as the middle of Mode 3. The RCS pressure and temperature are being reduced, but are still relatively high. Signals for safety injection and steamline isolation are blocked and the reactor trip breakers are open. The same events in POS 1 are applicable except for rod withdrawal. Loss of feedwater control is no longer an issue either.

POS 2 and POS 6 are very similar except for the direction of transition. In POS 2 the plant is transitioning down (toward shutdown) and in POS 6 the plant is transitioning up (returning to power). The key difference in POS 2 and POS 6 is the decay heat level. With the lower decay heat levels when returning to power, due to the additional time being shut down, the operators have a longer time to respond to events.

POS 3 and POS 5

This state is defined as the lower part of Mode 3 and the upper part of Mode 4. The RCS pressure and temperature are significantly reduced from power operation, therefore, many of the events associated with the high RCS pressure (LOCAs/pipe breaks) have a reduced frequency. In addition, accumulators are isolated.

POS 3 and POS 5 are very similar except for the direction of transition. In POS 3 the plant is transitioning down (toward shutdown) and in POS 5 the plant is transitioning up (returning to power).

The key difference in POS 3 and POS 5 is the decay heat level. With the lower decay heat levels when returning to power, due to the additional time being shut down, the operators have a longer time to respond to events.

POS 4

This state is defined as the lower part of Mode 4 and the upper part of Mode 5. The transition from AFW cooling to RHR cooling occurs in this POS. The RCS pressure and temperature are significantly reduced from power operation, therefore, the LOCA events and SG tube rupture event are no longer applicable. The secondary side pressure is also reduced eliminating the secondary side break events. Loss of inventory related to the RCS cooling switch from AFW to RHR is an event that is added. This can occur when transitioning down or up. Cold overpressurization is also added.

Key Differences Between POS 1 and POS 2 (and POS 7 and POS 6)

- No switchover from MFW to AFW in POS 2
- Reactor trip breakers are open in POS 2
- SI and SLI signals are blocked in POS 2 (operator actuations required for some events)
- Most initiating events are the same, but rod withdrawal and many transient events (such as, loss of MFW and turbine trip) cannot occur in POS 2

Key Differences Between POS 2 and POS 3 (and POS 6 and POS 5)

- Reduced RCS pressure and temperature results in a reduced event frequency for pipe rupture type LOCAs and RCP seal LOCAs in POS 3
- Accumulators are isolated in POS 3

Key Differences Between POS 3 and POS 4 (and POS 5 and POS 4)

- Reduced RCS pressure and temperature in POS 4
- Reduced secondary side pressure and temperature in POS 4
- Loss of inventory and loss of decay heat removal important in POS 4 due to a transfer from AFW/SG cooling to RHR cooling
- Secondary side breaks cannot occur in POS 4 due to a reduced secondary side pressure
- Cold overpressurization needs to be addressed in POS 4

6.2.2 Comparison of Endstates

To achieve Mode 4 as an endstate, the plant will need to transition through POS 1, POS 2, and into POS 3. To achieve Mode 5 as an endstate, the plant will need to transition through POS 1, POS 2, POS 3, and into POS 4. With either endstate, the plant will need to transition through POS 1, POS 2 and into POS 3. To determine the appropriate endstate (Mode 4 vs. Mode 5), the additional risk for the transition through POS 3 and into POS 4 needs to be considered, as well as the risk of remaining in POS 3 (Mode 4) as opposed to POS 4 (Mode 5).

Several of the key differences between POS 3 and 4 are:

- The frequency of loss of decay heat removal is at an increased level with POS 4 as the endstate due to the system re-alignments required. Loss of decay heat removal events in POS 3 can be addressed with AFW (all pumps available) or the RHR system following depressurization of the RCS. In POS 4, the TD AFW pump will not be available to address similar events. In addition, the automatic AFW start signal is available in POS 3, but not POS 4. Therefore, additional options are available in POS 3 for decay heat removal.

- The frequencies of loss of inventory (LOCA) events can be at an increased level with POS 4 as the endstate due to the system re-alignments required. Loss of inventory events in POS 3 can be addressed with the available train of ECCS. In POS 4 a full train of ECCS is not available. The SI pumps are out of service. Inventory control is dependent on the charging system. Therefore, additional options are available in POS 3 for inventory control.

- Mitigation of loss of offsite power (LOSP)/station blackout (SBO) events in POS 3 can be provided by the AFW system including the turbine-driven pump. Availability of the turbine-driven pump is particularly important in case the event degrades to a station blackout. In POS 4, the AFW system turbine-driven pump will not be available for decay heat removal, and the plant will be dependent on restoring electric power to the RHR system. Again, additional options are available in POS 3 for event mitigation.

- The cold overpressurization event needs to be considered in POS 4, but not in POS 3. Although not a large risk contributor, this event is addressed in POS 4.

- Secondary side breaks are considered in POS 3, but not in POS 4. In POS 3 the secondary side will be less than normal operating pressure, but in POS 4 this pressure is greatly reduced, reducing the likelihood of a secondary side break. Secondary side breaks are not typically large contributors to risk, therefore, this assumption for POS 4 has a small risk impact.

- Risk in the shutdown modes is very dependent on electric power availability. There are more required independent sources of electrical power in POS 3 than in POS 4 and there are more potential activities in POS 4 that could cause a loss of offsite power.

- In POS 3, there is more redundancy and diversity of mitigating and support systems required to be available than there is in POS 4.

6.2.3 Containment Considerations

In Modes 1-4, the containment, containment isolation valves, containment sprays, and containment cooling systems are required to be operable. The Bases for the Technical Specifications state that in these modes a design basis accident could cause a release of radioactive material to containment, and an increase in containment pressure and temperature that would require the use of these systems for accident mitigation. The Bases also indicate that in Modes 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature in these modes. Additional operability requirements are specified for Mode 6.

The risk analysis considers more than design basis events. Events can occur in Mode 5 that may lead to core damage and may cause releases outside containment. With the reduced RCS pressure and temperature in Mode 5, as opposed to Modes 1-4, the need for containment cooling systems and containment spray is not as important. Mode 4 does require the containment, containment isolation, and containment cooling and spray systems to be operable.

6.2.4 Conclusions from the Qualitative Risk Assessment

Based on Sections 6.2.3 and 6.2.4, there are advantages, from the risk perspective and also the defense-in-depth perspective, to remain in POS 3 (Mode 4) as opposed to POS 4 (Mode 5) for shutdowns required by the Technical Specifications. In POS 3, the initiators are generally at a reduced probability of occurrence compared to power operation and there are additional systems available for event mitigation compared to POS 4 (Mode 5).

6.3 QUANTITATIVE RISK ASSESSMENT

As with the qualitative risk assessment, the quantitative assessment is also directed at the risk associated with plant operation in Mode 4 and Mode 5, and transitions to and from these modes. Evaluating the risk requires an assessment of the likelihood of applicable initiating events. Consistent with the Technical Specification mode definitions, this evaluation assumes that the unit has entered the mode of interest when conditions associated with the upper end of the mode are met. The following sections provide a quantitative assessment of the risk associated with transitioning a plant to, from, and remaining in Modes 4 and 5. All equipment is assumed to be available, unless operating procedures direct equipment to be isolated or locked-out.

6.3.1 Transition Risk Model

A generic transition risk model that is representative of Westinghouse Nuclear Steam Supply System (NSSS) plants is presented in this section. Only high level information of the model is presented as well as the model quantification results. The risk metric to be determined is core damage probability (CDP). A separate risk model, similar to a level 1, internal event, at-power PRA model, is developed for each POS. Only internal initiating events are included in the transition risk model.

This model is based on the information identified in the qualitative risk discussion in Section 6.2. The POSs are defined on Table 6-4 and 6-5, and the associated initiating events that are addressed are provided on Table 6-6. Equipment availability is discussed in Section 6.2.1.

The plant response or event trees for each initiating event in each POS, the time spent in each POS, the initiating event frequencies, mitigation equipment availability, and human error probabilities for operator actions are developed and discussed in the following sections. This is followed by the model quantification that provides the CDP values for each POS, along with the initiating event CDP distribution for POS 3 and POS 4. This quantification assumes all equipment is available except for equipment removed from service, as the plant transitions though the modes, following plant procedures.

The model is based on a single unit site with standard two train systems. Several key characteristics of this model include:

- AFW system – two MD pumps and one TD pump
- Emergency core cooling system (ECCS) – two train system, each train includes a high pressure and low pressure subsystem
- Reactor protection system (RPS) – SSPS
- Service water (SW) system – two train system
- Component cooling water (CCW) system – two train system
- Electrical power system – a two train system with one emergency diesel generator (EDG) per train

The transition risk PRA model is based on a single unit, therefore, it does not take credit for the availability of shared systems. The AFW system is similar to many Westinghouse NSSS plants. The AFW system does not include a diesel-driven pump that would provide diverse mitigation for loss of offsite power events. Based on the results presented in Table 6-10, the loss of offsite power event is a larger percent contributor to the POS 3 risk than it is to the POS 4 risk and this design difference would be more beneficial for Mode 4 (POS 3).

The ECCS design is also similar to many Westinghouse NSSS plants. The centrifugal charging pumps are used for high pressure safety injection. High pressure safety injection does not include a set of separate safety injection pumps, but the general success criteria of requiring one train is common for Westinghouse NSSS PRA models. Plants with Intermediate Head SI (IHSI) pumps typically require the IHSI pumps to be disabled in Mode 5. The POS 4 PRA model assumes that two trains of charging (SI) pumps are available and the model includes an available swing pump. Therefore, high pressure SI is modeled as being available in the POS 4 PRA model, when it may not be available based on plant procedures. This is a conservative approach because it reduces the risk in Mode 5 when, in fact, the pump may not be available.

Low pressure SI and recirculation is performed by the RHR pumps and heat exchangers. This is similar to many other Westinghouse NSSS plants, although some plants have low pressure SI pumps separate from the RHR system. Separate low pressure SI pumps are not modeled in any of the POS PRA models. The availability of separate low pressure SI pumps would reduce the risk in Mode 5 if modeled for POS 4. However, the POS 3 PRA model does not credit switching to RHR cooling if AFW cooling fails. This is a conservative approach with respect to the estimated risk for POS 3. Modeling separate low pressure SI pumps for POS 4 and modeling RHR cooling for POS 3 would result in lower risk for both plant operating states. Therefore, the conclusions of the WCAP would not change.

The reactor protection system for the transition risk PRA model is based on a solid state protection system. While many Westinghouse NSSS plants have relay protection systems, the reliability of the two systems is not significantly different, therefore, the model is applicable to both protection systems.

The modeled CCW system consists of two separate trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants. Plants with a common CCW header would be expected to have a higher failure probability for the CCW system. This would affect the risk for both the POS 3 and POS 4 PRA models, however, it would have a greater effect on the POS 4 model, because it would directly contribute to the loss of RHR cooling, whereas it does not directly contribute to the loss of AFW cooling for POS 3. Some insight to the relative effect of different CCW designs can be gained by comparing the POS 3 and POS 4 CDPs for the two cases presented in Table 6-14. The table provides the results from modeling one CCW train unavailable. The difference between the two cases is the number of CCW pumps assumed to be unavailable. The POS 4 results for the two cases do not differ by a large amount, and the POS 3 results differ by even less. However, the POS 4 CDP is more than 20 times greater than the POS 3 CDP for these cases. Plant design differences in the CCW system will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The modeled SW system consists of two trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants. Plants with a common SW header would be expected to have a higher failure probability for the SW system. This would affect the risk for both the POS 3 and POS 4 PRA models, however, it would have a greater effect on the POS 4 model because it would contribute to the loss of RHR cooling, whereas it does not directly contribute to the loss of AFW cooling for POS 3. Some insight to the relative effect of different SW designs can be gained by comparing the POS 3 and POS 4 CDPs for the two cases presented in Table 6-15. The table provides the results from modeling one SW train unavailable. The difference between the two cases is the number of SW pumps assumed to be unavailable. The POS 4 results for the two cases do not differ by a large amount and the POS 3 results differ by even less. However, the POS 4 CDP is approximately 10 times greater than the POS 3 CDP for these cases. Plant design differences in the SW system will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The electric power system modeled is a two train system with one diesel generator per train. This is a common design among the Westinghouse NSSS plants. Some plants have more redundancy in their design, including shared diesels between units. Based on the results presented in Table 6-10, the loss of offsite power event is a larger percent contributor to the POS 3 risk, than it is to the POS 4 risk and a more redundant design would be more beneficial for Mode 4 (POS 3).

The use containment spray or the containment coolers for backup cooling during recirculation has been modeled only to a limited extent for POS 3 for small LOCAs, and does not have a very large effect on the results (see Table 6-13). More detailed modeling to take credit for these systems may result in a larger risk decrease for POS 4 than it would for POS 3 because the probability of losing RHR cooling is greater than the probability of losing AFW cooling. However, the risk for POS 3 would also decrease and the POS 3 risk is approximately a factor of 7 lower than the risk for POS 4 (see Table 6-9). More detailed modeling will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The model chosen includes many safety system features and support system features that are common among many of the Westinghouse NSSS plants. The evaluation of design differences indicates that while the Westinghouse NSSS plant designs vary for the systems modeled, the model used provides representative results whose conclusions are applicable to all Westinghouse NSSS plants.

6.3.1.1 Plant Response Model

The following discusses the response of the plant to the initiating events in each POS. This model is based on a 3-loop Westinghouse NSSS plant at-power PRA model that has undergone the Owners Group Peer Review Process and has been updated in response to Peer Review comments. As discussed in Section 6.2, many of the events that can occur in the upper modes (Modes 2 and 3), or POSs, are mitigated similarly to the events that occur when at-power. Therefore, much of the modeling used in the at-power PRA model is also used in the modeling for events that occur in the transition modes. The events included in the baseline at-power model are:

Loss of Coolant Accidents

- Small LOCA
- Medium LOCA
- Large LOCA
- Interfacing Systems LOCA
- Reactor Vessel Rupture

Secondary Side Breaks

- Inside Containment
- Outside Containment

Transients

- Loss of Main Feedwater
- Partial Loss of Main Feedwater
- Loss of Condenser
- Positive Reactivity Insertion
- Primary System Transient
- Loss of Reactor Coolant Flow
- Reactor Trip
- Inadvertent Safety Injection Signal
- Turbine Trip
- Inadvertent Opening of a Steam Valve

Special Initiators

- Loss of CCW
- Loss of SW
- Loss of one DC Bus
- Loss of one AC Bus
- Loss of Instrument Air

Internal Flooding

- CCW Pipe Breaks
- SW Pipe Breaks

Others

- Loss of Offsite Power
- Steam Generator Tube Rupture
- Anticipated Transient Without Scram

POS 1 and POS 7: Plant Response Model

In POS 1, the plant conditions, in terms of RCS and secondary side pressures and temperatures, and system availabilities, are very similar to at-power plant conditions. Therefore, the at-power plant PRA model is used with several modifications. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- Medium LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- Small LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- RCP Seal LOCAs: A number of events can lead to a loss of seal cooling. These are:
  - Loss of CCW
  - Loss of SW
  - Loss of Offsite Power
  - Internal Flooding Events

  Event mitigation is identical to that modeled in the at-power PRA model.
- Loss of Feedwater Control: This leads to a loss of decay heat removal event which is modeled as a loss of main feedwater. Event mitigation is identical to that modeled in the at-power PRA model.
- Loss of Decay Heat Removal: This can occur when the plant has already transitioned to the auxiliary feedwater or startup feedwater system. The event is failure of the system to continue to

operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: Event mitigation is identical to that modeled in the at-power PRA model.
- SG Tube Rupture: Event mitigation is identical to that modeled in the at-power PRA model.
- Secondary Side Breaks: Event mitigation is identical to that modeled in the at-power PRA model.
- Boron Dilution: The positive reactivity insertion event mitigation is identical to that modeled in the at-power model.
- Rod Withdrawal: The positive reactivity insertion event mitigation is identical to that modeled in the at-power model.

The decay heat level in POS 7 is lower than that for POS 1. The lower decay heat level in POS 7 provides additional response time for the operator, however, no credit is taken for lower human error probabilities in POS 7.

POS 2 and POS 6: Plant Response Model

In POS 2, the RCS pressures and temperatures are very similar to at-power plant conditions, although both are being reduced. It is assumed that the plant is at the at-power pressure and temperature conditions for the RCS. The secondary side pressure is at the normal operating pressure. The reactor trip breakers are open so rod withdrawal is no longer a possible event. Signals for safety injection and steamline isolation are blocked. Operator actions are required to start equipment to mitigate a number of the potential events. The switchover from main feedwater to auxiliary feedwater (or startup feedwater) has occurred, therefore, loss of feedwater control is no longer an issue either. The at-power plant PRA model is applicable with several modifications. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- Medium LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- Small LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- RCP Seal LOCAs: A number of events can lead to loss of seal cooling. These are:
  - Loss of CCW
  - Loss of SW
  - Loss of Offsite Power
  - Internal Flooding Events

  Event mitigation is identical to that modeled in the at-power PRA model.

- Loss of Decay Heat Removal: This event is the failure of the decay heat removal source which is either the auxiliary feedwater system or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: Event mitigation is identical to that modeled in the at-power PRA model.

- SG Tube Rupture: Event mitigation is identical to that modeled in the at-power PRA model.

- Secondary Side Breaks: Event mitigation is identical to that modeled in the at-power PRA model.

POS 3 and POS 5: Plant Response Model

In POS 3, the RCS pressures and temperatures are significantly reduced compared to POS 1 and POS 2. Therefore, the LOCA events remain applicable, but at a reduced frequency. The secondary side is less than normal operating pressure. Similar to POS 2, the reactor trip breakers are open so rod withdrawal is no longer a possible event. Also, like POS 2, signals for safety injection and steamline isolation are blocked, therefore, operator actions are required to start equipment to mitigate a number of the potential events. Accumulators are isolated. Again, loss of feedwater control is no longer an issue. The at-power plant PRA model is applicable, with several modifications, to model this POS. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

- Medium LOCA: The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

- Small LOCA: The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

- Loss of Decay Heat Removal: This event is the failure of the decay heat removal source which is either the auxiliary feedwater system or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

- SG Tube Rupture: The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

- Secondary Side Breaks: The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

POS 4: Plant Response Model

In POS 4, the RCS pressures and temperatures are significantly reduced compared to POS 3. The secondary side conditions have also been significantly reduced. Therefore, the LOCA events, including RCP seal LOCAs, SGTR, and secondary side breaks are no longer applicable. The reactor trip breakers remain open so rod withdrawal is not a possible event. Core cooling has been switched to the RHR system (shutdown cooling). Due to this switchover, loss of inventory events are important. In addition, the loss of decay heat removal event is now related to the RHR system. Also, like POS 2 and POS 3, signals for safety injection and steamline isolation are blocked, therefore, operator actions are required to start equipment to mitigate a number of the potential events. Finally, cold overpressurization needs to be considered.

The at-power plant PRA model is no longer applicable due to the significant changes in the operating temperature and pressure on both the primary and secondary sides, available systems, and the events that can occur. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Loss of Inventory: Loss of inventory events are associated with alignment issues when transferring to RHR (shutdown) cooling. These events can be divided between isolable events and non-isolable events. If the leak is not isolable or if isolation fails, then high pressure injection, via SI/charging, is required followed by recirculation. If isolation is successful, then high pressure injection is required to make up for the inventory lost prior to the leak being isolated. Decay heat removal is also required.

- Loss of Decay Heat Removal: Loss of decay heat removal events are primarily loss of RHR events. These can occur during the switchover from AFW cooling to RHR cooling or from failure of the RHR system. Mitigation of the event depends on the availability of the AFW system. SG cooling credits the MD AFW pumps, but not the TD AFW pump due to the reduced secondary side pressure. If AFW is not available, then feed and bleed is required. While on feed and bleed, recovery of RHR is modeled.

- Loss of Offsite Power: The causes of loss of offsite power events in POS 4 are similar to these events that occur when at-power. The consequences are different in that RCP seal LOCAs are no longer an issue, but the TD AFW pump is not available for decay heat removal. In addition, the lower decay heat level provides additional time for recovery from the event. For event mitigation, recovery of offsite power is initially addressed, and if successful, is followed by decay heat removal equipment. If offsite power recovery fails, then the availability of decay heat removal mitigating systems with power from the EDGs is addressed. Decay heat removal can be provided by the RHR system, AFW systems (MD pumps only), or by feed and bleed. While on feed and bleed, recovery of RHR is modeled.

- Cold Overpressurization: Mitigation of cold overpressure events requires the operators to control charging and letdown. If this is not successful, then pressure relief via a pressurizer PORV or a RHR relief valve is necessary. Following successful pressure relief, the pressurizer PORV or a RHR relief valve is required to re-close. If it does not reclose, a loss of inventory event occurs. This model does not address mitigation of the possible loss of inventory event, but assumes the event leads to core damage.

- Boron Dilution: The high flux at shutdown alarm will indicate the event. The operators are required to identify and terminate the dilution source. If the dilution is not terminated, the operator must initiate boration.

6.3.1.2 Time in Each Plant Operating State

The time spent in each POS is important to the calculation of initiating event probabilities. When shutting the plant down due to Technical Specification requirements, the Technical Specifications control the maximum length of time allowed in each mode.

If inoperable equipment is not restored to operable status, or the Required Actions and associated Completion Times are not met, the Required Actions require that the unit be in Mode 3 in 6 hours and Mode 5 in 36 hours. Based on this, the time spent in the POSs while shutting the plant down are provided in Table 6-7.

The time spent in each POS when returning to power is not controlled by the Technical Specifications, but rather on the reason for the forced outage and satisfying the applicable Technical Specifications prior to returning to power operation. Information was collected from several Westinghouse NSSS plants for startup times following a forced non-refueling outage. Table 6-7 provides a summary of the time in each POS based on this information. Note that the times begin from the time that the decision is made to shutdown and do not include any time prior to the shutdown because the Completion Times associated with the Required Actions to restore the equipment to operable status are not changing.

6.3.1.3 Initiating Event Probabilities

Table 6-8 provides a summary of the initiating event probabilities. Event probabilities are used, not frequencies, since the risk metric being used (core damage) is determined on a shutdown basis, not a yearly basis. The core damage probability (CDP) can be converted to a CDF by multiplying the CDP by the number of shutdowns per year. The event probabilities for each POS are determined by multiplying the event frequencies by the time in the POS. Additional information on each initiating event probability is provided in the following:

- Large LOCA: In POS 1, POS 2, POS 6, and POS 7 the at-power frequency is used. In POS 3 and POS 5 the RCS pressure is significantly reduced compared to at-power, therefore, the frequency is reduced by a factor of 20, based on Reference 4. The at-power initiating event frequency is 5.0E-06/yr.

Table 6-7 Summary of the Time Spent in the Plant Operating States		
Plant Operating State	**Time in the Plant Operating State**	**Justification**
1	7 hours	POS 1 covers the transition out of Mode 1, through Mode 2, and into Mode 3. The Technical Specification Required Actions require that this be completed in 6 hours. An additional 1 hour was added to prepare for the shutdown.
2	3 hours	POS 2 covers the mid part of Mode 3 (also viewed as transitioning from the upper end of Mode 3 to the lower end of Mode 3). It is assumed that Mode 4 will be entered in 13 hours. This results in 6 hours (13 hours - 7 hours) to go from the upper end of Mode 3 to the upper end of Mode 4. Assuming that half this time is used to transition from the upper end of Mode 3 to the lower end of Mode 3, the time in POS 2 in 3 hours.
3 (shutdown to Mode 5)	3 hours	POS 3 covers the lower end of Mode 3 to the upper end of Mode 4. Following the logic presented in POS 2, half the time to go from Mode 3 to Mode 4 (6 hours) is assigned to POS 3.
3 (shutdown and repair in Mode 4)	49 hours	For a shutdown and repair in Mode 4, the 3 hours to go from Mode 3 to Mode 4 are added to the 46 hours for repair assumed for POS 4. This conservatively includes the startup time assigned to POS 4. See the discussion for POS 4 (startup).
4 (shutdown)	24 hours	POS 4 (shutdown) covers the transition through Mode 4 to Mode 5. Since the Technical Specification Required Actions require Mode 5 to be entered in 36 hours, and an additional 1 hour was added to prepare for the shutdown, the time in this POS is 37 - 7 - 3 - 3 = 24 hrs.
4 (startup)	46 hours	POS 4 (startup) covers the transition from Mode 5 to Mode 4. As discussed in Section 6.3.1.2, 46 hrs is based on plant operating experience.
5	19 hours	POS 5 covers the transition from Mode 4 to the lower end of Mode 3. As discussed in Section 6.3.1.2, 19 hrs is based on plant operating experience.
6	52 hours	POS 6 covers the middle part of Mode 3 (also viewed as moving from the lower end of Mode 3 to the upper end of Mode 3). As discussed in Section 6.3.1.2, 52 hrs is based on plant operating experience.
7	39 hours	POS 7 covers the upper end of Mode 3, through Mode 2, and the transition through Mode 1 to power operation. As discussed in Section 6.3.1.2, 39 hrs is based on plant operating experience.

Table 6-8 Summary of Initiating Event Probabilities for Each POS

Initiating Event	Initiating Event Probabilities							
	POS 1	POS 2	POS 3 (3 hr)	POS 3 (49 hr)	POS 4	POS 5	POS 6	POS 7
Large LOCA	4.0E-09	1.7E-09	8.6E-11	1.4E-09	NA	5.4E-10	3.0E-08	2.2E-08
Medium LOCA	3.2E-08	1.4E-08	6.9E-10	1.1E-08	NA	4.3E-09	2.4E-07	1.8E-07
Small LOCA	2.4E-06	1.0E-06	5.1E-08	8.4E-07	NA	3.3E-07	1.8E-05	1.3E-05
Interfacing Systems LOCA	1.1E-09	4.7E-10	4.7E-10	7.6E-09	NA	3.0E-09	8.1E-09	6.1E-09
Reactor Vessel Rupture	8.0E-11	3.4E-11	3.4E-11	5.6E-10	NA	2.2E-10	5.9E-10	4.5E-10
Loss of Inventory	NA	NA	NA	NA	1.4E-04	NA	NA	NA
RCP Seal LOCAs[1]	9.6E-07 + FT	4.1E-07 + FT	NA	NA	NA	NA	7.1E-06 + FT	5.3E-06 + FT
Loss of Feedwater Control[3]	6.8E-02 + FT	NA	NA	NA	NA	NA	NA	8.8E-02 + FT
Loss of Decay Heat Removal	NA	FT[2]	FT[2]	FT[2]	FT[2]	FT[2]	FT[2]	NA
Loss of Offsite Power	2.0E-05	8.5E-06	8.5E-05	1.4E-04	2.0E-04	5.4E-05	1.5E-04	1.1E-04
Cold Overpressurization	NA	NA	NA	NA	1.5E-03	NA	NA	NA
SG Tube Rupture	2.2E-06	9.3E-07	9.3E-07	1.5E-05	NA	5.9E-06	1.6E-05	1.2E-05
Secondary Side Breaks								
• Inside containment	5.7E-06	2.5E-06	2.5E-06	4.0E-05	NA	1.6E-05	4.3E-05	3.2E-05
• Outside containment	5.7E-06	2.5E-06	2.5E-06	4.0E-05	NA	1.6E-05	4.3E-05	3.2E-05
Boron Dilution	Note 4	NA	NA	NA	4.2E-03	NA	NA	Note 4
Rod Withdrawal	7.9E-05	NA	NA	NA	NA	NA	NA	4.4E-04

Notes:

1. RCP Seal LOCAs are caused by loss of CCW and loss of SW events, in addition to several flooding events that lead to degraded CCW and SW. The numerical values are the IE probability due to flooding events. The FT indicates that fault tree evaluations are used to determine the IE probability for the loss of CCW and loss of SW events. This evaluation is done as part of the model quantification.
2. The FT indicates that fault tree evaluations are used to determine the IE probability for the loss of decay heat removal events. This evaluation is done as part of the model quantification.
3. The loss of AFW is modeled as an initiating event for the time in the POS after AFW has been initiated. Fault tree evaluations are used to determine the IE probability.
4. Boron dilution IE probability is included in the rod withdrawal probability.

- Medium LOCA: In POS 1, POS 2, POS 6, and POS 7 the at-power frequency is used. In POS 3 and POS 5 the RCS pressure is significantly reduced compared to at-power, therefore, the frequency is reduced by a factor of 20, based on Reference 4. The at-power initiating event frequency is 4.0E-05/yr.

- Small LOCA: In POS 1, POS 2, POS 6, and POS 7 the at-power frequency is used. In POS 3 and POS 5 the RCS pressure is significantly reduced compared to at-power, therefore, the frequency is reduced by a factor of 20, based on Reference 4. The at-power initiating event frequency is 3.0E-03/yr.

- Interfacing Systems LOCA: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 1.36E-06/yr.

- Reactor Vessel Rupture: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 1.0E-07/yr.

- Loss of Inventory: This event is only applicable in POS 4. Based on Reference 4, the loss of inventory events are divided into three categories with the noted initiating event frequencies:

  - HLOCA – an event that results from an inadvertent transfer of reactor coolant out of the RCS
    IE frequency = 7.0E-03/yr

  - JLOCA – an event that occurs in a system connected to the RCS
    IE Frequency = 8.0E-03/yr

  - KLOCA – an event that results from a maintenance activity
    IE frequency = 3.0E-03/yr

  The total IE frequency is 1.8E-02/yr. The event probability is then determined by factoring in the length of time in POS 4.

- RCP Seal LOCAs: These are events that occur primarily due to a failure of seal cooling, such as, loss of CCW, loss of SW, and loss of offsite power. In addition, flooding events that are the result of breaks in the CCW or SW system can also result in RCP seal LOCAs due to loss of seal cooling. Loss of offsite power is addressed as a separate initiator and is discussed further below.

  The total CCW flooding frequency is 9.9E-04/yr and the total SW flooding frequency is 2.1E-04/yr. The event probabilities are then determined by factoring in the length of time for the applicable POS (POS 1, POS 2, POS 6, POS 7).

  The IE frequencies for the loss of CCW and loss of SW events are determined from fault tree evaluations in the model quantification. The component operating times in these fault trees are changed as required for the time in the POS.

- Loss of Feedwater Control: This event is applicable in POS 1 and POS 7 only. It considers reactor trips caused by loss of feedwater during the power reduction (POS 1) and power ascension (POS 7). Section 8.4 of Reference 5 provides a value for the probability of a reactor trip during a shutdown as 0.068 and the probability of a reactor trip during a startup as 0.088.

- Loss of Decay Heat Removal: This event is applicable in all the POSs. In POS 1 and POS 7 it is considered part of the Loss of Feedwater Control event or the loss of AFW after AFW is initiated. In POS 2, POS 3, POS 5, and POS 6 it is considered loss of auxiliary feedwater. In POS 4 it is considered loss of RHR. The IE frequencies for the loss of AFW and the loss of RHR are determined from fault tree evaluations in the model quantification.

- Loss of Offsite Power: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power IE frequency is used. The at-power IE frequency is 2.5E-02/yr. In POS 4 the events in Reference 6 were reviewed and an IE frequency determined. This frequency was similar to the 2.5E-02/yr value for the at-power LOSP event, therefore, the IE probability for POS 4 was also based on the 2.5E-02/yr value.

- Cold Overpressurization: This event only applies to POS 4. References 4 and 7 were reviewed to determine an appropriate frequency. Reference 7 was used since it provided a higher value of 1.8E-01/yr.

- SG Tube Rupture: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 2.7E-03/yr.

- Secondary Side Breaks Inside and Outside Containment: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 7.2E-03/yr.

- Rod Withdrawal: This event is only considered in POS 1 and POS 7. For POS 1 and POS 7, it is combined with the Boron Dilution initiating event and modeled as a positive reactivity insertion event with the at-power IE frequency of 9.9E-02/yr.

- Boron Dilution: For POS 1 and POS 7, it is combined with the Rod Withdrawal initiating event, and modeled as a positive reactivity insertion event with the at-power IE frequency of 9.9E-02/yr. For POS 4, Boron Dilution is modeled by itself with a frequency of 6.0E-05/hr based on Reference 4.

6.3.1.4 System Unavailabilities

System models for a typical Westinghouse NSSS plant were included in the model. As noted in Section 6.3.1, the base PRA model includes unavailability models for the following system configurations:

- AFW system – two MD pumps and one TD pump
- ECCS – two train system with each train including a high head and low head subsystem
- RPS – SSPS

- SW – two train system
- CCW – two train system
- Electrical power system – a two train system with one EDG per train

All system unavailabilities due to test and maintenance activities that are typically included in the at-power PRA model were eliminated. When a plant is starting up or shutting down, the specific plant configuration is modeled. The availability of components is known. Therefore, all component unavailability related to testing and maintenance activities were removed from the model.

6.3.1.5 Operator Actions

Operator actions and human error probabilities are key parameters for mitigating events in the transition states, particularly after blocking the automatic signals. Safety injection and steamline isolation signals are blocked in POS 2 and 3. Safety injection, steamline isolation, and AFW start signals are blocked in POS 4. For mitigation of several events, consecutive dependent operator actions are required. Dependencies between operator actions are addressed and accounted for as necessary.

6.3.2 Model Quantification

The base core damage probability results from quantifying the POS models are provided on Table 6-9. These results assume all the equipment is available, that is, no equipment is out of service or inoperable and all the unavailabilities in the model for test and maintenance are set to zero. Based on this, the core damage probabilities for transitioning to and returning from Mode 4 (POS 3) and to and from Mode 5 (POS 4) are:

- CDP (transition to and from POS 3/Mode 4) = 6.02E-06
- CDP (transition to and from POS 4/Mode 5) = 9.52E-06

There is an increase in CDP with the additional transition required to achieve Mode 5 as opposed to Mode 4. This is related to the risk associated with the transition from SG cooling to the shutdown (RHR) cooling and operator actions being required to initiate event mitigation equipment. The key initiating event is loss of RHR cooling with operator failure to establish alternate cooling.

Table 6-9 Core Damage Probability Results by Plant Operating State			
POS	Core Damage Probability		Time in POS (hours)
	Cooldown to POS 3	Cooldown to POS 4	
1	2.18E-07	2.18E-07	7
2	1.66E-07	1.66E-07	3
3	5.95E-07	7.09E-08	49 (cooldown to POS 3) 3 (Cooldown to POS 4)
4	NA	4.03E-06	70
5	4.79E-07	4.79E-07	19
6	3.30E-06	3.30E-06	52
7	1.26E-06	1.26E-06	39
Total	6.02E-06	9.52E-06	

Since the primary objective of this analysis is to identify the appropriate endstate, Mode 4 or Mode 5, an examination of the initiating event contributors to core damage for each of these endstates provides relevant insights. This information is provided on Table 6-10. The values in this table are given as core damage probabilities based on the plant remaining in the POS for the time indicated in Table 6-9 with all systems available. The following is concluded:

- The core damage probability in Mode 5 (POS 4) is more than 6 times greater than that for Mode 4 (POS 3).
- The largest initiating event contributor in either mode is a loss of decay heat removal. In Mode 4 the plant is using AFW for removal of decay heat and the event is initiated by a failure of the operating pump. The other motor-driven pump and the turbine-driven AFW pump are available. In Mode 5 the plant is using the RHR system for decay heat removal. Included in this initiating event is the switchover from SG (AFW) cooling to shutdown (RHR) cooling. All actuations of mitigating systems are by operator actions and the turbine-driven pump is not available.
- Small LOCAs or loss of inventory events are larger risk contributors in Mode 5 than in Mode 4. In Mode 5, loss of inventory events can be initiated by the alignment change from SG cooling to shutdown cooling. Event mitigation relies on operator actions in both endstates.

Based on these quantitative results, it is concluded that Mode 4 is preferred over Mode 5 as the endstate. The initiating event contributions for POS 3 and POS 4 in Table 6-9 are shown in Table 6-10. The top 100 cutsets for the POS 3 and POS 4 quantifications are presented in Appendix A. A sensitivity case that examines the time duration for POS 4 is presented in Section 6.5.

Table 6-10 Summary of Initiating Event Contribution for POS 3 and POS 4		
	Core Damage Probability	
Initiating Event	**POS 3 (Mode 4) Endstate**	**POS 4 (Mode 5) Endstate**
Loss of Decay Heat Removal/RHR	30.4%	57.9%
Loss of Offsite Power	23.1%	17.5%
Small LOCA/Loss of Inventory	11.1%	16.7%
SG Tube Rupture	18.2%	NA
Secondary Side Break Outside Containment	5.0%	NA
Secondary Side Break Inside Containment	9.8%	NA
Interfacing Systems LOCA	0.2%	NA
Reactor Vessel Rupture	0.1%	NA
Cold Overpressure	NA	<0.1%
Boron Dilution	NA	7.9%

6.4 EVALUATION OF TECHNICAL SPECIFICATION REQUIRED ACTION ENDSTATES

This section provides an evaluation of each Technical Specification for which the endstate is proposed to be changed from Mode 5 to Mode 4. The Technical Specifications are listed in numerical order by Specification number as contained in NUREG-1431 (Reference 3). Qualitative and quantitative evaluations are presented to support the endstate change from Mode 5 to Mode 4.

Quantitative evaluations are performed if the components/systems are modeled in the POS risk models described in Section 6.3. In the quantitative evaluation, specific components/systems are modeled as inoperable in the POS risk models and the conditional CDP for each applicable POS is calculated. The risk models for POS 5, 6, and 7 model the restart of the unit after the inoperable equipment has been restored to operable status, therefore, the CDPs for these POSs are not requantified. The risk calculation results are presented for a cooldown to POS 3 and for a cooldown to POS 4. In describing the results, the total CDP representing a cool down to POS 4 (Mode 5) and startup is compared to the total CDP representing a cool down to POS 3 (Mode 4) and startup.

6.4.1 Technical Specification 3.3.2 – Engineered Safety Features Actuation System (ESFAS) Instrumentation

The ESFAS instrumentation initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. There are numerous ESFAS function LCOs. For this Technical Specification, each function is addressed separately.

Function 1. a. Safety Injection – Manual Initiation

Description

The safety injection system provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting the peak clad temperature to ≤ 2200°F), and

2. Boration to ensure recovery and maintenance of shutdown margin ($k_{eff} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other functions (e.g., reactor trip).

The operator can initiate both trains of safety injection at any time from the control room by pushing one of two push buttons. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

Limiting Condition for Operation

Two channels shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 include automatic actuation of safety injection and manual actuation of the equipment, however, credit is not taken for the manual initiation of safety injection. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel is inoperable, the other channel is available for the operator to initiate safety injection. If the operator is shutting down the unit because of an inoperable channel, there will be a heightened awareness that this protection feature is not fully operational. The operators can be expected to be prepared to address a unit transient requiring safety injection with one channel inoperable. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. The LERP in Mode 4 would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel is inoperable, the other channel is available for the operator to initiate safety injection. There will be a heightened operator awareness that this protection feature is not fully operational. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 1. b. Safety Injection – Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 1. a., Safety Injection – Manual Initiation.

There are two trains for automatic actuation. In Mode 4 adequate time is available to manually actuate required components in the event of a design basis accident, however, because of the large number of components actuated, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 model the block of the automatic SI signal for POSs 2, 3, 4, 5, and 6. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate safety injection. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring safety injection knowing that manual initiation may be required. In this case, the operator will have both manual channels available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions and mitigation strategies, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate safety injection. In addition, there are two channels of manual actuation that can perform the function. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 2. a. Containment Spray – Manual Initiation

Description

The containment spray system provides three primary functions:

1. Lowers containment pressure and temperature after a HELB in containment,
2. Reduces the amount of radioactive iodine in the containment atmosphere, and
3. Adjusts the pH of the water in the containment recirculation sump after a LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure,
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure, and
- Minimize corrosion of the components and systems inside containment following a LOCA.

The operator can initiate containment spray at any time from the control room by simultaneously actuating two containment spray actuation switches in the same train. Because an inadvertent actuation of containment spray could have undesirable consequences, two switches must be actuated simultaneously. There are two sets of two switches in the control room. Simultaneously actuating the two switches in either set will start both trains of containment spray.

Limiting Condition for Operation

Two channels per train and two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel or train inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel or train must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on evaluating the core damage probability. The containment spray system does not have a significant impact on the core damage probability for the plant operating states modeled as described in Section 6.3. This is confirmed by the results in Table 6-13. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel or train is inoperable, the other train is available for the operator to initiate containment spray. If the operator is shutting down the unit because of an inoperable channel or train, there will be a heightened awareness that this protection feature is not fully operational. A cool down to Mode 4 places the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. The LERP in Mode 4 would be small due to the lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression due to lower temperatures and pressures, and the corresponding increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel or train is inoperable, the other train is available for the operator to initiate containment spray. There will be a heightened operator awareness that this protection feature is not fully operational. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are required to be operable in Mode 4. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 2. b. Containment Spray – Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 2. a., Containment Spray – Manual Initiation.

There are two trains for automatic actuation. In Mode 4 adequate time is available to manually actuate required components in the event of a design basis accident, however, because of the large number of components actuated, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on evaluating the core damage probability. The containment spray system does not have a significant impact on the core damage probability for the plant operating states modeled as described in Section 6.3. This is confirmed by the results in Table 6-13. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate containment spray. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring containment spray knowing that manual initiation may be required. In this case, the operator will have both manual trains available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate containment spray. In addition, there are two trains for manual initiation that can actuate containment spray equipment. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are available. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 3. a (1) Containment Isolation, Phase A Isolation, Manual Initiation

Description

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except CCW, at a relatively low containment pressure indicative of primary or secondary system leaks. Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of CCW, are isolated.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual actuation of Phase A Containment Isolation also actuates Containment Purge and Exhaust Isolation.

Limiting Condition for Operation

Two channels shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel is inoperable, the other channel is available for the operator to initiate containment isolation. If the operator is shutting down the unit because of an inoperable channel, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring containment isolation with one channel inoperable. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. The LERP in Mode 4 would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel is inoperable, the other channel is available for the operator to initiate containment isolation. In addition, the two trains of automatic actuation logic are available to actuate containment isolation equipment. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are available. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 3. a (2) Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 3. a (1), Containment Isolation, Phase A Isolation, Manual Initiation.

There are two trains for automatic actuation. In Mode 4 adequate time is available to manually actuate required components in the event of a design basis accident, however, because of the large number of components actuated, actuation is simplified by the use of the manual switches. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate containment isolation Phase A. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring containment isolation knowing that manual initiation may be required. In this case, the operator will have both manual channels available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate containment isolation Phase A. In addition, the two channels of manual actuation are available to perform the function. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are available. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 3. b (1) Containment Isolation, Phase B Isolation, Manual Initiation

Description

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

The Phase B signal isolates CCW. Manual Phase B containment isolation is accomplished by the same switches that actuate containment spray. When the two switches in either set are actuated simultaneously, Phase B containment isolation and containment spray will be actuated in both trains.

Limiting Condition for Operation

Two channels per train and two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel or train inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change. The manual actuation of containment isolation Phase B uses the same switches and logic as containment spray. The bases for the proposed change provided for Function 2. a., Containment Spray – Manual Initiation, and for Function 3. a (1), Containment Isolation, Phase A Isolation, Manual Initiation, also apply to the manual initiation of containment isolation Phase B.

Defense-in-Depth Considerations

The defense-in depth considerations provided for Function 2. a., Containment Spray – Manual Initiation, and for Function 3. a (1), Containment Isolation, Phase A Isolation, Manual Initiation, also apply to the manual initiation of containment isolation Phase B.

Function 3. b (2) Containment Isolation, Phase B Isolation, Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 3. b (1), Containment Isolation, Phase B Isolation, Manual Initiation.

There are two trains for automatic actuation. The same channels and trains used for actuating containment spray are used for actuating containment isolation Phase B. Just as for containment spray, the automatic actuation logic and relays are required to be operable to support the manual initiation of containment isolation Phase B.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change. The automatic actuation of containment isolation Phase B uses the same channels and logic as containment spray. The bases for the proposed change provided for Function 2. b., Containment Spray – Automatic Actuation Logic and Actuation Relays, and Function 3. a (2), Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays also apply to the automatic actuation of containment isolation Phase B.

Defense-in-Depth Considerations

The defense-in depth considerations provided for Function 2. b., Containment Spray – Automatic Actuation Logic and Actuation Relays, and Function 3. a (2), Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays, also apply to the automatic actuation of containment isolation Phase B.

Function 7. a. Automatic Switchover to Containment Sump, Automatic Actuation Logic and Actuation Relays

Description

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

There are two trains for automatic actuation and the logic and actuation relays consist of the same features and operate in the same manner as described for Function 1. b.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 do not include explicit modeling of two trains for this function. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate switchover to the containment sump. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring safety injection and recirculation knowing that manual initiation of the switchover from the RWST to the containment sump may be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate switchover to the containment sump. In addition, the operator can perform the switchover manually. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 7. b and 7. c. Automatic Switchover to Containment Sump – Refueling Water Storage Tank (RWST) Level – Low Low Coincident With Safety Injection, and RWST Level – Low Low Coincident With Containment Sump Level – High

Description

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal operation.

In some units, additional protection from spurious switchover is provided by requiring a Containment Sump Level – High signal as well as RWST Level – Low Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level – High signal must be present, in addition to the SI signal and the RWST Level – Low Low signal, to transfer the suction of the RHR pumps to the containment sump.

The RWST has four level transmitters. Units with the containment sump level circuitry also have four channels for the sump level instrumentation. The logic requires two out of four channels to initiate the switchover from the RWST to the containment sump.

Limiting Condition for Operation

Four channels shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel inoperable.

Current Required Action Endstate

The current endstate for Required Action K.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within [6] hours, or the unit must be in Mode 3 in [12] hours and Mode 5 in [42] hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action K.2.2 to be in Mode 4 in [18] hours if the inoperable channel is not restored to operable status in [6] hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 do not include explicit modeling of four channels for this function. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel is inoperable, the other three channels are available to initiate switchover to the containment sump In addition, if the operator is shutting down the unit because of an inoperable channel, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring safety injection and recirculation knowing that manual initiation of the switchover from the RWST to the containment sump may be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel is inoperable, the other three channels are available to initiate switchover to the containment sump. The system redundancy is such that a single channel failure in addition to one channel being inoperable will not defeat the initiation of switchover from the RWST to the containment sump. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.2 Technical Specification 3.3.7 – Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

Description

The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Auxiliary Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREFS initiates filtered ventilation and pressurization of the control room.

The actuation instrumentation consists of redundant radiation monitors in the air intakes and control room area. A high radiation signal from any of these detectors will initiate both trains of the CREFS. The operator can initiate the CREFS at any time by using either of two switches in the control room. The CREFS is also actuated by a SI signal.

Limiting Condition for Operation

Two trains and [2] channels shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel or train for one or more functions are inoperable, Required Action A.1 requires the operator to place one train of CREFS in emergency mode. If one or more functions with two channels or two trains are inoperable, Required Actions B.1.1 and B.1.2 require the operator to place one or both trains of CREFS in emergency mode. In the unlikely event that this does not occur, the inoperable equipment does not increase the likelihood of an initiating event. An independent initiating event with a radioactive release must occur for radiation in the control room to be a concern. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The system design provides redundancy and defense in depth from the multiple channels, trains, and functions available to actuate CREFS. If one or two channels or trains in one or more functions are inoperable, the Required Actions require one or both CREFS trains to be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. In the unlikely event that this is not accomplished and Condition C is entered, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. The system design maintains sufficient defense-in-depth when the endstate is changed from Mode 5 to Mode 4.

6.4.3 Technical Specification 3.3.8 – Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation

Description

The FBACS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident [involving handling recently irradiated fuel] or a LOCA are filtered and adsorbed prior to exhausting to the environment. The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal (gaseous or particulate) or a SI signal. Initiation may also be performed manually as needed from the main control room.

High gaseous and particulate radiation, each monitored by either of [two] monitors, provides FBACS initiation. Each FBACS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of [two] channels, one for each train. Each channel contains a gaseous and particulate monitor. High radiation detected by any monitor or an SI signal from the ESFAS initiates fuel building isolation and starts the FBACS.

Limiting Condition for Operation

Two trains and [two] channels shall be operable.

Applicability

Modes 1, 2, 3, and 4 during movement of [recently] irradiated fuel assemblies in the fuel building.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system does not affect conditional core damage probability and is not modeled in the risk models described in Section 6.3.1. FBACS is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel or train for one or more functions are inoperable, Required Action A.1 requires the operator to place one train of FBACS in operation. If one or more functions with two channels or two trains are inoperable, Required Actions B.1.1 and B.1.2 require the operator to place one train of FBACS in operation or both trains in emergency mode. In the unlikely event that this does not occur, the inoperable equipment does not increase the likelihood of an initiating event. An independent initiating event (e.g., LOCA or fuel handling accident) must occur to require the operation of FBACS. A cool down to Mode 4 reduces the likelihood of a LOCA, leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions and mitigation strategies, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2.

Defense-in-Depth Considerations

The system design provides redundancy and defense in depth from the multiple channels, trains, and functions available to actuate FBACS. If one or two channels or trains in one or more functions are inoperable, the Required Actions require one or both FBACS trains to be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. In the unlikely event that this is not accomplished and Condition C is entered, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. The system design maintains sufficient defense-in-depth when the endstate is changed from Mode 5 to Mode 4.

6.4.4 Technical Specification 3.4.13 – RCS Operational Leakage

Description

Verifying RCS leakage to be within the LCO limits ensures that the integrity of the reactor coolant pressure boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage.

Limiting Condition for Operation

RCS operational leakage shall be limited to:

a. No pressure boundary leakage,
b. 1 gpm unidentified leakage,
c. 10 gpm identified leakage,
d. 1 gpm total primary to secondary leakage through all steam generators (SGs), and
e. [500] gallons per day primary to secondary leakage through any one SG.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met, or pressure boundary leakage exists.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met, or pressure boundary leakage exists.

Basis for Proposed Change

A RCS leakage that is not large enough to be considered a small LOCA would typically be classified as an event leading to a controlled shutdown. Controlled shutdowns are not included in the risk models described in Section 6.3.1, therefore a qualitative evaluation is performed for this proposed endstate change.

RCS leakage can be reduced to lower amounts in Mode 5 compared to Mode 4 because of the lower RCS pressure in Mode 5, however, the RCS pressure in Mode 4 is already significantly lower than at power which will reduce the effects of the RCS leakage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

In Mode 4, the RCS pressure is significantly reduced which reduces the leakage. All LOCA mitigating systems with the exception of the accumulators are available and RHR serves as the backup to auxiliary feedwater for decay heat removal. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.5 Technical Specification 3.4.14 – RCS Pressure Isolation Valve (PIV) Leakage

Description

RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary, that separate the high pressure RCS from an attached low pressure system. The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified leakage, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified leakage before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational leakage if the other is leak-tight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components.

Limiting Condition for Operation

Leakage from each RCS PIV shall be within limit.

Applicability

Modes 1, 2, and 3, and Mode 4, except valves in the RHR flow path when in, or during the transition to or from, the RHR mode of operation.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met.

Basis for Proposed Change

PIV leakage would not be considered a PRA initiating event and would be classified as an event leading to a controlled shutdown. Controlled shutdowns are not included in the risk models described in Section 6.3.1, therefore a qualitative evaluation is performed for this proposed endstate change.

This Technical Specification limits leakage primarily because of the concern of overpressurizing a lower pressure system that can lead to an interfacing system LOCA. PIV leakage can be reduced to a lower level in Mode 5 compared to Mode 4 because of the lower RCS pressure in Mode 5, however, the RCS pressure in Mode 4 is already significantly lower than at power which will reduce the effects of the PIV leakage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

In Mode 4, the RCS pressure is significantly reduced which reduces the PIV leakage. All LOCA mitigating systems with the exception of the accumulators are available and RHR serves as the backup to auxiliary feedwater for decay heat removal. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.6 Technical Specification 3.4.15 – RCS Leakage Detection Instrumentation

Description

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified leakage.

Limiting Condition for Operation

The following RCS leakage detection instrumentation shall be operable:

a. One containment sump (level or discharge flow) monitor,
b. One containment atmosphere radioactivity monitor (gaseous or particulate), and
c. [One containment air cooler condensate flow rate monitor.]

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action E.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action E.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The RCS leakage detection functions; containment sump monitor, containment atmosphere radioactivity monitor, and containment air cooler condensate flow, are not modeled in the risk models described in Section 6.3.1. These functions are not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one function is declared inoperable, the other functions are available to provide indication of RCS leakage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one function is inoperable, the other functions are available to provide indication of RCS leakage. In the unlikely event that Condition E occurs, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for detecting RCS leakage. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.7 Technical Specification 3.5.3 – ECCS – Shutdown

Description

This Technical Specification is only applicable in Mode 4. In Mode 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and RHR (low head). The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank can be injected into the RCS if required following an accident.

Limiting Condition for Operation

One ECCS train shall be operable.

Applicability

Mode 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of [Condition B] not met.

Current Required Action Endstate

The current endstate for Required Action C.1 is Mode 5. Specifically, the unit must be in Mode 5 in 24 hours.

Proposed Required Action and Endstate

Condition A is revised from "Required ECCS residual heat removal (RHR) subsystem inoperable." to "Required ECCS train inoperable." Required Action A.1 is revised from "Initiate action to restore required ECCS RHR subsystem to operable status." to "Initiate action to restore required ECCS RHR train to operable status." This change allows the unit to remain in Mode 4, rather than transitioning to Mode 5 with an inoperable ECCS high head subsystem.

Basis for Proposed Change

This Technical Specification is only applicable in Mode 4. There are two subsystems addressed by this Technical Specification; the ECCS RHR subsystem and the ECCS high head subsystem. Both subsystems are included in the risk models described in Section 6.3.1, therefore, a quantitative evaluation is performed.

Current Condition A addresses both RHR trains inoperable and Required Action A.1 requires that action be initiated to restore the required RHR subsystem to operable status with an immediate Completion Time. Required Action A.1 and the immediate Completion Time acknowledge that in this condition it is inappropriate to require the unit to be placed in a Mode where the only means of decay heat removal is not available, rather than to remain in a Mode where steam generator cooling is also available for decay heat removal. Therefore, the change in endstate to evaluate applies to an inoperable high head subsystem, for which a transition to Mode 5 is currently required by Required Action C.1 if it is not returned to operable status within the Completion Time.

To model the inoperability of both train of ECCS high head, the three charging pumps are modeled as inoperable. Only POS 3 and POS 4 are quantified because this Technical Specification is only applicable in Mode 4. The resulting CDPs for each POS are presented in Table 6-11.

Table 6-11 Technical Specification 3.5.3 ECCS – Shutdown

POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	NA	NA
2	NA	NA
3	1.27E-05	2.39E-06
4	NA	9.23E-05
5	4.79E-07	4.79E-07
6	NA	NA
7	NA	NA
TOTAL	1.32E-05	9.52E-05

The unavailability of a complete train of ECCS results in an increase in the CDP for both POS 3 and 4. When comparing the base case to the inoperable ECCS train case, the POS 3 CDP increased by a larger factor than the POS 4 CDP, however, the POS 4 CDP for the inoperable ECCS train is approximately 7 times greater than the POS 3 CDP. Proceeding to Mode 5 does not increase the protection available and additional risk is introduced by switching from AFW cooling to RHR cooling. This case supports remaining in Mode 4 (POS 3) for this configuration rather than cooling down to Mode 5 (POS 4).

Defense-in-Depth Considerations

The proposed change to the Required Action C.1 endstate does not change the operability requirement for the ECCS. One train still must be operable in Mode 4. If one train of RHR is inoperable, then remaining in Mode 4 provides core cooling from the AFW pumps with the operable RHR pump as a backup. If both trains of RHR are inoperable, then the unit will remain on AFW cooling while one train is restored. The probability of transients occurring that require the ECCS are less likely in Mode 4 than at-power and the risk associated with transferring to RHR cooling from AFW cooling is eliminated by remaining in Mode 4. Sufficient defense-in-depth is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5.

6.4.8 Technical Specification 3.5.4 – Refueling Water Storage Tank (RWST)

Description

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

Limiting Condition for Operation

The RWST shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The RWST is included in the risk models described in Section 6.3.1, therefore, a quantitative assessment is made for changing the endstate. Because safety injection is dependent on the RWST for the source of borated water, its inoperability is expected to increase the core damage probabilities above the base case values. The RWST was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-12.

Table 6-12 Technical Specification 3.5.4, RWST

POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	1.00E-05	1.00E-05
2	3.36E-06	3.36E-06
3	1.24E-05	2.36E-06
4	NA	9.23E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	3.08E-05	1.13E-04

With the RWST unavailable, safety injection and recirculation are not possible. Therefore, any loss of inventory events that cannot be isolated lead to core damage. For the inoperability of the RWST, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4, i.e., the upper portion of Mode 5) reduces the total core damage probability by more than a factor of 3. The primary accidents that rely on the RWST are the LOCAs and steam line breaks. These accidents are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. In Mode 4, the control rods are inserted and the typical steamline break limiting assumption of the highest worth stuck rod is an unlikely scenario. In addition, the emergency boration system is likely to be available. In Mode 4, transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. Proceeding to Mode 5 does not increase the protection available and additional risk is introduced by switching from AFW cooling to RHR cooling. Variations in boron concentration are likely to be small, therefore, a shutdown to Mode 4 instead of Mode 5 is also appropriate. The RWST temperature variations are also expected to be small because the volume of the tank is large. The design basis accidents that conservatively use the RWST temperature are analyzed at power operation. Therefore, a shutdown to Mode 4 is also appropriate. Based on the risk results in Table 6-12 and the above discussion, if the RWST is inoperable for reasons other than boron concentration or temperature, a shutdown to Mode 4 is appropriate.

Defense-in-Depth Considerations

In Mode 4, the transient conditions are less severe than at power so that variations in the RWST parameters or other reasons of inoperability are less significant. In addition, if the boron concentration is low, the emergency boration equipment is likely to be available to increase the RCS boron concentration. By changing the endstate for Required Action C.2 to Mode 4, the possibility of having a loss of inventory event due to switching to RHR cooling is eliminated, reducing the possibility that the RWST inventory would be required. Sufficient defense-in-depth is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5.

6.4.9 Technical Specification 3.6.1 – Containment (Atmospheric, Subatmospheric, Ice Condenser, and Dual) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.10 Technical Specification 3.6.2 – Containment Air Locks (Atmospheric, Subatmospheric, Ice Condenser, and Dual) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.11 Technical Specification 3.6.3 – Containment Isolation Valves (Atmospheric, Subatmospheric, Ice Condenser, and Dual) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.12 Technical Specification 3.6.4A – Containment Pressure (Atmospheric, Dual, and Ice Condenser) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.13 Technical Specification 3.6.4B – Containment Pressure (Subatmospheric) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.14 Technical Specification 3.6.5A – Containment Air Temperature (Atmospheric and Dual) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.15 Technical Specification 3.6.5B – Containment Air Temperature (Ice Condenser) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.16 Technical Specification 3.6.5C – Containment Air Temperature (Subatmospheric) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.17 Technical Specification 3.6.6A – Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

Description

The containment spray and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits.

The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of SW. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units are operating. The fans are normally operated at high speed with SW supplied to the cooling coils. In post accident operation following an actuation signal, the containment cooling system fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere.

Limiting Condition for Operation

Two containment spray trains and [two] containment cooling trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met, and Required Action and associated Completion Time of Condition C or D not met.

Current Required Action Endstate

The current endstate for Required Actions B.2 and E.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours for Condition B, and the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours for Condition E.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 60 hours, and revise the endstate for Required Action E.2 to be in Mode 4 in 12 hours.

Basis for Proposed Change

The containment spray system and containment cooling units are modeled for a few sequences in the risk models described in Section 6.3.1. In the fault tree models these systems provide backup cooling for recirculation. Their impact on CDF is minimal. The main impact of the inoperability of the containment spray and containment cooling units is in the containment response in the Level 2 analysis, which is not included in the risk models. For POS 4, neither system is credited for providing a backup cooling function. Note that POS 4 includes the upper portion of Mode 5 and these systems are not required to be operable.

Technical Specification 3.6.6A, Actions A, C, and D address combinations of inoperable trains of containment spray and containment cooling units. Technical Specification 3.6.6B, Actions A through E also address combinations of inoperable trains of containment spray and containment cooling units. The risk models described in Section 6.3.1 are used to model the combinations of inoperable equipment and determine the resulting CDP for POS 1, 2, and 3 for these two Specifications. The containment spray system and containment cooling system are not modeled for POS 4. Table 6-13 presents the combinations of inoperable equipment, the applicable Technical Specification and Action, and the resulting CDPs.

Table 6-13 Technical Specifications 3.6.6A and 3.6.6B Containment Spray and Containment Cooling Systems										
POS	Core Damage Probability									
	One Train Containment Spray Unavailable[1]		Two Trains Containment Spray Unavailable[2]		One Train Containment Cooling Units Unavailable[3]		Two Trains Containment Cooling Units Unavailable[4]		One Train Containment Spray, One Train Containment Cooling Units Unavailable[5]	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	2.18E-07	2.18E-07	2.18E-07	2.18E-07	2.22E-07	2.22E-07	3.98E-07	3.98E-07	2.22E-07	2.22E-07
2	1.66E-07	1.66E-07	1.66E-07	1.66E-07	1.67E-07	1.67E-07	2.11E-07	2.11E-07	1.67E-07	1.67E-07
3	5.95E-07	7.09E-08	5.95E-07	7.09E-08	5.98E-07	7.13E-08	7.11E-07	9.14E-08	5.98E-07	7.13E-08
4	NA	4.03E-06	NA	4.03E-06	NA	4.03E-06	NA	4.03E-06	NA	4.03E-06
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	6.02E-06	9.52E-06	6.02E-06	9.52E-06	6.03E-06	9.53E-06	6.36E-06	9.77E-06	6.03E-06	9.53E-06

Notes:

1. Technical Specifications 3.6.6A and 3.6.6B, Action A.
2. Technical Specification 3.6.6B, Action C.
3. Technical Specification 3.6.6A, Action C and Technical Specification 3.6.6B, Action B.
4. Technical Specification 3.6.6A, Action D and Technical Specification 3.6.6B, Action E.
5. Technical Specification 3.6.6B, Action D.

The results confirm that these two systems have little effect on the calculated CDPs from the base case. For containment spray, the results are the same as the base results. For the containment cooling units, the results are not significantly different. The conclusion for all five cases is that containment spray and containment cooling do not significantly affect the shutdown modes CDP and there is a risk increase by cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The containment spray and containment cooling systems are designed for accident conditions initiated at power. One train of each system satisfies the assumptions in the safety analyses and one train of containment spray is required to satisfy assumptions regarding iodine removal. If one train of either containment spray or containment cooling is inoperable the other train is available to mitigate the accident along with both trains of the other system. If both trains of containment cooling are inoperable, containment spray can serve as the cooling system and it also serves to remove iodine. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.18 Technical Specification 3.6.6B – Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit not taken for iodine removal by the Containment Spray System)

Description

The containment spray and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits.

The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of SW. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units are operating. The fans are normally operated at high speed with SW supplied to the cooling coils. In post accident operation following an actuation signal, the containment cooling system fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere.

Limiting Condition for Operation

Two containment spray trains and [two] containment cooling trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A, B, C, D, or E not met.

Current Required Action Endstate

The current endstate for Required Action F.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action F.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A, B, C, D, or E not met.

Basis for Proposed Change

This Technical Specification is very similar to Technical Specification 3.6.6A. However, because no credit is taken for iodine removal, a Required Action is provided to restore two inoperable trains of containment spray. If two trains of containment spray are inoperable, the containment cooling units are still available to provide containment cooling. The cases presented in Table 6-13 demonstrate that the containment spray case results are the same as the base results. For the containment cooling units, the results are not significantly different. The conclusion for all five cases is that containment spray and containment cooling do not significantly affect the shutdown modes CDP and there is a risk increase by cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The containment spray and containment cooling systems are designed for accident conditions initiated at power. One train of each system satisfies the assumptions in the safety analyses. If one train of either containment spray or containment cooling is inoperable the other train is available to mitigate the accident conditions along with both trains of the other system. If both trains of one system are unavailable, the two trains of the other system are available to provide containment cooling. Note that this Technical Specification does not take credit for iodine removal by the containment spray system. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.19 Technical Specification 3.6.6C – Containment Spray System (Ice Condenser)

Description

The containment spray system provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a design basis accident.

Each train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of RHR to additional redundant spray headers completes the containment spray system heat removal capability. Each RHR train is capable of supplying spray coverage, if required, to supplement the containment spray system. The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the ECCS is operating in the recirculation mode.

Limiting Condition for Operation

Two containment spray trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 60 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

If one train of containment spray is inoperable, the other train is still available to provide accident mitigation. The inoperability of one train of containment spray would not significantly affect the shutdown modes CDP and there is risk associated with cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The containment spray system is designed for accident conditions initiated at power. One train satisfies the assumptions in the safety analyses. In addition, the containment ice condenser is available and it is designed to handle a heat load in excess of the initial blowdown of a design basis LOCA, or any feedwater or steamline break event inside containment. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. In addition, RHR spray could be used if necessary for continued containment cooling. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.20 Technical Specification 3.6.6D – Quench Spray (QS) System (Subatmospheric)

Description

The quench spray system is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS system, operating in conjunction with the recirculation spray (RS) system, is designed to cool and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a design basis accident. Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a design basis accident.

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the QS system. The QS system is actuated either automatically by a containment High-High pressure signal or manually. Each train of the QS system provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal.

Limiting Condition for Operation

Two QS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

If one train of quench spray is inoperable, the other train is still available to provide accident mitigation. The inoperability of one train of quench spray would not significantly affect the shutdown modes CDP and there is risk associated with cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The quench spray system is designed for accident conditions initiated at power. One train satisfies the assumptions in the safety analyses. In addition, the containment temperature and pressure limits are set to account for the effects of an energy release during an event in full power operation. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.21 Technical Specification 3.6.6E – Recirculation Spray (RS) System (Subatmospheric)

Description

The recirculation spray system, operating in conjunction with the quench spray system, is designed to limit the post accident pressure and temperature in the containment to less than the design values and to depressurize the containment structure to a subatmospheric pressure in less than 60 minutes following a design basis accident. The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission product radioactivity from containment to the environment in the event of a design basis accident.

The RS system consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one 50% capacity spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls. The two outside RS subsystems' spray pumps are located outside containment and the two inside RS subsystems' spray pumps are located inside containment. Each RS train (one inside and one outside RS subsystem) is powered from a separate ESF bus. Each train of the RS system provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal.

Limiting Condition for Operation

Four RS subsystems [and a casing cooling tank] shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action F.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action F.2 to be in Mode 4 in 60 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include recirculation spray because recirculation spray is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If any of Technical Specification 3.6.6E Conditions A through E are entered, the recirculation spray system can still perform its safety function. Note that if the casing cooling tank is inoperable, the net positive suction head available to the outside RS subsystem pumps may not be sufficient. This situation is the same as Condition D. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur. This evaluation does not take credit for location differences of the subsystems and is, therefore, applicable to similar system configurations that have all of the pumps and heat exchangers located either inside or outside of the containment.

Defense-in-Depth Considerations

The recirculation spray system is designed for accident conditions initiated at power. One train (two subsystems) satisfies the assumptions in the safety analyses. In addition, the containment temperature and pressure limits are set to account for the effects of an energy release during an event in full power operation. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.22 Technical Specification 3.6.7 – Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

Description

The spray additive system is a subsystem of the containment spray system that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a design basis accident.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a design basis accident. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms.

For an eductor feed system, the spray additive system consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line.

For a gravity feed system, the spray additive system consists of one spray additive tank, two parallel redundant motor operated valves in the line between the additive tank and the RWST, instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by a balanced gravity feed from the additive tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the spray pump suction.

Limiting Condition for Operation

The spray additive system shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 60 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the spray additive system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. The spray additive system assists in reducing the iodine fission product inventory. Containment spray by itself removes some iodine from the containment atmosphere, so iodine removal will still occur with an inoperable spray additive system. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The spray additive system is designed for accident conditions initiated at power. The containment spray system will remove some iodine from the containment atmosphere without the additive system and two trains of containment spray are required to be operable. The spray additive system also serves to provide the proper pH in the containment sump. For most containments, a backup system for containment sump pH is not available, but proceeding to Mode 5 does not increase the protection available. Note that the ice condenser containments have ice that is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere and minimizes the occurrence of the chloride and caustic stress corrosion on mechanical systems and components. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.22a Recirculation Fluid pH Control System

Some Westinghouse NSSS plants have replaced the spray additive system with a passive ECCS recirculation fluid pH control system. Although the Technical Specification for this system is not contained in NUREG-1431, the endstate is Mode 5 if the system is inoperable, and the Required Action and associated Completion Time are not met. The system consists of baskets in the containment sump with a specified amount of trisodium phosphate in each basket. The trisodium phosphate dissolves when the containment sump level increases to the level of the baskets.

It is highly unlikely that all of the baskets would be empty, therefore, an inoperable recirculation fluid pH control system would still provide some pH control. The justification for changing the endstate to Mode 4 for Technical Specification 3.6.7, "Spray Additive System," is also applicable to the recirculation fluid pH control system, since they perform the same function.

The recirculation fluid pH control system Technical Specification currently requires the unit to be in Mode 3 in 6 hours and Mode 5 in 84 hours if the system is inoperable, and the Required Action and associated Completion Time are not met. The current Mode 5 endstate is proposed to be changed to require the unit to be in Mode 4 in 60 hours if the Required Action and associated Completion Time are not met.

6.4.23 Technical Specification 3.6.8 – Shield Building (Dual and Ice Condenser) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.24 Technical Specification 3.6.11 – Iodine Cleanup System (ICS) (Atmospheric and Subatmospheric)

Description

The iodine cleanup system functions together with the containment spray and cooling systems following a design basis accident to reduce the potential release of radioactive material, principally iodine, from the containment to the environment.

The iodine cleanup system consists of two 100% capacity, separate, independent, and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, a demister, a HEPA filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. Each ICS train is powered from a separate ESF bus and is provided with a separate power panel and control panel. During normal operation, the containment cooling system is aligned to bypass the ICS HEPA filters and charcoal adsorbers. For ICS operation following a design basis accident, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

Limiting Condition for Operation

Two ICS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the iodine cleanup system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. The iodine cleanup system assists in reducing the iodine fission product inventory. If one iodine cleanup system train is inoperable, the other is available to perform its function. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The iodine cleanup system is designed for accident conditions initiated at power. One train of the iodine cleanup system is available and capable of performing its design basis function. In addition, the containment spray system will also remove iodine from the containment atmosphere and two trains of containment spray are available. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.25 Technical Specification 3.6.12 – Vacuum Relief Valves (Atmospheric and Ice Condenser)

Description

The purpose of the vacuum relief lines is to protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of containment cooling features, such as the containment spray system. Multiple equipment failures or human errors are necessary to cause inadvertent actuation of these systems.

The containment pressure vessel contains two 100% vacuum relief lines that protect the containment from excessive external loading.

Limiting Condition for Operation

[Two] vacuum relief lines shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the vacuum relief valves because they are a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. The vacuum relief valves protect the containment from negative pressure due to an inadvertent actuation of the containment spray system. Inadvertent actuation of the containment spray system does not lead directly to core damage and large early releases. Another event needs to occur to cause the core damage. In addition, if one vacuum relief line is inoperable, the other line is available to provide the containment protection. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

Inadvertent actuation of containment spray is unlikely due to the instrumentation design for automatic and manual initiation. Inadvertent actuation of the containment spray system does not lead directly to core damage and large early releases by itself. Another event needs to occur to cause core damage. In addition, if one vacuum relief line is inoperable, the other line is available to provide the containment protection. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.26 Technical Specification 3.6.13 – Shield Building Air Cleanup System (SBACS) (Dual and Ice Condenser)

Description

The containment has a secondary containment called the shield building, that is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a LOCA. This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The SBACS establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment.

The SBACS consists of two separate and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, moisture separators, a HEPA filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. During normal operation, the shield building cooling system is aligned to bypass the SBACS's HEPA filters and charcoal adsorbers. For SBACS operation following a design basis accident, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

Limiting Condition for Operation

Two SBACS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the SBACS because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one SBACS train is inoperable, the other train is available to provide the annulus air cleanup. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one SBACS train is inoperable, the other train is available to provide the annulus air cleanup. In addition, two trains of containment spray are available to mitigate radioactive releases after an event. Significant leakage from containment is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and reduced likelihood of a LOCA in the shutdown modes, and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.27 Technical Specification 3.6.14 – Air Return System (ARS) (Ice Condenser)

Description

The air return system is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a design basis accident. The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting the post accident pressure and temperature in containment to less than design values. The air return system provides post accident hydrogen mixing in selected areas of containment. The air return system also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the containment spray system can cool it.

The air return system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers with isolation valves. The ARS fans are automatically started and the hydrogen collection header isolation valves are opened by the containment pressure High-High signal 10 minutes after the containment pressure reaches the pressure setpoint. Each train is powered from a separate ESF bus.

Limiting Condition for Operation

Two ARS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the air return system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one air return train is inoperable, the other train is available to assist in cooling the containment atmosphere. In addition, two trains of containment spray are available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one air return train is inoperable, the other train is available to assist in cooling the containment atmosphere. Containment cooling is still available from the containment ice condenser and from two trains of containment spray. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.28 Technical Specification 3.6.15 – Ice Bed (Ice Condenser) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.29 Technical Specification 3.6.16 – Ice Condenser Doors (Ice Condenser) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.30 Technical Specification 3.6.17 – Divider Barrier Integrity (Ice Condenser) ***SECTION REMOVED***

Text within this section has been removed; section headings, however, have been retained to preserve section numbering.

6.4.31 Technical Specification 3.6.18 – Containment Recirculation Drains (Ice Condenser)

Description

The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. [Twenty of the 24] ice condenser bays have a floor drain at the bottom to drain the melted ice into the lower compartment (in the [4] bays that do not have drains, the water drains through the floor drains in the adjacent bays). A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, the check valve opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it drains to the floor and provides a source of borated water at the containment sump for long term use by the ECCS and the containment spray system during the recirculation mode of operation.

The two refueling canal drains are at low points in the refueling canal. In the event of a design basis accident, the refueling canal drains are the main return path to the lower compartment for containment spray system water sprayed into the upper compartment.

Limiting Condition for Operation

The ice condenser floor drains and the refueling canal drains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the ice condenser and refueling canal drains because they are containment systems and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one drain is inoperable, there are other drains available to perform the function of transferring water to its intended destination. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one ice condenser floor drain is inoperable, there are [19] others available to drain the water to the lower compartment. If one refueling canal drain is inoperable, there is another refueling canal drain to transfer the containment spray water to the lower compartment. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.32 Technical Specification 3.7.7 – Component Cooling Water (CCW) System

Description

The component cooling water system provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident. The CCW system serves as a barrier to prevent the release of radioactive byproducts between potentially radioactive systems and the service water system, and then to the environment.

A typical CCW System is arranged as two independent, full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components are isolated.

Limiting Condition for Operation

Two CCW trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met.

Basis for Proposed Change

The CCW system is included in the risk models described in Section 6.3.1. The model includes two independent trains with one pump in each train, and a third pump that can be aligned to either train. A quantitative evaluation is performed using the risk models.

Two scenarios are modeled. In the first scenario, CCW Train A, that has two pumps aligned to the train, is assumed to be inoperable. The Train A configuration is conservative for units that have a backup pump in each train. In the second scenario, CCW Train B, that has one pump aligned to the train, is assumed to be inoperable. Both cases are analyzed and the results are presented in Table 6-14.

Table 6-14	Technical Specification 3.7.7, CCW			
POS	Core Damage Probability			
	Train A – Two Pumps Unavailable		Train B – One Pump Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	3.29E-06	3.29E-06	1.47E-06	1.47E-06
2	1.12E-06	1.12E-06	1.22E-06	1.22E-06
3	6.85E-07	7.65E-08	6.20E-07	7.25E-08
4	NA	1.51E-05	NA	1.44E-05
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	1.01E-05	2.46E-05	8.35E-06	2.22E-05

For the inoperability of CCW Train A that includes the swing pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by a factor of 2. The POS 1 and 2 contributions are relatively high because of the loss of CCW initiating event and resulting RCP seal LOCAs that are modeled in these POSs, and this case assumes two CCW pumps are inoperable.

For the inoperability of CCW Train B that includes one pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of 2. The CCW Train B case total CDP and the POS 4 CDP are less than the Train A case results because the Train A case models two inoperable CCW pumps. The conclusion for both cases is there is less risk associated with a cool down to Mode 4 than there is with a cool down to Mode 5 when a train of CCW is inoperable.

Defense-in-Depth Considerations

One CCW train will be operating when the unit enters Mode 4, therefore, failures of the pump to start or valves to open are not applicable. Each train is designed to handle 100% of the heat loads during power operation and accident conditions. If the unit design includes a swing pump, it is highly probable that the swing pump would be available to backup the operating pump. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.33 Technical Specification 3.7.8 – Service Water System (SWS)

Description

The service water system provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident. During normal operation, and a normal shutdown, the

SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

A typical SWS consists of two separate, 100% capacity, safety related, cooling water trains. Each train consists of two 100% capacity pumps, one component cooling water heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW system and typically is the backup water supply to the auxiliary feedwater system.

Limiting Condition for Operation

Two SWS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met.

Basis for Proposed Change

The service water system is included in the risk models described in Section 6.3.1. The model includes two independent trains with one pump in each train, and a third pump that can be aligned to either train. A quantitative evaluation is performed using the risk model.

Two scenarios are modeled. In the first scenario, SWS Train A, that has two pumps aligned to the train, is assumed to be inoperable. The Train A configuration is conservative for units that have a backup pump in each train. In the second scenario, SWS Train B, that has one pump aligned to the train, is assumed to be inoperable. Both cases are analyzed and the results are presented in Table 6-15.

Table 6-15 Technical Specification 3.7.8, SWS				
POS	Core Damage Probability			
	Train A – Two Pumps Unavailable		Train B – One Pump Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	1.56E-05	1.56E-05	8.68E-07	8.68E-07
2	8.99E-07	8.99E-07	3.67E-07	3.67E-07
3	2.58E-06	2.06E-07	1.82E-06	1.58E-07
4	NA	2.21E-05	NA	2.07E-05
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	2.41E-05	4.38E-05	8.09E-06	2.71E-05

For the inoperability of SWS Train A that includes the swing pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by almost a factor of 2.

For the inoperability of SWS Train B that includes one pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of 3. This results in a greater reduction in CDP than the Train A case, although the POS 4 CDP for the Train A case is greater than the POS 4 CDP for the Train B case. The greater reduction in CDP for the Train B case is because the POS 4 CDP dominates the Train B results, whereas the Train A results have larger CDPs, particularly for POS 1. The main conclusion of both cases is the same; a cool down to Mode 4 instead of Mode 5 reduces the risk of the shutdown process when a train of the SWS is inoperable.

Defense-in-Depth Considerations

One SWS train will be operating when the unit enters Mode 4, therefore, failures of the pump to start or valves to open are not applicable. Each train is designed to handle 100% of the heat loads during power operation and accident conditions. If the plant design includes a swing pump or redundant pumps in each train, it is highly probable that another pump would be available to backup the operating pump. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.34 Technical Specification 3.7.9 – Ultimate Heat Sink (UHS)

Description

The ultimate heat sink provides a heat sink for the removal of process and operating heat from safety related components during an accident, as well as during normal operation. This is done by utilizing the service water system and the component cooling water system.

The UHS has been defined as the water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the FSAR. If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the heat sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of water sources are used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the water sources include a water source contained by a structure, it is likely that a second source will be required.

Limiting Condition for Operation

The UHS shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

[Required Action and associated Completion Time of Condition A or B not met, or] UHS inoperable [for reasons other than Condition A or B].

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the [Required Action and associated Completion Time of Condition A or B not met, or] UHS inoperable [for reasons other than Condition A or B].

Basis for Proposed Change

The risk models described in Section 6.3.1 do not include cooling towers because the models are based on a plant that has a cooling pond that supplies cooling water to the service water system. Therefore, a qualitative approach is used for this endstate change.

The Actions of Technical Specification 3.7.9 address degradations to the cooling capability of the ultimate heat sink. Because of the limitations on water temperature and the variety of designs of the ultimate heat sink, the most likely scenario for entering Condition C is that the cooling capability of the ultimate heat sink is only partially degraded. A cool down to Mode 4 places the unit in a state in which the heat loads are significantly less than at full power. There are additional risks associated with a cool down to

Mode 5, e.g., switching to RHR cooling, and transferring the heat load to the component cooling water system.

Defense-in-Depth Considerations

The ultimate heat sink is designed to remove 100% of the heat loads generated during power operation and accident conditions. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.35 Technical Specification 3.7.10 – Control Room Emergency Filtration System (CREFS)

Description

The CREFS provides a protected environment from which the operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters is downstream of the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank.

Limiting Condition for Operation

Two CREFS trains shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one CREFS train is inoperable, the other train provides the necessary filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event with a radioactive release must occur for radioactive filtration to be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2 of the WCAP. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event was to occur.

CREFS also provides protection from chemical releases, toxic gas releases, or radiation releases from other sources on-site and offsite. The likelihood of these events occurring are independent of the unit operating Mode.

Defense-in-Depth Considerations

If one CREFS train is inoperable, the other train remains available to provide control room filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event and radioactive release must occur for filtration to be required. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.36 Technical Specification 3.7.11 – Control Room Emergency Air Temperature Control System (CREATCS)

Description

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent and redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control.

Limiting Condition for Operation

Two CREATCS trains shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one CREATCS train is inoperable, the other train provides the necessary temperature control. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one CREATCS train is inoperable, the other train remains available to provide control room temperature control. The slower nature of accident event progression in the shutdown modes, and increased time for operator actions and mitigation strategies, limit the severity of accidents in the shutdown modes. The inoperability of equipment does not affect the likelihood of an event occurring and some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.37 Technical Specification 3.7.12 – Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

Description

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a LOCA. The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower areas of the Auxiliary Building.

The ECCS PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters is downstream of the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a SI signal.

Limiting Condition for Operation

Two ECCS PREACS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. This system has no impact on plant CDP. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one ECCS PREACS train is inoperable, the other train provides the necessary filtration. If two trains are inoperable due to an inoperable ECCS pump room boundary, a LOCA must also occur to require the operation of the ECCS PREACS. A LOCA in Mode 4 is less likely due to the reduced RCS temperature and pressure. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one ECCS PREACS train is inoperable, the other train remains available to provide pump room air filtration. If two trains are inoperable due to an inoperable ECCS pump room boundary, a LOCA must also occur to require operation of the ECCS PREACS. The slower nature of accident event progression in the shutdown modes, and increased time for operator actions and mitigation strategies, limit the severity of accidents in the shutdown modes. In addition, a LOCA is less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.38 Technical Specification 3.7.13 – Fuel Building Air Cleanup System (FBACS)

Description

The FBACS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or a LOCA. A LOCA is analyzed to address radioactive leakage from the ECCS. The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters, functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters is downstream of the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal.

Limiting Condition for Operation

Two FBACS trains shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

[Required Action and associated Completion Time of Condition A or B not met in Mode 1, 2, 3, or 4.] or Two FBACS trains inoperable in Mode 1, 2, 3, or 4 for reasons other than Condition B.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the [Required Action and associated Completion Time of Condition A or B not met in Mode 1, 2, 3, or 4.] or Two FBACS trains inoperable in Mode 1, 2, 3, or 4 for reasons other than Condition B.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. This system has no impact on plant CDP. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one FBACS train is inoperable, the other train provides the necessary filtration. If two FBACS trains are inoperable, a LOCA or fuel handling accident must also occur to require the operation of the FBACS. A LOCA in Mode 4 is less likely due to the reduced RCS temperature and pressure. If irradiated fuel is being moved, Condition E of the Specification requires that the movement be suspended. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one FBACS train is inoperable, the other train remains available to provide fuel building air filtration. If two FBACS trains are inoperable, a LOCA or fuel handling accident must also occur to require operation of the FBACs. LOCAs are less likely in Mode 4 because of the reduced RCS temperature and pressure in Mode 4 and Condition E reduces the probability of a fuel handling accident. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.39 Technical Specification 3.7.14 – Penetration Room Exhaust Air Cleanup System (PREACS)

Description

The PREACS filters air from the penetration area between containment and the Auxiliary Building.

The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the air stream, also form part of the system. A second bank of HEPA filters, downstream of the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection signal.

Limiting Condition for Operation

Two PREACS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. This system has no impact on plant CDP. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one PREACS train is inoperable, the other train provides the necessary filtration. If two PREACS trains are inoperable due to an inoperable penetration room boundary, a LOCA and a passive failure in the penetration room must occur to require air filtration. LOCAs are less likely in Mode 4 because of the reduced RCS temperature and pressure. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one PREACS train is declared inoperable, the other train remains available to provide penetration room air filtration. If two PREACS trains are inoperable due to an inoperable penetration room boundary, a LOCA and passive failure in the penetration room must occur to require air filtration. A LOCA is less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.40 Technical Specification 3.8.1 – AC Sources – Operating

Description

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B DGs). The design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the ESF systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

Offsite power is typically supplied to the unit switchyard(s) from the transmission network by two transmission lines. From the switchyard(s), two electrically and physically separated circuits provide AC power, through step down station auxiliary transformers, to the 4.16 kV ESF buses. An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1] minute after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.

After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone.

Limiting Condition for Operation

The following AC electrical sources shall be operable:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System,

b. Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and

c. [Automatic load sequencers for Train A and Train B.]

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A, B, C, D, E, or [F] not met.

Current Required Action Endstate

The current endstate for Required Action G.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action G.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A, B, C, D, E, or [F] not met.

Basis for Proposed Change

Electric power is included in the risk models described in Section 6.3.1. Each condition of this Technical Specification is evaluated quantitatively by modeling the inoperable equipment. The Conditions are grouped according to equipment and are not necessarily in the order in which the Condition appears in the Technical Specification.

Conditions A and C – Offsite Power Circuits

Condition A – One Offsite Circuit Inoperable

A transformer in the risk models was selected as the component to use for one offsite circuit inoperable. The transformer was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The results are presented in Table 6-16.

Condition C – Two Offsite Circuits Inoperable

A transformer common cause basic event in the risk models was selected to model two offsite circuits inoperable. The common cause basic event was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The results are presented in Table 6-16.

There is a significant increase in the CDP when two offsite power circuits are inoperable compared to one circuit being inoperable. This is expected because the unit is dependent on the diesel generators for power when both circuits are inoperable. For one circuit inoperable, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by almost a factor of 2. When both offsite circuits are inoperable, the decrease in CDP for remaining in Mode 4 is approximately a factor of 5.

Table 6-16	Technical Specification 3.8.1, Conditions A and C, Offsite AC Circuits Unavailable			
POS	Core Damage Probability			
	One Offsite Circuit Unavailable		Two Offsite Circuits Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	5.82E-07	5.82E-07	8.62E-04	8.62E-04
2	2.35E-07	2.35E-07	1.47E-04	1.47E-04
3	9.34E-07	1.37E-07	1.09E-04	1.47E-04
4	NA	6.53E-06	NA	4.86E-03
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	6.79E-06	1.25E-05	1.12E-03	6.02E-03

<u>Conditions B and E – Diesel Generators</u>

Condition B – One Diesel Generator (DG) Inoperable

One diesel generator was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The resulting CDPs are presented in Table 6-17.

Condition E – Two Diesel Generators Inoperable

A common cause basic event for both diesel generators was selected to model the inoperability of two diesel generators. The POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-17.

Table 6-17	Technical Specification 3.8.1, Conditions B and E, Diesel Generator(s) Unavailable			
POS	Core Damage Probability			
	One DG Unavailable		Two DGs Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	3.54E-07	3.54E-07	2.38E-06	2.38E-06
2	2.15E-07	2.15E-07	9.51E-07	9.51E-07
3	1.68E-06	1.49E-07	1.79E-05	1.31E-06
4	NA	1.04E-05	NA	1.05E-04
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	7.29E-06	1.62E-05	2.63E-05	1.15E-04

For the inoperability of one DG, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of 2. When both DGs are inoperable, the decrease in CDP for remaining in Mode 4 is approximately a factor 4.

Condition D – One Offsite Circuit and One DG Inoperable

A transformer and diesel generator were selected to model the inoperable equipment for this condition. The POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-18.

Table 6-18 Technical Specification 3.8.1, Condition D One Offsite Circuit and One DG Unavailable

POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	4.64E-06	4.64E-06
2	1.03E-06	1.03E-06
3	5.95E-06	9.37E-07
4	NA	3.95E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.67E-05	5.11E-05

The CDPs for one offsite AC circuit and one DG inoperable, as expected, are greater than the CDPs for one offsite AC circuit inoperable and are also greater than the CDPs for one DG inoperable. The total CDP decreases by more than a factor of 3 when the unit is cooled down to Mode 4 (POS 3) instead of Mode 5 (POS 4). This is a larger decrease than for either the one offsite AC circuit inoperable case or the one DG inoperable case.

Condition F – One Load Sequencer Inoperable

One load sequencer was selected to model this condition. The load sequencer was modeled as inoperable and the POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-19.

Table 6-19	Technical Specification 3.8.1, Condition F One Load Sequencer Unavailable	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	3.94E-07	3.94E-07
2	2.32E-07	2.32E-07
3	1.81E-06	1.57E-07
4	NA	1.09E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	7.48E-06	1.67E-05

The results for this case are similar to the results for the one DG inoperable case, which is expected, because the consequences of the inoperable equipment are similar. For the inoperability of one load sequencer, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of 2.

For each of the Conditions of this Technical Specification that were evaluated, the results show that a cool down to Mode 4 instead of Mode 5 reduces the risk of the shutdown process when the unit has inoperable electrical power components. The reduction in risk ranges from a factor of 2 to a factor of 5.

Defense-in-Depth Considerations

The electric power design maintains defense-in-depth when the Conditions for this Technical Specification are considered. Two trains of diesel generators are available if two offsite power circuits are inoperable and two offsite power circuits are available if two diesel generators are inoperable. If an offsite power circuit and/or a diesel generator are inoperable, at least one of each remains available. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.41 Technical Specification 3.8.4 – DC Sources – Operating

Description

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). The DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure.

The typical 125/250 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A and Train B). Each subsystem consists of two 125 VDC batteries (each battery with 50% capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling.

The typical 250 VDC source is obtained by use of the two 125 VDC batteries connected in series. Additionally there is one spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service. If the spare battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, the 125/250 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

Limiting Condition for Operation

The Train A and Train B DC electrical power subsystems shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

DC power is included in the risk models described in Section 6.3.1. Each Condition of this Technical Specification is evaluated quantitatively by modeling the inoperable equipment.

Condition A – One [or Two] Battery Charger[s on one train] Inoperable

There is one battery charger per battery (and per train) in the risk models described in Section 6.3.1. One battery charger was modeled as inoperable and the POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-20.

Table 6-20	Technical Specification 3.8.4, Condition A Battery Charger Unavailable	
POS	**Core Damage Probability**	
	Cooldown to POS 3	**Cooldown to POS 4**
1	2.18E-07	2.18E-07
2	1.66E-07	1.66E-07
3	5.95E-07	7.14E-08
4	NA	4.03E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	6.02E-06	9.52E-06

The results are basically the same as the base case. The CDP is reduced by slightly more than a factor of 1.5 when the unit is cooled down to Mode 4 instead of Mode 5.

Condition B – One Battery/Two Batteries on One Train Inoperable

There is one battery modeled per train in the risk models described in Section 6.3.1. One battery was modeled as inoperable and the POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-21.

Table 6-21	Technical Specification 3.8.4, Condition B One Battery Unavailable	
POS	**Core Damage Probability**	
	Cooldown to POS 3	**Cooldown to POS 4**
1	3.85E-07	3.85E-07
2	2.36E-07	2.36E-07
3	1.78E-06	1.71E-07
4	NA	1.10E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	7.44E-06	1.68E-05

The results are expected to be similar to those for the battery charger. The inoperable battery CDPs are larger than the CDPs for an inoperable battery charger. The battery chargers are the primary source for DC power and the batteries serve as the backup, however, the failure probability for the battery chargers is twice that of the batteries. Therefore, modeling the battery as inoperable results in a larger change in the CDPs. For this case, the CDP is reduced by more than a factor of 2 when the unit is cooled down to Mode 4 instead of Mode 5.

Condition C – One DC Subsystem (train) Inoperable for Reasons Other Than Condition A or B

A DC panel was selected to model the inoperability of one DC train. The risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-22.

Table 6-22 Technical Specification 3.8.4, Condition C One DC Subsystem Unavailable for Reasons Other Than Condition A or B

POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	5.73E-05	5.73E-05
2	6.11E-05	6.11E-05
3	8.05E-06	6.10E-05
4	NA	3.49E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.31E-04	2.19E-04

The CDP values for POS 1, 2, 3, and 4 are larger than those of the previous two cases. This is expected because one entire DC subsystem is inoperable. For this case, the CDP is reduced by a factor of approximately 1.5 when the unit is cooled down to Mode 4 instead of Mode 5. This is a smaller decrease than the other two cases. The POS 4 CDP is greater for this case than the other two cases, however, the CDPs for the POS 1, 2, and 3 are much greater for this case than they are for the previous two cases. The POS 1, 2, and 3 CDPs lessen the effect of the POS 4 contribution.

All three cases presented demonstrate a reduction in risk when the unit is cooled down to Mode 4 instead of Mode 5.

Defense-in-Depth Considerations

The DC power system is designed for the battery chargers to provide the normal DC power. The batteries back up the chargers in the event that AC power to the chargers is lost. There are two redundant trains of DC power so if one is inoperable, the other is available to provide the necessary DC power. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.42 Technical Specification 3.8.7 – Inverters – Operating

Description

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverters is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the RPS and ESFAS. The four inverters [(two per train)] ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized.

Limiting Condition for Operation

The required Train A and Train B inverters shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The inverters are included in the risk models described in Section 6.3.1. Based on the specific design of the unit chosen for the generic PRA model, the two inverter trains are not symmetrically modeled. Train A includes one inverter and Train B includes two inverters. Two cases were evaluated for this Technical Specification. One case models Train A inoperable and the other case models Train B inoperable. The risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4 for each case. The resulting CDPs for each POS are presented in Table 6-23.

Table 6-23	Technical Specification 3.8.7, Inverters – Operating			
POS	Core Damage Probability			
	Train A – One Inverter Unavailable		Train B – Two Inverters Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	4.05E-07	4.05E-07	4.00E-07	4.00E-07
2	2.36E-07	2.36E-07	2.35E-07	2.35E-07
3	1.82E-06	1.58E-07	1.73E-06	1.52E-07
4	NA	1.09E-05	NA	1.04E-05
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	7.50E-06	1.67E-05	7.40E-06	1.62E-05

Although the fault tree modeling is slightly different for the two trains, the results are basically the same. For these two cases, the CDP is reduced by more than a factor of 2 when the unit is cooled down to Mode 4 instead of Mode 5.

Defense-in-Depth Considerations

The inverters can be powered from AC sources or from the batteries. There are two redundant trains of inverters so if one is inoperable, the other is available to provide the necessary AC power. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.43 Technical Specification 3.8.9 – Distribution Systems – Operating

Description

The typical onsite Class 1E AC, DC, and AC vital bus electrical power distribution systems are divided by trains into two redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems.

The AC electrical power subsystem for each train consists of a primary ESF 4.16 kV bus and secondary 480 and 120 V buses, distribution panels, motor control centers and load centers. Each 4.16 kV ESF bus has at least one separate and independent offsite source of power as well as a dedicated onsite DG source. Each 4.16 kV ESF bus is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, a transfer to the alternate offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

The secondary AC electrical power distribution subsystem for each train includes safety related buses, load centers, motor control centers, and distribution panels.

The 120 VAC vital buses are arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the vital buses are Class 1E constant voltage source transformers powered from the same train as the associated inverter. Each constant voltage source transformer is powered from a Class 1E AC bus.

The DC electrical power distribution subsystem typically consists of 125 V buses and distribution panels.

Limiting Condition for Operation

Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The electrical distribution systems are included in the risk models described in Section 6.3.1. Each Condition of this Technical Specification is evaluated quantitatively by modeling the inoperable equipment.

Condition A – One or More AC Electrical Power Distribution Subsystems Inoperable

A representative electrical bus was chosen to provide representative results for one AC electrical power distribution subsystem inoperable. The bus was modeled as inoperable and the risk models described in

Section 6.3.1 were requantified for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-24.

Table 6-24 Technical Specification 3.8.9, Condition A AC Power Distribution Subsystem Unavailable

POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	2.11E-05	2.11E-05
2	2.52E-06	2.52E-06
3	8.17E-06	2.43E-06
4	NA	3.53E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	3.68E-05	6.64E-05

The CDPs for this case are greater than those for the loss of an offsite source (Table 6-16). They are also greater than those for having one DG inoperable (Table 6-17). This is expected because an inoperable onsite source disables the offsite power feed as well as the associated diesel generator. For this case, the CDP is reduced by almost a factor of 2 when the unit is cooled down to Mode 4 instead of Mode 5.

Condition B – One or More AC Vital Buses Inoperable

Two cases are examined for this Technical Specification Condition; one inoperable panel and three inoperable panels to model one or more vital buses. The equipment was modeled as inoperable and the risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-25.

Table 6-25	Technical Specification 3.8.9, Condition B AC Vital Buses Unavailable			
POS	Core Damage Probability			
	One Panel Unavailable		Three Panels Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	4.05E-07	4.05E-07	5.13E-06	5.13E-06
2	2.36E-07	2.36E-07	1.99E-06	1.99E-06
3	1.82E-06	1.58E-07	2.00E-05	1.45E-06
4	NA	1.09E-05	NA	1.05E-04
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	7.50E-06	1.67E-05	3.22E-05	1.19E-04

The CDPs for the three panel case are greater than the one panel case for POSs 1, 2, 3, and 4. For the one panel case, the CDP is reduced by more than a factor of 2 when the unit is cooled down to Mode 4 instead of Mode 5. For the three panel case, the CDP is reduced by more than a factor of 3 when the unit is cooled down to Mode 4 instead of Mode 5.

Condition C – One or More DC Electrical Power Distribution Subsystems Inoperable

This Condition evaluates one DC subsystem being inoperable. If more than one DC subsystem is inoperable, the results will differ, however, the results of Technical Specification 3.8.4 Condition C (Table 6-22) and other DC cases run have demonstrated the same conclusion; the risk of the shutdown process, in terms of CDP, is reduced when the unit is cooled down to Mode 4 instead of Mode 5. Therefore, a separate quantitative risk evaluation was not performed for this Condition.

The risk evaluation cases run for Technical Specification 3.8.9 demonstrate that the risk of the shutdown process, in terms of CDP, is reduced when the unit is cooled down to Mode 4 instead of Mode 5.

<u>Defense-in-Depth Considerations</u>

If Technical Specification 3.8.9 Condition D applies, there is no loss of safety function and the operable electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to mitigate accidents and shut down the reactor and maintain it in a safe shutdown condition. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.5 SENSITIVITY ANALYSIS

6.5.1 POS 4 Time Sensitivity

The time spent in POS 4 was estimated to be 70 hours as described in Section 6.3.1.2. 24 hours were assumed to begin and complete the transition from AFW to RHR cooling as required by the Technical Specifications to be in Mode 5 from Mode 4. The remaining 46 hours were estimated from plant startup data. The number of hours in the POS has an important effect on the predicted CDP for that POS. Because the quantitative evaluations in this report are used to demonstrate an increase in risk by cooling down to POS 4, a sensitivity case is performed to examine the effects of reducing the amount of time assumed for POS 4. 24 hours were chosen for the sensitivity study because the base model assumes that it will take 24 hours to complete the transition from Mode 4 to Mode 5.

Basic events and initiating event probabilities that are dependent on the time assumed for POS 4 were recalculated and the POS 4 fault tree was requantified. Table 6-26 presents the revised POS 4 CDP with the base case results.

Table 6-26 Pos 4 Time Sensitivity Case

Core Damage Probability		
POS	**24 hour POS 4**	**Base Case**
1	2.18E-07	2.18E-07
2	1.66E-07	1.66E-07
3	7.09E-08	7.09E-08
4	2.08E-06	4.03E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	7.57E-06	9.52E-06

The CDP for the 24 hour POS 4 case is 2.08E-06. This is reduced from the base case POS 4 CDP of 4.03E-06 as expected. The base case CDP for POS 3 is 5.95E-07 and it assumes a 49 hour mission time. This provides further evidence that a shutdown to Mode 4 (POS 3) has less risk associated with it than a shutdown to Mode 5 (POS 4).

6.5.2 Steam Generator Tube Rupture Initiating Event Frequency Sensitivity

To address the various types of steam generators installed in the Westinghouse NSSS designed plants, the effects of assuming a larger SGTR initiating event frequency was investigated. As stated in the text following Table 6-8, the SGTR initiating event frequency used was 2.7E-03/yr. This value was increased by a factor of 4 for the base case and the results are provided in Table 6-27.

Table 6-27 Steam Generator Tube Rupture Frequency Sensitivity		
	Core Damage Probability	
POS	Cooldown to POS 3	Cooldown to POS 4
1	2.19E-07	2.19E-07
2	1.86E-07	1.86E-07
3	9.22E-07	9.09E-08
4	NA	4.03E-06
5	6.05E-07	6.05E-07
6	3.65E-06	3.65E-06
7	1.26E-06	1.26E-06
Total	6.84E-06	1.00E-05

POS 3 and POS 5 are the most affected by the increase in the SGTR frequency. The increase is small for the totals. Note that POS 4 does not include the SGTR initiating event and, therefore, is not affected by the sensitivity. Although there is an increase in the core damage probability, the increase is small compared to the change in the initiating event frequency and the changes do not alter the conclusions drawn from the model quantifications presented in Section 6.4.

6.5.3 Probability of Power Recovery Sensitivity

For the loss of offsite power in POS 4, the probability of not recovering power within 2 hours is 0.5. This is based on power recovery information in Reference 6. This probability also includes the probability of equipment failing to start after the power recovery and operator actions. A sensitivity case was run assuming that the probability of not recovering power within 2 hours is 0.1. The revised base case POS 4 results are reduced from a core damage probability of 4.03E-06 to 3.46E-06. This is not a large change in CDP for a factor of 5 reduction in the probability of not recovering power. Although there is a decrease in the POS 4 core damage probability, the decrease is small compared to the change in the probability of not recovering power and the change does not alter the conclusions drawn from the model quantifications presented in Section 6.4.

6.6 TIER 2 AND 3 REQUIREMENTS

Regulatory Guide 1.177 defines a Tier 2 assessment as providing reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out-of-service consistent with the proposed Technical Specification change. The risk-informed evaluations provided in Section 6.4 address one Technical Specification Condition at a time, evaluating a change to the endstate from Mode 5 to Mode 4. The endstate change does not change the Technical Specification requirements for other equipment required to be operable in the shutdown modes. In addition, no completion times are being extended and no surveillance test intervals are being changed. The evaluations in Section 6.4 demonstrate that there is a reduction in risk and advantages in defense-in-depth when the unit cools down to Mode 4 instead of continuing the cooldown to Mode 5. The Mode 4 (POS 3) PRA model includes the auxiliary feedwater turbine-driven pump. The availability of this pump is important to both the qualitative and quantitative bases for determining that the risk associated with a plant shutdown to Mode 4 is less than the risk associated with a plant shutdown to Mode 5. Therefore, a Tier 2 requirement is added to the Technical Specification Bases for the Technical Specifications listed in Table 2-1.

Regulatory Guide 1.177 defines a Tier 3 assessment as a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. 10 CFR 50.65 (a)(4) requires that prior to performing maintenance activities, risk assessments shall be performed to assess and manage the increased risk that may result from proposed maintenance activities. These requirements are applicable to all plant modes. The plant-specific implementation of 10 CFR 50.65 (a)(4) provides assurance that risk-significant plant equipment outage configurations will not occur when equipment in addition to the equipment associated with the Technical Specification Condition that resulted in the shutdown is taken out of service.

6.7 EXTERNAL EVENTS EVALUATION

As noted in Section 6.3.1.3, CCW and SW internal flooding events are included in the PRA model because of their impact on RCP seal LOCAs. Due to reduced RCS pressure, RCP seal LOCAs are not modeled for POS 3, 4, and 5. Therefore, conclusions regarding the effects of flooding on the POS 3 and 4 risk cannot be drawn directly from the quantitative results. By the time the unit has entered POS 3, feedwater system operation has been terminated and secondary cooling is provided by the auxiliary feedwater system. Therefore, feedwater system flooding events are not applicable. The flooding events of interest for POS 3 and 4 involve the CCW and SW systems. The quantitative results presented in Tables 6-14 and 6-15 for CCW and SW, respectively, show that the risk for POS 4 is much greater than that for POS 3 when a train of these systems is inoperable. A partial or total loss of these systems and the subsequent consequences due to flooding events would exhibit a similar difference in risk between POS 3 and POS 4. It is therefore concluded that consideration of flooding events would not change the conclusions presented in Section 6.4.

For fire events, the Appendix R evaluations address the safe shutdown requirements. Operating procedures are in place to respond to fires. The operators will respond to fires occurring in Mode 4 similar to fires occurring in Mode 5. Additionally, more safety system equipment is required to be operable in Mode 4 than in Mode 5 by the Technical Specifications. This provides additional assurance of successful mitigation of fires and other external events.

A seismic PRA is not available for Modes 4 and 5, however, several observations can be made. It is reasonable to assume that the trains of a seismically qualified system will respond to the event in a similar manner for either mode. If one train of a system can still perform its functions after a seismic event, then it is likely that all trains will be able to perform their functions after a seismic event and the defense in depth rational developed for changing the endstate to Mode 4 remains applicable. In addition, a seismic event is likely to lead to a loss of power and there are more independent sources of power required to be available in Mode 4 than in Mode 5. It is concluded that considerations of seismic events would not change the conclusions developed in Section 6.4.

Although not included in the PRA models used to assess the risk for the various plant operating states, external events would not change the conclusions presented in Section 6.4. The risk associated with POS 3 is less than that for POS 4. With more equipment options available for POS 3, there is less likelihood that all redundancy and defense-in-depth will be defeated by an external event in POS 3 than in POS 4.

7 SUMMARY OF RESULTS

An evaluation is performed to assess the risk associated with a cooldown to Mode 4 instead of Mode 5 for specific Technical Specification Required Actions. To perform this evaluation, plant operating states are defined for the changing conditions as a plant shuts down from full power operation and cools down to Mode 5. These POSs are evaluated qualitatively and quantitatively. The qualitative evaluation of the POSs demonstrates that there are advantages in risk and in defense-in-depth when the plant remains in POS 3 (Mode 4) rather than continuing the cooldown to POS 4 (Mode 5). Each Technical Specification listed in Table 2-1 is evaluated for a change in endstate from Mode 5 to Mode 4. These evaluations are presented in Sections 6.4.1 through 6.4.43. Technical Specifications addressing equipment included in the POS PRA models are evaluated quantitatively. In all cases, it is demonstrated that there is an increase in plant risk if the plant cools down to POS 4 (Mode 5) instead of POS 3 (Mode 4). Technical Specifications addressing equipment not included in the POS PRA models are evaluated qualitatively. Defense-in-depth is addressed for each Technical Specification and Sections 6.4.1 through 6.4.43 demonstrate that the endstates for the Technical Specifications listed in Table 2-1 can be changed from Mode 5 to Mode 4. Sensitivity cases are presented in Section 6.5 that examine the influence of the time modeled for POS 4, the effects of assuming a different SGTR initiating event frequency, and a change in the probability of not recovering power for POS 4. In support of these results, Section 6.6 addresses Tier 2 requirements and Section 6.7 qualitatively evaluates external events.

8 CONCLUSIONS

The Technical Specifications listed in Table 2-1 have been evaluated for a change in endstate from Mode 5 to Mode 4. The qualitative evaluation of the POSs in Section 6.2 concludes that there are advantages in risk and in defense-in-depth when the plant remains in POS 3 (Mode 4) rather than continuing to cooldown to POS 4 (Mode 5). Safety margins are not reduced as discussed in Section 5.2. Technical Specification-specific evaluations are presented in Sections 6.4.1 through 6.4.43. These qualitative and quantitative evaluations demonstrate that there is less risk if the endstates for these Technical Specifications are changed from Mode 5 to Mode 4. The model used for the quantitative evaluations is described and shown to be representative of Westinghouse NSSS plants. Defense-in-depth is addressed for each Technical Specification and the evaluations support the conclusion that the endstate can be changed from Mode 5 to Mode 4.

From a sensitivity case presented in Section 6.5, the results demonstrate that changes to the time duration modeled for POS 4 (Mode 5) does not affect the conclusion that there is less risk associated with a cooldown to Mode 4 than there is for a cooldown to Mode 5. Another sensitivity case of assuming a different SGTR initiating event frequency was applied to all POSs and the conclusion that there is less risk associated with a cooldown to Mode 4 than there is for a cooldown to Mode 5 remains applicable. The final sensitivity case examines the effects of changing the probability of not recovering power for POS 4. The results do not change the previously stated conclusions.

In support of the evaluations of risk for the Technical Specifications, Section 6.6 addresses Tier 2 and Tier 3 requirements. A Tier 2 requirement is added to the Technical Specification Bases to ensure the availability of the turbine-driven auxiliary feedwater pump. For Tier 3 requirements, it is concluded that the program used to assess and manage the risk associated with maintenance as required by 10CFR50.65 (a)(4) provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment in addition to the affected Technical Specification is taken out of service.

Section 6.7 qualitatively evaluates external events and concludes that the inclusion of external events into the PRA models for the POSs would not change the conclusions presented in Section 6.4.

The above conclusions support changing the endstate from Mode 5 to Mode 4 for the Technical Specifications listed in Table 2-1 as discussed in Section 6.4 and shown in the markups in Appendix B.

9 REFERENCES

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Rev. 1, November 2002.

2. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998.

3. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.

4. NUREG/CR-6144, Vol. 2, Parts 1A and 1B, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," June 1994.

5. WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

6. EPRI Technical Report 10029987, "Losses of Offsite Power at U.S. Nuclear Power Plants through 2001," April 2002.

7. WCAP-11737, "Low Temperature Overpressurization," March 1989.

APPENDIX A
TOP 100 CUTSETS FOR POS 3 AND POS 4

The top 100 cutsets for POS 3 are presented in Table A-1. These cutsets are from the base case that assumes that no equipment is out of service for testing or maintenance. The time duration for POS 3 is 49 hours. The CDP for the top 100 cutsets is 5.43E-07 which is slightly more than 90% of 5.95E-07, the total CDP calculated for POS 3. The largest initiating event contributor to the POS 3 CDP is Loss of Decay Heat Removal (EFW). This event is the loss of emergency feedwater (also known as AFW) and it accounts for 1.81E-07 (approximately 30%) of the total CDP.

The top 100 cutsets for POS 4 are presented in Table A-2. These cutsets are from the base case that assumes that no equipment is out of service for testing or maintenance. The time duration for POS 4 is 70 hours. The CDP for the top 100 cutsets is 3.89E-06 which is 96% of 4.03E-06, the total CDP calculated for POS 4. The largest initiating event contributor to the POS 4 CDP is Loss of RHR. This event accounts for 2.33E-06 (approximately 58%) of the total CDP.

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
1	%SGR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	1.52E-05	9.73E-08
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAI_1	OPERATOR FAILS TO DIAGNOSE,IDENTIFY, & ISOLATE RUPTURED SG	6.40E-03	
2	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	8.16E-08
	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF	5.44E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
3	%SLO	SMALL LOCA INITIATING EVENT	8.39E-07	6.29E-08
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
4	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	2.05E-08
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	UC-FTIFLTSGCHE	HUMAN ERROR FAILURE TO CLOSE EFW FLOW VALVES TO STEAM GEN C	6.80E-03	
5	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.61E-08
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
6	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	1.56E-08
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	OAT2	OPERATOR FAILS TO TERMINATE SI GIVEN SSB	5.20E-03	
7	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	1.56E-08
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	OAT2	OPERATOR FAILS TO TERMINATE SI GIVEN SSB	5.20E-03	
8	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	1.34E-08
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
9	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	1.15E-08
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
10	%MLO	MEDIUM LOCA INITIATING EVENT	1.12E-08	1.12E-08
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
11	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	1.09E-08
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
12	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.00E-08
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
13	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	9.22E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
14	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	8.98E-09
	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF	5.44E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
15	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	8.01E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	CNU_1	CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)	3.99E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1-SUCCESS	XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS	6.35E-01	
16	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	7.94E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-DG-FR	*1 combination of 2 of 2 DGs*	2.34E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
17	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	7.52E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
18	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	6.53E-09
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW8P		5.00E-01	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
19	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	6.02E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	UCAVIFV3551-FC	FAILURE TO CLOSE IFV-3551 WHICH SUPPLIES MD HEADER FLOW	2.00E-03	
20	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	6.02E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	UCAVIFV3556-FC	FAILURE TO CLOSE IFV-3556 WHICH SUPPLIES TD HEADER FLOW	2.00E-03	
21	%SGR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	1.52E-05	5.81E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAD_1	OPERATOR FAILS TO DEPRESSURIZE SECONDARY SIDE (NORMAL COOL DOWN)	5.10E-03	
	OAESF3		7.50E-02	
22	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.51E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	CNU_1	CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)	3.99E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1-SUCCESS	XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS	6.35E-01	
23	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	4.89E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	CNU_4	CORE IS UNCOVERED AT 4 HOURS (WITH RCS COOLDOWN)	2.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
24	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	4.49E-09
	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF	5.44E-06	
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	OAESF3		7.50E-02	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
25	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	4.45E-09
	1HR_1-SUCCESS	1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR	6.32E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
26	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	4.05E-09
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW8P		5.00E-01	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
27	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	3.36E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	CNU_4	CORE IS UNCOVERED AT 4 HOURS (WITH RCS COOLDOWN)	2.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
28	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	3.13E-09
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	OAT2	OPERATOR FAILS TO TERMINATE SI GIVEN SSB	5.20E-03	
29	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	3.13E-09
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	OAT2	OPERATOR FAILS TO TERMINATE SI GIVEN SSB	5.20E-03	
30	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	3.06E-09
	1HR_1-SUCCESS	1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR	6.32E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
31	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	2.85E-09
	D-AV-IFV2030FO	AIR-OP FLOW CONTROL IFV-2030FAILS TO OPEN DUE TO LOCAL FLT	1.30E-03	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
32	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	2.50E-09
	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF	5.44E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	Q-SIRWSTLOLOFA	NO SAFGUARDS ACTUATION SIGNAL (RWST LO-LO LEVEL)	4.60E-04	
33	%SGR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	1.52E-05	2.28E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	UAAVIFV3531-FC	FAILURE TO CLOSEIFV-3531 WHICH SUPPLIES MD HEADER FLOW	2.00E-03	
34	%SGR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT	1.52E-05	2.28E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
	UAAVIFV3536-FC	FAILURE TO CLOSEIFV-3536 WHICH SUPPLIES TD HEADER FLOW	2.00E-03	
35	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	2.10E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC2	XVM-2801A AND -2801B FAIL DUE TO COMMON CAUSE	6.99E-04	
	OAESF3		7.50E-02	
36	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	2.10E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC3	XVM-2801B AND -2801C FAIL DUE TO COMMON CAUSE	6.99E-04	
	OAESF3		7.50E-02	
37	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	2.10E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC1	XVM-2801A AND -2801B FAIL DUE TO COMMON CAUSE	6.99E-04	
	OAESF3		7.50E-02	
38	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	2.10E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC2	XVM-2801A AND -2801B FAIL DUE TO COMMON CAUSE	6.99E-04	
	OAESF3		7.50E-02	
39	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	2.10E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC3	XVM-2801B AND -2801C FAIL DUE TO COMMON CAUSE	6.99E-04	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	OAESF3		7.50E-02	
40	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.82E-09
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	DBPMI-XPP21BFR	MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	XPP21AB8		2.50E-01	
41	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT	4.01E-05	1.80E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC4	XVM-2801A, -2801B, AND -2801C FAIL DUE TO COMMON CAUSE	5.98E-04	
	OAESF3		7.50E-02	
42	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	1.80E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	MSIV-CC-FC4	XVM-2801A, -2801B, AND -2801C FAIL DUE TO COMMON CAUSE	5.98E-04	
	OAESF3		7.50E-02	
43	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.77E-09
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
44	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.77E-09
	D-AV-IFV2030FO	AIR-OP FLOW CONTROL IFV-2030FAILS TO OPEN DUE TO LOCAL FLT	1.30E-03	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
45	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.62E-09
	D-VLVMISPOS-HE	HUMAN ERROR FAIL TO RESTORE VALVE SETTINGS AFTER TEST	7.40E-04	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
46	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.48E-09

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	DBPMI-XPP21BFR	MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW8P		5.00E-01	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	XPP21AB		5.00E-01	
47	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.48E-09
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	EFWICC-XPP-FR03	XPP21B, XPP8 FAIL TO RUN BY CCF	1.21E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
48	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.48E-09
	DBPMI-XPP21BFR	MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	EFWICC-XPP-FR02	XPP21A, XPP8 FAIL TO RUN BY CCF	1.21E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
49	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.40E-09
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	EFT-CC-SACV-FO4	SPRING ASSISTED CHECK VALVES XVC-1009A, -1009B, -1009C CCF TO OPEN	1.15E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
50	%LLO	LARGE LOCA INITIATING EVENT	1.40E-09	1.40E-09
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
51	%SLO	SMALL LOCA INITIATING EVENT	8.39E-07	1.38E-09
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
52	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.31E-09
	ALLTRIPS-ATWS-PF	INITIATORS THAT RESULT IN A PARTIAL FLOW ATWS	1.00E+00	
	D-MVSPURSIGNFA	DURG TDP OPER SPUR SIGN TO ISLXTIE SUPPL STM CLOSE 2802A&B	6.00E-04	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
53	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	1.27E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	DBPT----XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
54	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT	4.01E-05	1.22E-09
	EAAVXVM2801AFC	FAILURE TO ISOL MS FLOW FROM SG A, XVM-2801A FAILS TO CLOSE	4.49E-03	
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	UA-FTIFLTSGAHE	HUMAN ERROR FAILURE TO CLOSEEFW FLOW VALVES TO STEAM GEN A	6.80E-03	
55	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	1.18E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-BUSBK-FO	1 combination of 2 of 2 Bus feeder breakers	3.00E-04	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
56	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.10E-09
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
57	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.10E-09
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	EFW-CC-XVC-F16	CV XVC 1014, 1016, 1013A, 1013B, 1048A, 1048B FAIL BY CCF	9.01E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
58	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	1.02E-09
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	ACP-CC-BUSBK-FO	1 combination of 2 of 2 Bus feeder breakers	3.00E-04	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
59	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	1.01E-09
	D-VLVMISPOS-HE	HUMAN ERROR FAIL TO RESTORE VALVE SETTINGS AFTER TEST	7.40E-04	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
60	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	9.65E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-BUSBK-FO	1 combination of 2 of 2 Bus feeder breakers	3.00E-04	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
61	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	8.85E-10
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAESF3		7.50E-02	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
62	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	8.82E-10
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	ESF-CC-720BU-FOR	1 combination of 2 of 2 7200 VAC buses	1.20E-07	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
63	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	8.73E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	DBPT----XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
64	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	8.60E-10
	1HR_1-SUCCESS	1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR	6.32E-01	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	OAQ_1	OPERATOR FAILS TO RESTORE EQUIPMENT AFTER SBO & RECOVERY OF OFFSITE POWER	2.90E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
65	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	8.46E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	BBPM--XPP39BFS	LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START	3.68E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
66	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	8.16E-10
	ALLTRIPS-ATWS-PF	INITIATORS THAT RESULT IN A PARTIAL FLOW ATWS	1.00E+00	
	D-MVSPURSIGNFA	DURG TDP OPER SPUR SIGN TO ISLXTIE SUPPL STM CLOSE 2802A&B	6.00E-04	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
67	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	7.28E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	BBPM--XPP39BFS	LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START	3.68E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
68	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	7.18E-10
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW8P		5.00E-01	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	

Table A-1 POS 3 Top 100 Cutsets				
Cutset Number	Events	Description	Event Probability	Cutset Probability
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
69	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	7.18E-10
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAESF3		7.50E-02	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
70	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	7.06E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-BUSBK-FO	1 combination of 2 of 2 Bus feeder breakers	3.00E-04	
	CNU_1	CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)	3.99E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1-SUCCESS	XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS	6.35E-01	
71	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	6.90E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	BBPM--XPP39BFS	LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START	3.68E-03	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
72	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	6.89E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AACB-----DGAFC	DIESEL GENERATORBREAKER FAILS TOCLOSE	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
73	%LSP	LOSS OF OFFSITE POWER INITIATING.EVENT	1.38E-04	6.89E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AACB--XSW1DAFO	BUS XSW1DA FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
74	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	6.89E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB-----DGBFC	DIESEL GENERATOR BREAKER FAILS TOCLOSE	3.00E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
75	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	6.89E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB--XSW1DBFO	BUS XSW1DB FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
76	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	6.44E-10
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	D-AV-IFV2030FO	AIR-OP FLOW CONTROL IFV-2030FAILS TO OPEN DUE TO LOCAL FLT	1.30E-03	
	DBPMI-XPP21BFR	MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
	XPP21AB		5.00E-01	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
77	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.93E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AACB-----DGAFC	DIESEL GENERATORBREAKER FAILS TOCLOSE	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
78	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.93E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AACB--XSW1DAFO	BUS XSW1DA FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
79	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.93E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB-----DGBFC	DIESEL GENERATOR BREAKER FAILS TOCLOSE	3.00E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
80	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.93E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB--XSW1DBFO	BUS XSW1DB FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	3.65E-01	
81	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.92E-10
	1HR_1-SUCCESS	1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR	6.32E-01	
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	OAQ_1	OPERATOR FAILS TO RESTORE EQUIPMENT AFTER SBO & RECOVERY OF OFFSITE POWER	2.90E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
82	%ISL	INTERFACING SYSTEMS LOCA INITIATING EVENT	7.61E-09	5.71E-10
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	OAESF3		7.50E-02	
83	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.63E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AACB-----DGAFC	DIESEL GENERATORBREAKER FAILS TOCLOSE	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
84	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.63E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AACB--XSW1DAFO	BUS XSW1DA FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
85	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.63E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB-----DGBFC	DIESEL GENERATOR BREAKER FAILS TOCLOSE	3.00E-03	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	

Table A-1 POS 3 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
86	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.63E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB--XSW1DBFO	BUS XSW1DB FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)	3.46E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
87	%VRP	REACTOR VESSEL RUPTURE INITIATING EVENT	5.59E-10	5.59E-10
88	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	5.50E-10
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAESF3		7.50E-02	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
89	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	5.44E-10
	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF	5.44E-06	
	F-CVXVC08926FO	XVC-8926 FAILS TO OPEN	1.00E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
90	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.15E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW8P		5.00E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
91	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.05E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	BBPM--XPP39BFS	LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START	3.68E-03	

Table A-1	POS 3 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	CNU_1	CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)	3.99E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_1-SUCCESS	XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS	6.35E-01	
92	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	5.02E-10
	DAPMI-XPP21AFR	MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS	8.13E-03	
	DBPM--XPP21BFS	MDP XPP-21B FAILS TO START DUE TO LOCAL FAULTS	1.12E-03	
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFW21P		5.00E-01	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))	1.50E-02	
93	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	5.01E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	OAQ_1	OPERATOR FAILS TO RESTORE EQUIPMENT AFTER SBO & RECOVERY OF OFFSITE POWER	2.90E-03	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
94	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	4.94E-10
	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE	7.35E-03	
	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.46E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	Q-SIRWSTLOLOFA	NO SAFGUARDS ACTUATION SIGNAL (RWST LO-LO LEVEL)	4.60E-04	
95	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	4.89E-10
	1HR_1-SUCCESS	1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR	6.32E-01	
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
96	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	4.73E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-INV-FOR	1 combination of 2 of 2 Inverters	1.20E-04	

Table A-1	*POS 3 Top 100 Cutsets*			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
97	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	4.69E-10
	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF	5.44E-06	
	HPI-CC-PM43-FS	1 combination of 2 of 2 pumps	8.62E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
98	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.38E-04	4.57E-10
	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR	3.68E-01	
	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR	1.83E-01	
	ACP-CC-DGOKR-FC	1 combination of 2 of 2 DG output breakers	1.16E-04	
	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS	4.24E-01	
99	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	4.46E-10
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW8P		5.00E-01	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
	OARC	OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)	1.50E-01	
100	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT	1.00E+00	4.46E-10
	DBPT----XPP8FS	TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE	5.96E-03	
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)	1.00E+00	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OAESF3		7.50E-02	
	OAH_1	OPERATOR FAILS TO ALIGN CCW TO RHR HXs	1.10E-02	
		Total CDP		5.43E-07

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
1	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.64E-07
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
2	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	3.50E-07
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
	SIPOS4HE	OPERATOR ACTION TO ACTUATE HPSI	4.10E-03	
3	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	3.35E-07
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
4	%DIL	BORON DILUTION EVENT IN POS 4	4.20E-03	3.12E-07
	OAE_1	OPERATOR FAILS TO IMPLEMENT EMERGENCY BORATION	3.10E-04	
	POS4OADILTM	OPERATOR FAILS TO TERMINATE DILUTION EVENT	5.00E-04	
	OAE_1D	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO IMPLEMENT EMERGENCY BORATION	4.80E+02	
5	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.36E-07
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4CC-PM31-FR	RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT	2.27E-04	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
6	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	2.30E-07
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	

Table A-2 POS 4 Top 100 Cutsets				
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
7	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.12E-07
	POS4OAISOLEAK	OPERATOR ACTION TO ISOLATE LEAK	1.00E-02	
	SIPOS4HE	OPERATOR ACTION TO ACTUATE HPSI	4.10E-03	
	SIPOS4HE-D	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE HPSI	3.70E+01	
8	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.10E-07
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
9	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	9.65E-08
	CAMVXVB9503AFO	MOTOR-OPERATED VALVE XVB-9503A FAILS TO OPEN	6.16E-04	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
10	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	9.65E-08
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IAMVXVG8811AFO	VALVE FAILS TO OPEN	6.16E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
11	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	9.65E-08
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IAMVXVG8812AFO	VALVE FAILS TO OPEN	6.16E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
12	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	9.31E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OARHRREC-L	RHR RECOVERY DURING BLEED AND FEED	1.00E-01	
	POS4CC-PM31-FR	RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT	2.27E-04	
13	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.00E-08
	ESF-CC-720BU-FOR	1 combination of 2 of 2 7200 VAC buses	1.20E-07	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
14	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	5.89E-08
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	GAPMXPP0031AFS	XPP-31A FAILS TO START	3.76E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
15	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	4.61E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO01	XVG8811A, XVG8811B FAIL TO OPEN BY CCF	4.44E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
16	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	4.61E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO03	XVG8811A, XVG8812B FAIL TO OPEN BY CCF	4.44E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
17	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	4.61E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	LPR-CCMVFO08	XVG8811B, XVG8812A FAIL TO OPEN BY CCF	4.44E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
18	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	4.61E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO11	XVG8812A, XVG8812B FAIL TO OPEN BY CCF	4.44E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
19	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.32E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4CC-PM31-FR	RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT	2.27E-04	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
20	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	3.16E-08
	OAR2	OPERATOR FAILS TO ALIGN FOR LP CL RECIRCULATION	3.70E-04	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
21	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.05E-08
	CAMVXVB9503AFO	MOTOR-OPERATED VALVE XVB-9503A FAILS TO OPEN	6.16E-04	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
22	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.05E-08
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	IAMVXVG8811AFO	VALVE FAILS TO OPEN	6.16E-04	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
23	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.05E-08
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IAMVXVG8812AFO	VALVE FAILS TO OPEN	6.16E-04	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
24	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.82E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO04	XVG8811A, XVG8811B, XVG8812A FAIL TO OPEN BY CCF	2.71E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
25	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.82E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO05	XVG8811A, XVG8811B, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
26	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.82E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO06	XVG8811A, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	

Table A-2 POS 4 Top 100 Cutsets				
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
27	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.82E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO10	XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
28	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.47E-08
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	HPR-CC-PM31-FS	1 combination of 2 of 2 pumps	2.38E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
29	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	2.11E-08
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	BBPM--XPP39BFS	LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START	3.68E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
30	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.86E-08
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	GAPMXPP0031AFS	XPP-31A FAILS TO START	3.76E-04	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
31	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	1.48E-08
	ACP-CC-BUSBK-FO	1 combination of 2 of 2 Bus feeder breakers	3.00E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
	PWRRECD	FAILURE TO RESET BREAKER	5.00E-01	
32	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.20E-08
	ESF-CC-720BU-FOR	1 combination of 2 of 2 7200 VAC buses	1.20E-07	

Table A-2 POS 4 Top 100 Cutsets				
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	OARHRREC-L	*RHR RECOVERY DURING BLEED AND FEED*	*1.00E-01*	
33	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	1.18E-08
	ACP-CC-INV-FOR	1 combination of 2 of 2 Inverters	1.20E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
34	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.06E-08
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	OAH_1IE	OPERATOR FAILS TO ALIGN CCW TO RHR HXs (NORMAL TRANSITION IN MODE 4)	1.17E-03	
	OARHRREC-L	RHR RECOVERY DURING BLEED AND FEED	1.00E-01	
35	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	9.34E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	POS4XPP0031BFR	RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT	2.12E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
36	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	8.62E-09
	AACB-----DGAFC	DIESEL GENERATORBREAKER FAILS TOCLOSE	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
	PWRRECD	FAILURE TO RESET BREAKER	5.00E-01	
37	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	8.62E-09
	AACB--XSW1DAFO	BUS XSW1DA FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
	PWRRECD	FAILURE TO RESET BREAKER	5.00E-01	
38	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	8.62E-09
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB-----DGBFC	DIESEL GENERATOR BREAKER FAILS TOCLOSE	3.00E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
	PWRRECD	FAILURE TO RESET BREAKER	5.00E-01	
39	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	8.62E-09
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABCB--XSW1DBFO	BUS XSW1DB FEEDER BREAKER FAILS TO OPEN	3.00E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
	PWRRECD	FAILURE TO RESET BREAKER	5.00E-01	
40	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	8.54E-09
	F-CVXVC08926FO	XVC-8926 FAILS TO OPEN	1.00E-04	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
41	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	8.27E-09
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFS	DIESEL GENERATORFAILS TO START DUE TO RANDOM FAULTS	1.44E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
42	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	8.27E-09
	AADG-----DGAFS	DIESEL GENERATORFAILS TO START DUE TO RANDOM FAULTS	1.44E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
43	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	7.52E-09
	AACT-XSW1DA1OP	7200/480 VAC TRANSFORMER FAILS DURING OPERATION	4.80E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
44	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	7.52E-09
	AACT-XSW1DA2OP	7200/480 VAC TRANSFORMER FAILS DURING OPERATION	4.80E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	

Table A-2	POS 4 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
45	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	7.36E-09
	HPI-CC-PM43-FS	1 combination of 2 of 2 pumps	8.62E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
46	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.96E-09
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO02	XVG8811A, XVG8812A FAIL TO OPEN BY CCF	4.44E-05	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
47	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	6.64E-09
	HPR-CC-PM31-FR	1 combination of 2 of 2 pumps	7.77E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
48	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.49E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO01	XVG8811A, XVG8811B FAIL TO OPEN BY CCF	4.44E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
49	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.49E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO03	XVG8811A, XVG8812B FAIL TO OPEN BY CCF	4.44E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
50	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.49E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO08	XVG8811B, XVG8812A FAIL TO OPEN BY CCF	4.44E-05	

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
51	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	6.49E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO11	XVG8812A, XVG8812B FAIL TO OPEN BY CCF	4.44E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
52	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	5.71E-09
	ACP-CC-DGOKR-FC	1 combination of 2 of 2 DG output breakers	1.16E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
	PWRRECD	FAILURE TO RESET BREAKER	5.00E-01	
53	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	5.49E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO07	XVG8811A, XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	5.28E-06	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
54	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	4.92E-09
	ACP-CC-7248T-SO	1 combination of 2 of 2 7200/480 transformers	2.40E-06	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
55	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	4.08E-09
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	ACP-CC-TRFM-FOR	1 combination of 2 of 2 transformers	2.40E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
56	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.96E-09

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO04	XVG8811A, XVG8811B, XVG8812A FAIL TO OPEN BY CCF	2.71E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
57	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.96E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO05	XVG8811A, XVG8811B, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
58	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.96E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO06	XVG8811A, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
59	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.96E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO10	XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
60	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	3.79E-09
	LPR-CCMVFO01	XVG8811A, XVG8811B FAIL TO OPEN BY CCF	4.44E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
61	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	3.79E-09
	LPR-CCMVFO03	XVG8811A, XVG8812B FAIL TO OPEN BY CCF	4.44E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
62	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	3.79E-09

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	LPR-CCMVFO08	XVG8811B, XVG8812A FAIL TO OPEN BY CCF	4.44E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
63	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	3.79E-09
	LPR-CCMVFO11	XVG8812A, XVG8812B FAIL TO OPEN BY CCF	4.44E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
64	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.76E-09
	AACBXMC1DA2YCO	480 VAC BREAKER TRANSFERS OPEN DURING OPERATION	2.40E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
65	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.76E-09
	AACB-XSW1DA1CO	480 VAC BREAKER TRANSFERS OPEN DURING OPERATION	2.40E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
66	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.76E-09
	AACB-XSW1DA2CO	480 VAC BREAKER TRANSFERS OPEN DURING OPERATION	2.40E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
67	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.76E-09
	AACB--XSW1EACO	7200 VAC BKR XSW1EA XFERS OPEN DURING OPS	2.40E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	

Table A-2 POS 4 Top 100 Cutsets				
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	1.40E+03	
68	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.69E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OARHRREC-L	RHR RECOVERY DURING BLEED AND FEED	1.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	POS4XPP0031BFR	RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT	2.12E-03	
69	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	3.54E-09
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	BBMVXVB3116BFO	LOCAL FAULTS OF MOTOR-OPERATED VALVE XVB-3116B	6.16E-04	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
70	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	3.48E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	HPR-CC-PM31-FS	1 combination of 2 of 2 pumps	2.38E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D3	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	3.10E+00	
71	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.81E-09
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	ACP-CC-TRFM-FOR	1 combination of 2 of 2 transformers	2.40E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
72	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.71E-09
	CBMVXVB9503BFO	MOTOR-OPERATED VALVE XVB-9503B FAILS TO OPEN	6.16E-04	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
73	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.71E-09

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IBMVXVG8811BFO	VALVE FAILS TO OPEN	6.16E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
74	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.71E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IBMVXVG8812BFO	VALVE FAILS TO OPEN	6.16E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
75	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.56E-09
	HPR-CCMVFO03	XVG8706A, XVG8706B, XVG8885 FAIL BY CCF	3.00E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
76	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.45E-09
	CBPM---XPP1BFS	PUMP XPP-1B FAILS TO START	5.55E-04	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
77	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.38E-09
	AACT-XSW1DA1OP	7200/480 VAC TRANSFORMER FAILS DURING OPERATION	4.80E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	

Table A-2 POS 4 Top 100 Cutsets

Cutset Number	Events	Description	Event Probability	Cutset Probability
78	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.38E-09
	AACT-XSW1DA2OP	7200/480 VAC TRANSFORMER FAILS DURING OPERATION	4.80E-05	
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
79	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.31E-09
	LPR-CCMVFO04	XVG8811A, XVG8811B, XVG8812A FAIL TO OPEN BY CCF	2.71E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
80	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.31E-09
	LPR-CCMVFO05	XVG8811A, XVG8811B, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
81	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.31E-09
	LPR-CCMVFO06	XVG8811A, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
82	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.31E-09
	LPR-CCMVFO10	XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF	2.71E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
83	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.20E-09
	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B	4.20E-03	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	LPR-CCMVFO02	XVG8811A, XVG8812A FAIL TO OPEN BY CCF	4.44E-05	
	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV	2.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV	2.50E+02	
84	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.06E-09
	EFW-CC-PM21-FS	1 combination of 2 of 2 MD pumps to start	9.07E-05	
	OARHRREC-L	RHR RECOVERY DURING BLEED AND FEED	1.00E-01	
	POS4CC-PM31-FR	RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT	2.27E-04	
85	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	2.04E-09

Table A-2 POS 4 Top 100 Cutsets				
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	ESF-CC-720BK-SO	1 combination of 2 of 2 7200 VAC feed breakers	1.20E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
86	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	2.03E-09
	HPR-CC-PM31-FS	1 combination of 2 of 2 pumps	2.38E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
87	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	1.94E-09
	HPHLRCCMVFO07	XVG8884, XVG8886, XVG8706A, XVG8706B FAIL BY CCF	2.27E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
88	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	1.94E-09
	HPI-CCMVFO07	LCV115B, LCV115D, XVG8801A, XVG8801B FAIL BY CCF	2.27E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
89	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4	1.40E-04	1.94E-09
	HPR-CCMVFC07	LCV115B, LCV115D, XVG8809A, XVG8809B FAIL BY CCF	2.27E-05	
	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE	6.10E-01	
90	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.66E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	SWS-CC-STR-BL04	TRAVELING SCREENS XRS-2A, 2B, 2C FAIL BY CCF	1.60E-06	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
91	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.66E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	GBPMXPP0031BFS	XPP-31B FAILS TO START	3.76E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
92	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.64E-09
	CCW-CCPMFR04	XPP1A, XPP1B, XPP1C FAIL TO RUN BY CCF	1.58E-06	

Table A-2	POS 4 Top 100 Cutsets			
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
93	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.57E-09
	CBPM---XPP1BFR	PUMP XPP-1B FAILS TO RUN - RANDOM	3.55E-04	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT	4.24E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
94	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.40E-09
	ACP-CC-DG-FR	1 combination of 2 of 2 DGs	2.34E-03	
	ESF-CC-720BK-SO	1 combination of 2 of 2 7200 VAC feed breakers	1.20E-06	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
95	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.39E-09
	EFW-CC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF	1.19E-05	
	OAH_1IE	OPERATOR FAILS TO ALIGN CCW TO RHR HXs (NORMAL TRANSITION IN MODE 4)	1.17E-03	
	OARHRREC-L	RHR RECOVERY DURING BLEED AND FEED	1.00E-01	
96	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	1.38E-09
	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS	5.83E-02	
	ABIV-XIT5903OP	INVERTER XIT-5903 FAILS DURING OPERATION	2.40E-03	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	NVERT1		1.00E-01	
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
97	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT	1.97E-04	1.38E-09
	AAIV-XIT5901OP	INVERTER XIT-5901 FAILS DURING OPERATION	2.40E-03	
	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS	5.83E-02	
	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER	1.00E+00	
	NVERT1		1.00E-01	

Table A-2 POS 4 Top 100 Cutsets				
Cutset Number	**Events**	**Description**	**Event Probability**	**Cutset Probability**
	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS	5.00E-01	
98	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.36E-09
	CAMVXVB9503AFO	MOTOR-OPERATED VALVE XVB-9503A FAILS TO OPEN	6.16E-04	
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031BFR	RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT	2.12E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
99	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.36E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IAMVXVG8811AFO	VALVE FAILS TO OPEN	6.16E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031BFR	RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT	2.12E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
100	%LRHR	LOSS OF RHR INITIATING EVENT	1.00E+00	1.36E-09
	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS	4.10E-03	
	IAMVXVG8812AFO	VALVE FAILS TO OPEN	6.16E-04	
	OAB2-HS	INITIATE FEED AND BLEED	1.30E-02	
	OARHRREC	FAILURE TO RECOVER RHR	5.00E-01	
	POS4XPP0031BFR	RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT	2.12E-03	
	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED	3.90E+01	
		Total CDP		3.89E-06

APPENDIX B
MARKED-UP TECHNICAL SPECIFICATIONS AND BASES

The marked-up Technical Specifications are from TSTF-432 in Technical Specifications Task Force letter TSTF-09-25, December 22, 2009.

Technical Specification Task Force
Improved Standard Technical Specifications Change Traveler

Change in Technical Specifications End States (WCAP-16294)

NUREGs Affected: ☐ 1430 ☑ 1431 ☐ 1432 ☐ 1433 ☐ 1434

Classification 1) Technical Change — Recommended for CLIIP?: Yes

Correction or Improvement: Improvement — NRC Fee Status: Exempt

Benefit: Shortens Outages

Industry Contact: Ken Schrader, (805) 545-4328, kjse@pge.com

See Attached.

Revision History

OG Revision 0 Revision Status: Active

Revision Proposed by: WOG

Revision Description:
Original Issue.

Owners Group Review Information

Date Originated by OG: 25-Nov-09

Owners Group Comments
(No Comments)

Owners Group Resolution: Approved Date: 14-Dec-09

TSTF Review Information

TSTF Received Date: 25-Nov-09 Date Distributed for Review 14-Dec-09

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF Comments:
(No Comments)

TSTF Resolution: Approved Date: 22-Dec-09

NRC Review Information

NRC Received Date: 22-Dec-09

Affected Technical Specifications

Ref. 3.3.2 Bases ESFAS Instrumentation

Action 3.3.2.B	ESFAS Instrumentation
Action 3.3.2.B Bases	ESFAS Instrumentation
Action 3.3.2.C	ESFAS Instrumentation
Action 3.3.2.C Bases	ESFAS Instrumentation
Action 3.3.2.D Bases	ESFAS Instrumentation
Action 3.3.2.E Bases	ESFAS Instrumentation
Action 3.3.2.G Bases	ESFAS Instrumentation
Action 3.3.2.H Bases	ESFAS Instrumentation
Action 3.3.2.I Bases	ESFAS Instrumentation
Action 3.3.2.J Bases	ESFAS Instrumentation
Action 3.3.2.K	ESFAS Instrumentation
Action 3.3.2.K Bases	ESFAS Instrumentation
SR 3.3.2.2 Bases	ESFAS Instrumentation
SR 3.3.2.4 Bases	ESFAS Instrumentation
SR 3.3.2.5 Bases	ESFAS Instrumentation
SR 3.3.2.10 Bases	ESFAS Instrumentation
Ref. 3.3.7 Bases	Control Room Emergency Filtration System Actuation Instrumentation
Action 3.3.7.C	Control Room Emergency Filtration System Actuation Instrumentation
Action 3.3.7.C Bases	Control Room Emergency Filtration System Actuation Instrumentation
SR 3.3.7.5 Bases	Control Room Emergency Filtration System Actuation Instrumentation
SR 3.3.7.6 Bases	Control Room Emergency Filtration System Actuation Instrumentation
Ref. 3.3.8 Bases	Fuel Building Air Cleanup System Actuation Instrumentation
Action 3.3.8.D	Fuel Building Air Cleanup System Actuation Instrumentation
Action 3.3.8.D Bases	Fuel Building Air Cleanup System Actuation Instrumentation
Ref. 3.4.13 Bases	RCS Operational Leakage

Action 3.4.13.B	RCS Operational Leakage
Action 3.4.13.B Bases	RCS Operational Leakage
SR 3.4.13.2 Bases	RCS Operational Leakage
Ref. 3.4.14 Bases	RCS Pressure Isolation Valve Leakage
Action 3.4.14.B	RCS Pressure Isolation Valve Leakage
Action 3.4.14.B Bases	RCS Pressure Isolation Valve Leakage
SR 3.4.14.1 Bases	RCS Pressure Isolation Valve Leakage
Ref. 3.4.15 Bases	RCS Leakage Detection Instrumentation
Action 3.4.15.E	RCS Leakage Detection Instrumentation
Action 3.4.15.E Bases	RCS Leakage Detection Instrumentation
Ref. 3.5.3 Bases	Emergency Core Cooling System - Shutdown Change Description: Deleted
Action 3.5.3.A	Emergency Core Cooling System - Shutdown
Action 3.5.3.A Bases	Emergency Core Cooling System - Shutdown
Action 3.5.3.B	Emergency Core Cooling System - Shutdown Change Description: Deleted
Action 3.5.3.B Bases	Emergency Core Cooling System - Shutdown Change Description: Deleted
Action 3.5.3.C	Emergency Core Cooling System - Shutdown Change Description: Deleted
Action 3.5.3.C Bases	Emergency Core Cooling System - Shutdown Change Description: Deleted
Ref. 3.5.4 Bases	Refueling Water Storage Tank
Action 3.5.4.C	Refueling Water Storage Tank
Action 3.5.4.C Bases	Refueling Water Storage Tank
Ref. 3.6.6B Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
Ref. 3.6.6A Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
Ref. 3.6.6C Bases	Containment Spray System (Ice Condenser)
Ref. 3.6.6D Bases	Quench Spray System (Subatmospheric)

Ref. 3.6.6E Bases	Recirculation Spray System (Subatmospheric)
Action 3.6.6A.B	Containment Spray and Cooling Systems (Atmospheric and Dual)
Action 3.6.6C.B	Containment Spray System (Ice Condenser)
Action 3.6.6D.B	Quench Spray System (Subatmospheric)
Action 3.6.6A.B Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
Action 3.6.6C.B Bases	Containment Spray System (Ice Condenser)
Action 3.6.6D.B Bases	Quench Spray System (Subatmospheric)
Action 3.6.6A.E	Containment Spray and Cooling Systems (Atmospheric and Dual)
Action 3.6.6A.E Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
Action 3.6.6B.F	Containment Spray and Cooling Systems (Atmospheric and Dual)
Action 3.6.6E.F	Recirculation Spray System (Subatmospheric)
Action 3.6.6B.F Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
Action 3.6.6E.F Bases	Recirculation Spray System (Subatmospheric)
SR 3.6.6C.2 Bases	Containment Spray System (Ice Condenser)
SR 3.6.6D.2 Bases	Quench Spray System (Subatmospheric)
SR 3.6.6.4 Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
Ref. 3.6.6B.4 Bases	Containment Spray and Cooling Systems (Atmospheric and Dual)
SR 3.6.6E.5 Bases	Recirculation Spray System (Subatmospheric)
Ref. 3.6.7 Bases	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)
Action 3.6.7.B	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)
Action 3.6.7.B Bases	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)
Ref. 3.6.11 Bases	Iodine Cleanup System (Atmospheric and Subatmospheric)
Action 3.6.11.B	Iodine Cleanup System (Atmospheric and Subatmospheric)

Action 3.6.11.B Bases	Iodine Cleanup System (Atmospheric and Subatmospheric)
Ref. 3.6.12 Bases	Vacuum Relief Valves (Atmospheric and Ice Condenser)
Action 3.6.12.B	Vacuum Relief Valves (Atmospheric and Ice Condenser)
Action 3.6.12.B Bases	Vacuum Relief Valves (Atmospheric and Ice Condenser)
SR 3.6.12.1 Bases	Vacuum Relief Valves (Atmospheric and Ice Condenser)
Action 3.6.13.B	Shield Building Air Cleanup System (Dual and Ice Condenser)
Action 3.6.13.B Bases	Shield Building Air Cleanup System (Dual and Ice Condenser)
Ref. 3.6.14 Bases	Air Return System (Ice Condenser)
Action 3.6.14.B	Air Return System (Ice Condenser)
Action 3.6.14.B Bases	Air Return System (Ice Condenser)
Ref. 3.6.18 Bases	Containment Recirculation Drains (Ice Condenser)
Action 3.6.18.C	Containment Recirculation Drains (Ice Condenser)
Action 3.6.18.C Bases	Containment Recirculation Drains (Ice Condenser)
LCO 3.7.5 Bases	Auxiliary Feedwater System
Ref. 3.7.7 Bases	Component Cooling Water System
Action 3.7.7.B	Component Cooling Water System
Action 3.7.7.B Bases	Component Cooling Water System
Ref. 3.7.8 Bases	Service Water System
Action 3.7.8.B	Service Water System
Action 3.7.8.B Bases	Service Water System
Ref. 3.7.9 Bases	Ultimate Heat Sink
Action 3.7.9.C	Ultimate Heat Sink
Action 3.7.9.C Bases	Ultimate Heat Sink
Ref. 3.7.10 Bases	Control Room Emergency Filtration System
Action 3.7.10.C	Control Room Emergency Filtration System
Action 3.7.10.C Bases	Control Room Emergency Filtration System
SR 3.7.10.3 Bases	Control Room Emergency Filtration System

SR 3.7.10.4 Bases	Control Room Emergency Filtration System
SR 3.7.10.10 Bases	Control Room Emergency Filtration System
Ref. 3.7.11 Bases	Control Room Emergency Air Temperature Control System
Action 3.7.11.B	Control Room Emergency Air Temperature Control System
Action 3.7.11.B Bases	Control Room Emergency Air Temperature Control System
Ref. 3.7.12 Bases	ECCS Pump Room Exhaust Air Cleanup System
Action 3.7.12.C	ECCS Pump Room Exhaust Air Cleanup System
Action 3.7.12.C Bases	ECCS Pump Room Exhaust Air Cleanup System
SR 3.7.12.4 Bases	ECCS Pump Room Exhaust Air Cleanup System
Ref. 3.7.13 Bases	Fuel Building Air Cleanup System
Ref. 3.7.13 Bases	Fuel Building Air Cleanup System
Action 3.7.13.C	Fuel Building Air Cleanup System
Action 3.7.13.C Bases	Fuel Building Air Cleanup System
SR 3.7.13.3 Bases	Fuel Building Air Cleanup System
SR 3.7.13.4 Bases	Fuel Building Air Cleanup System
SR 3.7.13.5 Bases	Fuel Building Air Cleanup System
SR 3.7.13.5 Bases	Fuel Building Air Cleanup System
Ref. 3.7.14 Bases	Penetration Room Exhaust Air Cleanup System
Action 3.7.14.C	Penetration Room Exhaust Air Cleanup System
Action 3.7.14.C Bases	Penetration Room Exhaust Air Cleanup System
SR 3.7.14.3 Bases	Penetration Room Exhaust Air Cleanup System
SR 3.7.14.4 Bases	Penetration Room Exhaust Air Cleanup System
SR 3.7.14.5 Bases	Penetration Room Exhaust Air Cleanup System
Ref. 3.8.1 Bases	AC Sources - Operating
Action 3.8.1.G	AC Sources - Operating
Action 3.8.1.G Bases	AC Sources - Operating
SR 3.8.1.5 Bases	AC Sources - Operating

SR 3.8.1.6 Bases	AC Sources - Operating
SR 3.8.1.9 Bases	AC Sources - Operating
SR 3.8.1.10 Bases	AC Sources - Operating
SR 3.8.1.11 Bases	AC Sources - Operating
SR 3.8.1.14 Bases	AC Sources - Operating
SR 3.8.1.15 Bases	AC Sources - Operating
SR 3.8.1.16 Bases	AC Sources - Operating
SR 3.8.1.17 Bases	AC Sources - Operating
SR 3.8.1.18 Bases	AC Sources - Operating
SR 3.8.1.20 Bases	AC Sources - Operating
Ref. 3.8.4 Bases	DC Sources - Operating
Action 3.8.4.D	DC Sources - Operating
Action 3.8.4.D Bases	DC Sources - Operating
SR 3.8.4.1 Bases	DC Sources - Operating
SR 3.8.4.2 Bases	DC Sources - Operating
SR 3.8.4.3 Bases	DC Sources - Operating
Ref. 3.8.7 Bases	Inverters - Operating
Action 3.8.7.B	Inverters - Operating
Action 3.8.7.B Bases	Inverters - Operating
Ref. 3.8.9 Bases	Distribution Systems - Operating
Action 3.8.9.D	Distribution Systems - Operating
Action 3.8.9.D Bases	Distribution Systems - Operating

1.0 Description

WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Tech Spec Required Action Endstates for Westinghouse NSSS PWRs," (Ref. 1) evaluated the endstates in which the Technical Specification Actions require the unit to be placed if the Required Actions and associated Completion Times are not met. The Technical Specifications contained in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3 (Ref. 2), were reviewed to determine the proposed changes to the Required Action endstates. The endstates currently required by the Technical Specifications are defined based on transitioning the unit to a Mode or other specified condition in the Applicability in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. Mode 5 is the current endstate for LCOs that are applicable in Modes 1 through 4.

The risk of the transition from Mode 1 to Mode 4 or Mode 5 depends on the equipment that is operable. For example, the transition from Mode 4 to Mode 5 can introduce additional risk since it is required to realign the unit from steam generator cooling to residual heat removal, or shutdown cooling. During this realignment, there is an increased potential for loss of shutdown cooling and loss of inventory events, which is reflected in the plant risk calculated using Probabilistic Risk Assessment (PRA). In addition, decay heat removal following a loss of offsite power event in Mode 5 is dependant on Emergency AC power, whereas, in Mode 4 the turbine-driven auxiliary feedwater pump is available without relying on Emergency AC power. Therefore, transitioning to Mode 5 may not always be the appropriate endstate from a risk perspective.

WCAP-16294-NP-A, Rev. 1 evaluates and identifies the appropriate endstate for a number of Technical Specification Required Actions based on the risk of transitioning the unit from Mode 1 to the lower Modes. Mode 4 is justified as an acceptable alternate endstate to Mode 5.

A risk-informed approach, consistent with Regulatory Guides 1.174 and 1.177 (References 3 and 4, respectively) was used to perform the endstate evaluation. The risk associated with the transition from Mode 1 to Modes 4 and 5, and then returning to Mode 1 operation, is assessed both qualitatively and quantitatively. In addition to assessing the risk impact, the impacts on defense-in-depth and safety margins are also considered.

2.0 Proposed Changes

The Technical Specification Required Action endstates evaluated for the endstate change are contained in NUREG-1431, "Standard Technical Specifications Westinghouse Plants" (Ref. 2). The Technical Specification number, title, Condition, and current endstate evaluated and the proposed endstate are provided in the following Table:

Proposed Changes to Endstates			
Technical Specification # - Condition	**Title**	**Current Endstate**	**Proposed Endstate**
3.3.2-B	ESFAS Instrumentation	5	4
3.3.2-C	ESFAS Instrumentation	5	4
3.3.2-K	ESFAS Instrumentation	5	4
3.3.7-C	Control Room Emergency Filtration System Actuation Instrumentation	5	4
3.3.8-D	Fuel Building Air Cleanup System Actuation Instrumentation	5	4
3.4.13-B	RCS Operational Leakage	5	4
3.4.14-B	RCS Pressure Isolation Valve Leakage	5	4
3.4.15-E	RCS Leakage Detection Instrumentation	5	4
3.5.3-A, B, C	Emergency Core Cooling System - Shutdown	5	4
3.5.4-C	Refueling Water Storage Tank	5	4
3.6.6A-B	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6A-E	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6B-F	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6C-B	Containment Spray System (Ice Condenser)	5	4
3.6.6D-B	Quench Spray System (Subatmospheric)	5	4
3.6.6E-F	Recirculation Spray System (Subatmospheric)	5	4
3.6.7-B	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)	5	4
3.6.11-B	Iodine Cleanup System (Atmospheric and Subatmospheric)	5	4
3.6.12-B	Vacuum Relief Valves (Atmospheric and Ice Condenser)	5	4
3.6.13-B	Shield Building Air Cleanup System (Dual and Ice Condenser)	5	4
3.6.14-B	Air Return System (Ice Condenser)	5	4
3.6.18-C	Containment Recirculation Drains (Ice Condenser)	5	4
3.7.7-B	Component Cooling Water System	5	4
3.7.8-B	Service Water System	5	4
3.7.9-C	Ultimate Heat Sink	5	4
3.7.10-C	Control Room Emergency Filtration System	5	4

Proposed Changes to Endstates			
Technical Specification # - Condition	**Title**	**Current Endstate**	**Proposed Endstate**
3.7.11-B	Control Room Emergency Air Temperature Control System	5	4
3.7.12-C	ECCS Pump Room Exhaust Air Cleanup System	5	4
3.7.13-C	Fuel Building Air Cleanup System	5	4
3.7.14-C	Penetration Room Exhaust Air Cleanup System	5	4
3.8.1-G	AC Sources – Operating	5	4
3.8.4-D	DC Sources – Operating	5	4
3.8.7-B	Inverters – Operating	5	4
3.8.9-D	Distribution Systems – Operating	5	4

Consistent with the Required Action endstate changes listed above, the associated Bases of each affected Technical Specification are revised to reflect the new endstate and to add a brief discussion of why it is appropriate to remain within the Mode of Applicability of the associated LCO.

The structure of Specification 3.5.3, "ECCS - Shutdown," is revised to implement the change in required end state. WCAP-16294-NP-A, Rev. 1, justified a change to Specification 3.5.3 to allow remaining in Mode 4 when the required ECCS High Head Safety Injection (HHSI) subsystem is inoperable. Specification 3.5.3 currently allows remaining in Mode 4 when the required ECCS Residual Heat Removal (RHR) subsystem is inoperable. TSTF-353 contained Condition A for an inoperable ECCS RHR subsystem and Conditions B and C for an inoperable ECCS HHSI subsystem. Conditions B and C are eliminated and Condition A is revised to apply to an inoperable ECCS train (RHR or HHSI), which applies a Mode 4 end state to any ECCS train inoperability.

The Bases for Technical Specification 3.7.5, "AFW System," was also revised to add a Reviewer's Note to describe a Tier 2 requirement for the turbine-driven auxiliary feedwater pump to be available to remove decay heat in Mode 4. The turbine-driven auxiliary feedwater pump must be available to remove decay heat in order for the plant to remain within the Mode of Applicability (Mode 4). This requirement assures that a diverse means of removing decay heat is available. The availability of a diverse means of decay heat removal in Mode 4 is important to the risk evaluations performed in WCAP-16294-NP-A, Rev. 1.

3.0 Background

As discussed in Regulatory Guide 1.177, (Ref. 4) acceptable reasons for requesting Technical Specification changes fall into one or more of the following categories:

- Improvement to Operational Safety: A change to the Technical Specifications can be made due to reductions in the plant risk or a reduction in the occupational exposure of plant personnel in complying with the Technical Specification requirements.
- Consistency with Risk Basis in Regulatory Requirements: Technical Specification requirements can be changed to reflect improved design features in a plant or to reflect equipment reliability improvements that make a previous requirement unnecessarily stringent or ineffective. Technical Specifications may be changed to establish consistently based requirements across the industry or across an industry group.
- Reduce Unnecessary Burdens: The change may be requested to reduce unnecessary burdens in complying with current Technical Specification requirements, based on operating history of the plant or industry in general. This includes extending completion times 1) that are too short to complete repairs when components fail with the plant at-power, 2) to complete additional maintenance activities at-power to reduce plant down time, and 3) to provide increased flexibility to plant operators.

The benefits of revising the Technical Specification Required Action endstates are related primarily to the first two categories.

With regard to operational safety, the risk of the transition from Mode 1 to Mode 4 is lower than the risk of the transition from Mode 1 to Mode 5. The additional mode transition (Mode 4 to Mode 5) involves re-aligning the unit from steam generator cooling to residual heat removal or shutdown cooling . This activity requires system alignment changes that can lead to loss of inventory events and loss of shutdown cooling in the PRA. In addition, in Mode 4, as opposed to Mode 5, additional systems are available for event mitigation that provide a reduced risk once the unit has transitioned to the required endstate. For example, for a loss of offsite power/station blackout (LOSP/SBO) event in the PRA, the turbine driven auxiliary feedwater pump will be available for decay heat removal in Mode 4. In Mode 5, this capability is not available.

Revising the Required Action endstate will also result in increasing unit availability by decreasing the time shutdown. The additional time required to transition to Mode 5 from Mode 4 when shutting down and also to Mode 4 from Mode 5 when restarting can be eliminated with the endstate change. A typical time for the transition from Mode 4 to Mode 5 during shutdown and from Mode 5 to 4 during startup is 24 hours. As such, this change will allow a time reduction of 24 hours.

4.0 Technical Analysis

The proposed endstate changes (described in Section 2.0 of this TSTF) are technically justified in WCAP-16294-A, Rev. 1 (Ref. 1).

5.0 Regulatory Analysis

5.1 No Significant Hazards Consideration

The TSTF has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change modifies the end state (e.g., mode or other specified condition) which the Required Actions specify must be entered if compliance with the LCO is not restored. The requested technical specifications (TS) permit an end state of Mode 4 rather than an end state of Mode 5 contained in the current TS. In some cases, other Conditions and Required Actions are revised to implement the proposed change. Required Actions are not an initiator of any accident previously evaluated. Therefore, the proposed change does not affect the probability of any accident previously evaluated. The affected systems continued to be required to be operable by the Technical Specifications and the Completion Times specified in the Technical Specifications to restore equipment to operable status or take other remedial Actions remain unchanged. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Tech Spec Required Action Endstates for Westinghouse NSSS PWRs," demonstrates that the proposed change does not significantly increase the consequences of any accident previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change modifies the end state (e.g., mode or other specified condition) which the Required Actions specify must be entered if compliance with the LCO is not restored. In some cases, other Conditions and Required Actions are revised to implement the proposed change. The change does not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the change does not impose any new requirements. The change does not alter assumptions made in the safety analysis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change modifies the end state (e.g., mode or other specified condition) which the Required Actions specify must be entered if compliance with the LCO is not restored. In some cases, other Conditions and Required Actions are revised to implement the proposed change. Remaining within the Applicability of the LCO is acceptable because WCAP-16294-NP-A demonstrates that the plant risk in MODE 4 is similar to or lower than MODE 5. As a result, no margin of safety is significantly affected.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, the TSTF concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The proposed changes do not affect the design requirements or operability requirements of any plant system. The proposed changes involve changes to Technical Specification Required Actions for inoperable components or systems. 10 CFR 50.36 is the only applicable regulatory requirement.

10 CFR 50.36 contains requirements applicable to the content of plant Technical Specifications (e.g., the criteria for selecting systems, structures, or components that should be included in the Technical Specifications, and that appropriate surveillances be included, etc.) however 10 CFR 50.36 does not specify the Required Actions to be included for each Technical Specification. As such, the Technical Specifications affected by the proposed changes continue to meet the requirements contained in 10 CFR 50.36. Therefore, the Technical Specifications affected by the proposed changes remain consistent with the requirements of 10 CFR 50.36.

Regulatory Guide (RG) 1.174, Rev. 1 and RG 1.177 (References 3 and 4) provide guidance for making Technical Specification changes using risk insights. The proposed risk-informed changes are consistent with the guidance provided in these Regulatory Guides .

In summary, the proposed changes do not involve any design changes or changes to the physical arrangement of components. The proposed changes only revise the current Technical Specification Required Actions and therefore continue to satisfy the requirements of 10 CFR 50.36 and the guidance contained in RG 1.174, Rev. 1 and RG 1.177 for changes to the Technical Specification. Therefore, the proposed changes do not adversely impact the design or performance characteristics of any components or systems.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the approval of the proposed changes will not be inimical to the common defense and security or to the health and safety of the public.

6.0 Environmental Considerations

A review has determined that the proposed changes would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed changes.

7.0 References

1. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [Date to be provided later.]
2. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Rev. 3, March 2004.
3. Regulatory Guide 1.174, Rev. 1 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
4. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998.

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2-1.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel or train inoperable.	B.1 Restore channel or train to OPERABLE status.	48 hours
	OR	
	B.2.1 Be in MODE 3.	54 hours
	AND	
	B.2.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------------	
	Be in MODE ~~5~~4.	~~84~~ 60 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	------------------NOTE------------------ One train may be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE. ---	
	C.1 Restore train to OPERABLE status.	24 hours
	OR	
	C.2.1 Be in MODE 3.	30 hours
	AND	
	C.2.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------	
	Be in MODE ~~5~~4.	~~60~~36 hours
D. One channel inoperable.	[-----------------NOTE------------------ The inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels. ---	
	----------REVIEWER'S NOTE---------- The below Note should be used for plants with installed bypass test capability: One channel may be bypassed for up to 12 hours for surveillance testing. --]	
	D.1 Place channel in trip.	72 hours
	OR	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
K. One channel inoperable.	[-----------------NOTE------------------ One additional channel may be bypassed for up to [4] hours for surveillance testing. --- ----------REVIEWER'S NOTE---------- The below Note should be used for plants with installed bypass test capability: One channel may be bypassed for up to 12 hours for surveillance testing. ---]	
	K.1 Place channel in bypass.	[6] hours
	OR	
	K.2.1 Be in MODE 3.	[12] hours
	AND	
	K.2.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	[~~12~~18] hours
L. One or more channels inoperable.	L.1 Verify interlock is in required state for existing unit condition.	1 hour
	OR	
	L.2.1 Be in MODE 3.	7 hours
	AND	
	L.2.2 Be in MODE 4.	13 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	B.1.2	Enter applicable Conditions and Required Actions for one CREFS train made inoperable by inoperable CREFS actuation instrumentation.	Immediately
	OR		
	B.2	Place both trains in emergency [radiation protection] mode.	Immediately
C. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	------------- NOTE ------------ LCO 3.0.4.a is not applicable when entering MODE 4. -----------------------------------	
		Be in MODE ~~5~~4.	~~36~~12 hours
D. Required Action and associated Completion Time for Condition A or B not met during movement of [recently] irradiated fuel assemblies.	D.1	Suspend movement of [recently] irradiated fuel assemblies.	Immediately
E. [Required Action and associated Completion Time for Condition A or B not met in MODE 5 or 6.	E.1	Initiate action to restore one CREFS train to OPERABLE status.	Immediately]

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met during movement of [recently] irradiated fuel assemblies in the fuel building.	C.1	Suspend movement of [recently] irradiated fuel assemblies in the fuel building.	Immediately
D. [Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	-------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	~~36~~12 hours]

SURVEILLANCE REQUIREMENTS

---NOTE---
Refer to Table 3.3.8-1 to determine which SRs apply for each FBACS Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	Perform COT.	92 days
SR 3.3.8.3	[Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS]
SR 3.3.8.4	----------------------------NOTE----------------------------	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	6 hours ~~36~~12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
	A.2	[Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
	[or]		
		Restore RCS PIV to within limits.	72 hours]
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	------------- NOTE ------------ LCO 3.0.4.a is not applicable when entering MODE 4. -----------------------------------	
		Be in MODE ~~5~~4.	~~36~~12 hours
C. [RHR System autoclosure interlock function inoperable.	C.1	Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. [Required containment atmosphere radioactivity monitor inoperable. AND Required containment air cooler condensate flow rate monitor inoperable.	D.1 Restore required containment atmosphere radioactivity monitor to OPERABLE status. OR D.2 Restore required containment air cooler condensate flow rate monitor to OPERABLE status.	30 days 30 days]
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	6 hours ~~36~~12 hours
F. All required monitors inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform COT of the required containment atmosphere radioactivity monitor.	92 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS - Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

-----NOTE-----
An RHR train may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned to the ECCS mode of operation.

APPLICABILITY: MODE 4.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to ECCS high head subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ~~[~~Required ECCS ~~residual heat removal (RHR) subsystem~~train inoperable.	A.1 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 4. -------------- Initiate action to restore required ECCS ~~RHR subsystem~~train to OPERABLE status.	Immediately ~~]~~
~~B. Required ECCS [high head subsystem] inoperable.~~	~~B.1 Restore required ECCS [high head subsystem] to OPERABLE status.~~	~~1 hour~~
~~C. Required Action and associated Completion Time [of Condition B] not met.~~	~~C.1 Be in MODE 5.~~	~~24 hours~~

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	6 hours ~~36~~12 hours

3.6 CONTAINMENT SYSTEMS

3.6.6A Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

LCO 3.6.6A Two containment spray trains and [two] containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	6 hours ~~84~~54 hours
C. One [required] containment cooling train inoperable.	C.1 Restore [required] containment cooling train to OPERABLE status.	7 days
D. Two [required] containment cooling trains inoperable.	D.1 Restore one [required] containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND	6 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
	E.2 ------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. -----------------------------------	
	Be in MODE ~~5~~4.	~~36~~12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two [required] containment cooling trains inoperable.	E.1 Restore one [required] containment cooling train to OPERABLE status.	72 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 ------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	6 hours ~~36~~12 hours
G. Any combination of three or more trains inoperable.	G.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6B.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6B.2	Operate each [required] containment cooling train fan unit for ≥ 15 minutes.	31 days
SR 3.6.6B.3	Verify each [required] containment cooling train cooling water flow rate is ≥ [700] gpm.	31 days

3.6 CONTAINMENT SYSTEMS

3.6.6C Containment Spray System (Ice Condenser)

LCO 3.6.6C Two containment spray trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE ~~5~~4.	6 hours ~~84~~54 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6C.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6C.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.6D Quench Spray (QS) System (Subatmospheric)

LCO 3.6.6D Two QS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One QS train inoperable.	A.1	Restore QS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 AND B.2	Be in MODE 3. -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	6 hours ~~36~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6D.1	Verify each QS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6D.2	Verify each QS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.6E Recirculation Spray (RS) System (Subatmospheric)

LCO 3.6.6E Four RS subsystems [and a casing cooling tank] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RS subsystem inoperable.	A.1 Restore RS subsystem to OPERABLE status.	7 days
B. Two RS subsystems inoperable in one train.	B.1 Restore one RS subsystem to OPERABLE status.	72 hours
C. [Two inside RS subsystems inoperable.	C.1 Restore one RS subsystem to OPERABLE status.	72 hours]
D. [Two outside RS subsystems inoperable.	D.1 Restore one RS subsystem to OPERABLE status.	72 hours]
E. [Casing cooling tank inoperable.	E.1 Restore casing cooling tank to OPERABLE status.	72 hours]
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. AND F.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE ~~5~~4.	6 hours ~~84~~54 hours

3.6 CONTAINMENT SYSTEMS

3.6.7 Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ------------- NOTE ------------ LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	6 hours ~~84~~ 54 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.7.2	Verify spray additive tank solution volume is ≥ [2568] gal and ≤ [4000] gal.	184 days

3.6 CONTAINMENT SYSTEMS

3.6.11 Iodine Cleanup System (ICS) (Atmospheric and Subatmospheric)

LCO 3.6.11 Two ICS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ICS train inoperable.	A.1 Restore ICS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	~~36~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.11.1	Operate each ICS train for [≥ 10 continuous hours with heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days
SR 3.6.11.2	Perform required ICS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.11.3	Verify each ICS train actuates on an actual or	[18] months

3.6 CONTAINMENT SYSTEMS

3.6.12 Vacuum Relief Valves (Atmospheric and Ice Condenser)

LCO 3.6.12 [Two] vacuum relief lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One vacuum relief line inoperable.	A.1	Restore vacuum relief line to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	-------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------------	
		Be in MODE ~~5~~4.	~~36~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.12.1	Verify each vacuum relief line is OPERABLE in accordance with the Inservice Testing Program.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.13 *Shield Building Air Cleanup System (SBACS) (Dual and Ice Condenser)*

LCO 3.6.13 Two SBACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SBACS train inoperable.	A.1 Restore SBACS train to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE ~~5~~4.	~~36~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.13.1	Operate each SBACS train for [≥ 10 continuous hours with heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days
SR 3.6.13.2	Perform required SBACS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.13.3	Verify each SBACS train actuates on an actual or	[18] months

3.6 CONTAINMENT SYSTEMS

3.6.14 Air Return System (ARS) (Ice Condenser)

LCO 3.6.14 Two ARS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ARS train inoperable.	A.1	Restore ARS train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	-------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. -----------------------------------	
		Be in MODE ~~5~~4.	~~36~~12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.14.1	Verify each ARS fan starts on an actual or simulated actuation signal, after a delay of ≥ [9.0] minutes and ≤ [11.0] minutes, and operates for ≥ 15 minutes.	[92] days
SR 3.6.14.2	Verify, with the ARS fan dampers closed, each ARS fan motor current is ≥ [20.5] amps and ≤ [35.5] amps [when the fan speed is ≥ [840] rpm and ≤ [900] rpm].	92 days

3.6 CONTAINMENT SYSTEMS

3.6.18 Containment Recirculation Drains (Ice Condenser)

LCO 3.6.18 The ice condenser floor drains and the refueling canal drains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ice condenser floor drain inoperable.	A.1 Restore ice condenser floor drain to OPERABLE status.	1 hour
B. One refueling canal drain inoperable.	B.1 Restore refueling canal drain to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	6 hours ~~36~~12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	--------------NOTE-------------- Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. ------------------------------------	
		Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	------------- NOTE ------------ LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------	
		Be in MODE ~~5~~4.	~~36~~12 hours

3.7 PLANT SYSTEMS

3.7.8 Service Water System (SWS)

LCO 3.7.8 Two SWS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SWS train inoperable.	A.1 --------------NOTES------------- 1. Enter applicable and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by SWS. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by SWS. ------------------------------------ Restore SWS train to OPERABLE status.	 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 -------------- NOTE ------------- <u>LCO 3.0.4.a is not applicable when entering MODE 4.</u> ------------------------------------ Be in MODE ~~5~~<u>4</u>.	6 hours ~~36~~<u>12</u> hours

3.7 PLANT SYSTEMS

3.7.9 Ultimate Heat Sink (UHS)

LCO 3.7.9 The UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. [One or more cooling towers with one cooling tower fan inoperable.	A.1	Restore cooling tower fan(s) to OPERABLE status.	7 days]
-----REVIEWER'S NOTE----- The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit. ------------------------------------- B. [Water temperature of the UHS > [90]°F and ≤ [ ]°F.	B.1	Verify water temperature of the UHS is ≤ [90]°F averaged over the previous 24 hour period.	Once per hour]
C. [Required Action and associated Completion Time of Condition A or B not met. OR] UHS inoperable [for reasons other than Condition A or B].	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	-------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE ~~5~~4.	~~36~~12 hours

3.7 PLANT SYSTEMS

3.7.10 *Control Room Emergency Filtration System (CREFS)*

LCO 3.7.10 Two CREFS trains shall be OPERABLE.

--NOTE--
The control room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, [5, and 6],
During movement of [recently] irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREFS train inoperable.	A.1 Restore CREFS train to OPERABLE status.	7 days
B. Two CREFS trains inoperable due to inoperable control room boundary in MODE 1, 2, 3, or 4.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	6 hours ~~36~~ 12 hours

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 Two CREATCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, [5, and 6],
During movement of [recently] irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREATCS train inoperable.	A.1 Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 ------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------- Be in MODE ~~5~~4.	6 hours ~~36~~ 12 hours
C. Required Action and associated Completion Time of Condition A not met [in MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies.	C.1 Place OPERABLE CREATCS train in operation. OR C.2 Suspend movement of [recently] irradiated fuel assemblies.	Immediately Immediately
D. Two CREATCS trains inoperable [in MODE 5 or 6, or] during movement of [recently] irradiated fuel	D.1 Suspend movement of [recently] irradiated fuel assemblies.	Immediately

3.7 PLANT SYSTEMS

3.7.12 Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.12 Two ECCS PREACS trains shall be OPERABLE.

--NOTE--
The ECCS pump room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECCS PREACS train inoperable.	A.1 Restore ECCS PREACS train to OPERABLE status.	7 days
B. Two ECCS PREACS trains inoperable due to inoperable ECCS pump room boundary.	B.1 Restore ECCS pump room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 <u>-------------- NOTE --------------</u> <u>LCO 3.0.4.a is not applicable when entering MODE 4.</u> ----------------------------------- Be in MODE ~~5~~<u>4</u>.	6 hours ~~36~~ <u>12</u> hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. [Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4. OR Two FBACS trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	C.1 Be in MODE 3. AND C.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	6 hours ~~36~~ 12 hours]
D. Required Action and associated Completion Time [of Condition A] not met during movement of [recently] irradiated fuel assemblies in the fuel building.	D.1 Place OPERABLE FBACS train in operation. OR D.2 Suspend movement of [recently] irradiated fuel assemblies in the fuel building.	Immediately Immediately
E. Two FBACS trains inoperable during movement of [recently] irradiated fuel assemblies in the fuel building.	E.1 Suspend movement of [recently] irradiated fuel assemblies in the fuel building.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each FBACS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].	31 days

3.7 PLANT SYSTEMS

3.7.14 Penetration Room Exhaust Air Cleanup System (PREACS)

LCO 3.7.14 Two PREACS trains shall be OPERABLE.

---NOTE---
The penetration room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PREACS train inoperable.	A.1 Restore PREACS train to OPERABLE status.	7 days
B. Two PREACS trains inoperable due to inoperable penetration room boundary.	B.1 Restore penetration room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 ------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. -----------------------------------	
	Be in MODE ~~5~~4.	~~36~~ 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----REVIEWER'S NOTE----- This Condition may be deleted if the unit design is such that any sequencer failure mode will only affect the ability of the associated DG to power its respective safety loads following a loss of offsite power independent of, or coincident with, a Design Basis Event. ------------------------------------- F. [One [required] [automatic load sequencer] inoperable.	F.1 Restore [required] [automatic load sequencer] to OPERABLE status.	[12] hours]
G. Required Action and associated Completion Time of Condition A, B, C, D, E, or [F] not met.	G.1 Be in MODE 3. AND G.2 -------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------- Be in MODE ~~5~~4.	6 hours ~~36~~ 12 hours
H. Three or more [required] AC sources inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.1.1	Verify correct breaker alignment and indicated power availability for each [required] offsite circuit.	7 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources - Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One [or two] battery charger[s on one train] inoperable.	A.1	Restore battery terminal voltage to greater than or equal to the minimum established float voltage.	2 hours
	AND		
	A.2	Verify battery float current ≤ [2] amps.	Once per [12] hours
	AND		
	A.3	Restore battery charger[s] to OPERABLE status.	7 days
[B. One [or two] batter[y][ies on one train] inoperable.	B.1	Restore batter[y][ies] to OPERABLE status.	[2] hours]
C. One DC electrical power subsystem inoperable for reasons other than Condition A [or B].	C.1	Restore DC electrical power subsystem to OPERABLE status.	[2] hours
D. Required Action and Associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering	

CONDITION	REQUIRED ACTION	COMPLETION TIME
	MODE 4. -------------------------------------	
	Be in MODE ~~5~~4.	~~36~~ 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2 ------------- NOTE ------------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	~~36~~ 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, [frequency], and alignment to required AC vital buses.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 ------------- NOTE ------------ LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE ~~5~~4.	6 hours ~~36~~ 12 hours
E. Two or more electrical power distribution subsystems inoperable that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to [required] AC, DC, and AC vital bus electrical power distribution subsystems.	7 days

BASES

ACTIONS (continued)

This action addresses the train orientation of the SSPS for the functions listed above. If a channel or train is inoperable, 24 hours is allowed to return it to an OPERABLE status. Note that for containment spray and Phase B isolation, failure of one or both channels in one train renders the train inoperable. Condition B, therefore, encompasses both situations. The specified Completion Time is reasonable considering that there are two automatic actuation trains and another manual initiation train OPERABLE for each Function, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. This is done by placing the unit in at least MODE 3 within an additional 6 hours (54 hours total time) and in MODE ~~5~~ 4 within an additional ~~30~~ 6 hours (~~84~~ 60 hours total time).

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

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

Condition C applies to the automatic actuation logic and actuation relays for the following functions:

- SI,
- Containment Spray,

- Phase A Isolation,
- Phase B Isolation, and
- Automatic Switchover to Containment Sump.

This action addresses the train orientation of the SSPS and the master and slave relays. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference ~~8~~9. The specified Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be restored to OPERABLE status, the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. This is done by placing the unit in at least MODE 3 within

BASES

ACTIONS (continued)

an additional 6 hours (30 hours total time) and in MODE ~~5~~ 4 within an additional ~~30~~ 6 hours (~~60~~ 36 hours total time).

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing, provided the other train is OPERABLE. This allowance is based on the reliability analysis assumption of WCAP-10271-P-A (Ref. ~~9~~10) that 4 hours is the average time required to perform train surveillance.

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

Condition D applies to:

- Containment Pressure - High 1,
- Pressurizer Pressure - Low (two, three, and four loop units),
- Steam Line Pressure - Low,
- Steam Line Differential Pressure - High,
- High Steam Flow in Two Steam Lines Coincident With T_{avg} - Low Low or Coincident With Steam Line Pressure - Low,

BASES

ACTIONS (continued)

If one channel is inoperable, 72 hours are allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Generally this Condition applies to functions that operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-three configuration that satisfies redundancy requirements. The 72 hours allowed to restore the channel to OPERABLE status or to place it in the tripped condition is justified in Reference ~~8~~9.

Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 6 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

[The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 12 hours for surveillance testing of other channels. The 12 hours allowed for testing, are justified in Reference ~~8~~9.]

---------------------------REVIEWER'S NOTE---
The below text should be used for plants with installed bypass test capability:

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. The 12 hour time limit is justified in Reference ~~8~~9.

E.1, E.2.1, and E.2.2

Condition E applies to:

- Containment Spray Containment Pressure - High 3 (High, High) (two, three, and four loop units), and
- Containment Phase B Isolation Containment Pressure - High 3 (High, High).

BASES

ACTIONS (continued)

None of these signals has input to a control function. Thus, two-out-of-three logic is necessary to meet acceptable protective requirements. However, a two-out-of-three design would require tripping a failed channel. This is undesirable because a single failure would then cause spurious containment spray initiation. Spurious spray actuation is undesirable because of the cleanup problems presented. Therefore, these channels are designed with two-out-of-four logic so that a failed channel may be bypassed rather than tripped. Note that one channel may be bypassed and still satisfy the single failure criterion. Furthermore, with one channel bypassed, a single instrumentation channel failure will not spuriously initiate containment spray.

To avoid the inadvertent actuation of containment spray and Phase B containment isolation, the inoperable channel should not be placed in the tripped condition. Instead it is bypassed. Restoring the channel to OPERABLE status, or placing the inoperable channel in the bypass condition within 72 hours, is sufficient to assure that the Function remains OPERABLE and minimizes the time that the Function may be in a partial trip condition (assuming the inoperable channel has failed high). The Completion Time is further justified based on the low probability of an event occurring during this interval. Failure to restore the inoperable channel to OPERABLE status, or place it in the bypassed condition within 6 hours, requires the unit be placed in MODE 3 within the following 6 hours and MODE 4 within the next 72 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 4, these Functions are no longer required OPERABLE.

[The Required Actions are modified by a Note that allows one additional channel to be bypassed for up to 12 hours for surveillance testing. Placing a second channel in the bypass condition for up to 12 hours for testing purposes is acceptable based on the results of Reference ~~8~~9.]

----------------------------REVIEWER'S NOTE---
The below text should be used for plants with installed bypass test capability:

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. The 12 hour time limit is justified in Reference ~~8~~9.

BASES

ACTIONS (continued)

The action addresses the train orientation of the SSPS and the master and slave relays for these functions. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference ~~8~~9. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. If the train cannot be returned to OPERABLE status, the unit must be brought to MODE 3 within the next 6 hours and MODE 4 within the following 6 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Placing the unit in MODE 4 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. ~~9~~10) assumption that 4 hours is the average time required to perform channel surveillance.

[H.1 and H.2

Condition H applies to the automatic actuation logic and actuation relays for the Turbine Trip and Feedwater Isolation Function.

This action addresses the train orientation of the SSPS and the master and slave relays for this Function. If one train is inoperable, 24 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the following 6 hours. The 24 hours allowed for restoring the inoperable train to OPERABLE status is justified in Reference ~~8~~9. The Completion Time for restoring a train to OPERABLE status is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. These Functions are no longer required in MODE 3. Placing the unit in MODE 3 removes all requirements for OPERABILITY of the protection channels and actuation functions. In this MODE, the unit does not have analyzed transients or conditions that require the explicit use of the protection functions noted above.

BASES

ACTIONS (continued)

The Required Actions are modified by a Note that allows one train to be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE. This allowance is based on the reliability analysis (Ref. ~~9~~10) assumption that 4 hours is the average time required to perform channel surveillance.]

I.1 and I.2

Condition I applies to:

- [SG Water Level - High High (P-14) (two, three, and four loop units), and]
- Undervoltage Reactor Coolant Pump.

If one channel is inoperable, 72 hours are allowed to restore one channel to OPERABLE status or to place it in the tripped condition. If placed in the tripped condition, the Function is then in a partial trip condition where one-out-of-two or one-out-of-three logic will result in actuation. Failure to restore the inoperable channel to OPERABLE status or place it in the tripped condition within 72 hours requires the unit to be placed in MODE 3 within the following 6 hours. The allowed Completion Time of 78 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, these Functions are no longer required OPERABLE.

[The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to [12] hours for surveillance testing of other channels. The 72 hours allowed to place the inoperable channel in the tripped condition, and the 12 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference ~~8~~9.]

---------------------------REVIEWER'S NOTE--
The below text should be used for plants with installed bypass test capability:

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. The 72 hours allowed to place the inoperable channel in the tripped condition, and the 12 hours allowed for a second channel to be in the bypassed condition for testing, are justified in Reference ~~8~~9.

BASES

ACTIONS (continued)

J.1 and J.2

Condition J applies to the AFW pump start on trip of all MFW pumps.

88This action addresses the train orientation of the SSPS for the auto start function of the AFW System on loss of all MFW pumps. The OPERABILITY of the AFW System must be assured by allowing automatic start of the AFW System pumps. If a channel is inoperable, 48 hours are allowed to return it to an OPERABLE status. If the function cannot be returned to an OPERABLE status, 6 hours are allowed to place the unit in MODE 3. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. In MODE 3, the unit does not have any analyzed transients or conditions that require the explicit use of the protection function noted above. The allowance of 48 hours to return the train to an OPERABLE status is justified in Reference ~~9~~10.

K.1, K.2.1, and K.2.2

Condition K applies to:

- RWST Level - Low Low Coincident with Safety Injection, and
- RWST Level - Low Low Coincident with Safety Injection and Coincident with Containment Sump Level - High.

RWST Level - Low Low Coincident With SI and Coincident With Containment Sump Level - High provides actuation of switchover to the containment sump. Note that this Function requires the bistables to energize to perform their required action. The failure of up to two channels will not prevent the operation of this Function. However, placing a failed channel in the tripped condition could result in a premature switchover to the sump, prior to the injection of the minimum volume from the RWST. Placing the inoperable channel in bypass results in a two-out-of-three logic configuration, which satisfies the requirement to allow another failure without disabling actuation of the switchover when required. Restoring the channel to OPERABLE status or placing the inoperable channel in the bypass condition within [6] hours is sufficient to ensure that the Function remains OPERABLE, and minimizes the time that the Function may be in a partial trip condition (assuming the

BASES

ACTIONS (continued)

inoperable channel has failed high). The [6] hour Completion Time is justified in Reference ~~10~~11. If the channel cannot be returned to OPERABLE status or placed in the bypass condition within 6 hours, the unit must be ~~brought to~~placed in a MODE in which overall plant risk is reduced. This is done by placing the unit in at least MODE 3 within the following [6] hours and MODE ~~5~~ 4 within the next ~~30~~ 6 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action K.2.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. ~~In MODE 5, the unit does not have any analyzed transients or conditions that require the explicit use of the protection functions noted above.~~

[The Required Actions are modified by a Note that allows placing a second channel in the bypass condition for up to [4] hours for surveillance testing. The total of [12] hours to reach MODE 3 and [4] hours for a second channel to be bypassed is acceptable based on the results of Reference ~~10~~11.]

---------------------------REVIEWER'S NOTE---------------------------
The below text should be used for plants with installed bypass test capability:

The Required Actions are modified by a Note that allows placing one channel in bypass for up to 12 hours while performing routine surveillance testing. The channel to be tested can be tested in bypass with the inoperable channel also in bypass. The total of [12] hours to reach MODE 3 and [4] hours for a second channel to be bypassed is acceptable based on the results of Reference ~~10~~11.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and reliability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. The SSPS is tested every 92 days on a STAGGERED TEST BASIS, using the semiautomatic tester. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and that there is an intact voltage signal path to the master relay coils. The Frequency of every 92 days on a STAGGERED TEST BASIS is justified in Reference ~~11~~12.

SR 3.3.2.3

SR 3.3.2.3 is the performance of an ACTUATION LOGIC TEST as described in SR 3.3.2.2, except that the semiautomatic tester is not used and the continuity check does not have to be performed, as explained in the Note. This SR is applied to the balance of plant actuation logic and relays that do not have the SSPS test circuits installed to utilize the semiautomatic tester or perform the continuity check. This test is also performed every 31 days on a STAGGERED TEST BASIS. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.4

SR 3.3.2.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 92 days on a STAGGERED TEST BASIS. The time allowed for the testing (4 hours) is justified in Reference ~~11~~12. The Frequency of 92 days is justified in Reference ~~9~~10.

SR 3.3.2.5

SR 3.3.2.5 is the performance of a COT.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of Reference 6.

The Frequency of 184 days is justified in Reference ~~11~~12.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.10

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the Technical Requirements Manual, Section 15 (Ref. ~~12~~13). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one with the resulting measured response time compared to the appropriate FSAR response time. Alternately, the response time test can be performed with the time constants set to their nominal value provided the required response time is analytically calculated assuming the time constants are set at their nominal values. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

----------------------------REVIEWER'S NOTE--
Applicable portions of the following Bases are applicable for plants adopting WCAP-13632-P-A (Ref. ~~9~~10). and/or WCAP-14036-P (Ref. ~~10~~11).

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor, signal processing and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," (Ref. ~~13~~14) dated January 1996, provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

BASES

SURVEILLANCE REQUIREMENTS (continued)

WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," (Ref. ~~14~~15) provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

ESF RESPONSE TIME tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching [1000] psig in the SGs.

SR 3.3.2.11

SR 3.3.2.11 is the performance of a TADOT as described in SR 3.3.2.8, except that it is performed for the P-4 Reactor Trip Interlock, and the Frequency is once per RTB cycle. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable TADOT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. This Frequency is based on operating experience demonstrating that undetected failure of the P-4 interlock sometimes occurs when the RTB is cycled.

The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Function tested has no associated setpoint.

BASES

REFERENCES

1. FSAR, Chapter [6].

2. FSAR, Chapter [7].

3. FSAR, Chapter [15].

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. Plant-specific setpoint methodology study.

7. NUREG-1218, April 1988.

8. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~8~~9. WCAP-14333-P-A, Rev. 1, October 1998.

~~9~~10. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

~~10~~11. [Plant specific evaluation reference.]

~~11~~12. WCAP-15376, Rev. 0. October 2000.

~~12~~13. Technical Requirements Manual, Section 15, "Response Times."

~~13~~14. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

~~14~~15. WCAP-14036-P, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," December 1995.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which ~~the LCO requirements are not applicable~~overall plant risk is reduced. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

D.1

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met when [recently] irradiated fuel assemblies are being moved. Movement of [recently] irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require CREFS actuation.

E.1

Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.4

SR 3.3.7.4 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 31 days on a STAGGERED TEST BASIS. The Frequency is acceptable based on instrument reliability and industry operating experience.

[SR 3.3.7.5

SR 3.3.7.5 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the bypass condition, thus preventing inadequate actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay coil is pulse tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed ever 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference ~~1~~2.

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.]

[SR 3.3.7.6

SR 3.3.7.6 is the performance of a MASTER RELAY TEST. The MASTER RELAY TEST is the energizing of the master relay, verifying contact operation and a low voltage continuity check of the slave relay coil. Upon master relay contact operation, a low voltage is injected to the slave relay coil. This voltage is insufficient to pick up the slave relay, but large enough to demonstrate signal path continuity. This test is performed every 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference ~~1~~2.

The SR is modified by a Note stating that the Surveillance is only applicable to the master relays of the ESFAS Instrumentation.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.9

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~1~~2. WCAP-15376, Rev. 0, October 2000.

BASES

ACTIONS (continued)

C.1

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and [recently] irradiated fuel assemblies are being moved in the fuel building. Movement of [recently] irradiated fuel assemblies in the fuel building must be suspended immediately to eliminate the potential for events that could require FBACS actuation.

D.1 and D.2

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which ~~the LCO requirements are not applicable~~overall plant risk is reduced. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that table 3.3.8-1 determines which SRs apply to which FBACS Actuation Functions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Specifications tests at least once per refueling interval with applicable extensions. In some instances, the test includes actuation of the end device (e.g., pump starts, valve cycles, etc.). The Frequency is based on operating experience and is consistent with the typical industry refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Functions tested have no setpoints associated with them.

SR 3.3.8.5

A CHANNEL CALIBRATION is performed every [18] months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.

2. Unit Specific Setpoint Calibration Procedure.

3. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE ~~5~~ 4 within ~~36~~ 12 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE ~~5~~4, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

SURVEILLANCE REQUIREMENTS (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.20, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference ~~5~~6. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. ~~5~~6).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [15].
4. NEI 97-06, "Steam Generator Program Guidelines."
5. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~5~~6. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

BASES

ACTIONS (continued)

[Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

[or]

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.]

----------------------------------REVIEWER'S NOTE----------------------------------
Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.

B.1 and B.2

If leakage cannot be reduced, [the system can not be isolated,] or the other Required Actions accomplished, the plant must be brought to a MODE in which ~~the requirement does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE ~~5~~ 4 within ~~36~~ 12 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 7). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or

other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

C.1

The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure function.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every [18] months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The [18 month] Frequency is consistent with 10 CFR 50.55a(g) (Ref. ~~8~~9) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code (Ref. ~~7~~8), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to

BASES

SURVEILLANCE REQUIREMENTS (continued)

be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

[SR 3.4.14.2 and SR 3.4.14.3

Verifying that the RHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of [600] psig. The interlock setpoint that prevents the valves from being opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.]

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, Section V, GDC 55.

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.

5. NUREG-0677, May 1980.

[6. Document containing list of PIVs.]

7. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~7~~8. ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~8~~9. 10 CFR 50.55a(g).

BASES

ACTIONS (continued)

[D.1 and D.2

With the required containment atmosphere radioactivity monitor and the required containment air cooler condensate flow rate monitor inoperable, the only means of detecting leakage is the containment sump monitor. This Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable required monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy time period.]

E.1 and E.2

If a Required Action of Condition A, B, [C], or [D] cannot be met, the plant must be brought to a MODE in which ~~the requirement does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action E.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a COT on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3, [SR 3.4.15.4, and SR 3.4.15.5]

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.
2. Regulatory Guide 1.45.
3. FSAR, Section [ ].
4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

~~B.1~~

With no ECCS high head subsystem OPERABLE, due to the inoperability of the centrifugal charging pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The ~~1 hour~~ Completion Time of immediately to initiate actions that would restore at least one ECCS high head subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity ~~or to initiate actions to place the plant in MODE 5, where an ECCS train is not required~~.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 1). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action A.1 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

~~C.1~~

~~When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.~~

SURVEILLANCE SR 3.5.3.1

REQUIREMENTS	The applicable Surveillance descriptions from Bases 3.5.2 apply.
REFERENCES	1. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE]. The applicable references from Bases 3.5.2 also apply.

BASES

ACTIONS (continued)

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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 3.5.4.1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter [6] and Chapter [15].
2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

B.1 and B.2

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ 54 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows 48 hours to restore ~~additional time for attempting restoration of~~ the containment spray train to OPERABLE status in MODE 3. This ~~and~~ is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

C.1

With one of the required containment cooling trains inoperable, the inoperable required containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System

BASES

ACTIONS (continued)

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action E.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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.

F.1

With two containment spray trains or any combination of three or more containment spray and cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.6.6A.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.3

Verifying that each [required] containment cooling train ESW cooling flow rate to each cooling unit is ≥ [700] gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6A.4

Verifying each containment 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 the ASME Code (Ref. ~~8~~9). Since the containment spray 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 abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6A.5 and SR 3.6.6A.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-3 pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is 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 [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.7

This SR requires verification that each [required] containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The [18] month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6A.5 and SR 3.6.6A.6, above, for further discussion of the basis for the [18] month Frequency.

SR 3.6.6A.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at [the first refueling and at] 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.
2. 10 CFR 50, Appendix K.
3. FSAR, Section [ ].
4. FSAR, Section [ ].
5. FSAR, Section [ ].
6. FSAR, Section [ ].
7. FSAR, Section [ ].
8. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~8~~9. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS (continued)

D.1 and D.2

If one required containment spray train is inoperable and one of the required containment cooling trains is inoperable, the inoperable containment spray train or the inoperable containment cooling train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as those given in Required Action C.1.

E.1

If two required containment cooling trains are inoperable, one of the required containment cooling trains must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as those given in Required Action C.1.

F.1 and F.2

If any of the Required Actions or associated Completion Times for Condition A, B, C, D, or E of this LCO are not met, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action F.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or

other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

G.1

With any combination of three or more containment spray and containment cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6B.1

Verifying the correct alignment for manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System flow path provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct positions prior to being secured. This SR does not require testing or valve manipulation. Rather, it involves verification that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6B.2

Operating each [required] containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed based on the known reliability of the fan units and controls, the two train redundancy available, and the low probability of significant degradation of the containment cooling train occurring between surveillances.

SR 3.6.6B.3

Verifying that each [required] containment cooling train ESW cooling flow rate to each cooling unit is ≥ [700] gpm provides assurance that the design flow rate assumed in the analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6B.4

Verifying that each containment 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 the ASME Code (Ref. ~~8~~9). Since the containment spray 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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6B.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, a test at [the first refueling and at] 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.

2. 10 CFR 50, Appendix A.

3. FSAR, Section [15].

4. FSAR, Section [6.2].

5. FSAR, Section [ ].

6. FSAR, Section [ ].

7. FSAR, Section [ ].

8. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~8~~9. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

APPLICABILITY

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

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one containment spray train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal and iodine removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

If the affected containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ 54 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows 48 hours to restore the containment spray train to OPERABLE status in MODE 3. This ~~additional time and~~ is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6C.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.2

Verifying that each containment 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 head are normal tests of centrifugal pump performance required by the ASME Code (Ref. ~~5~~6). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on bypass flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.6.3 and SR 3.6.6.4

These SRs require verification that each automatic containment spray valve actuates to its correct position and each containment spray pump starts upon receipt of an actual or simulated containment spray actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is 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 these components usually pass the Surveillances when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The surveillance of containment sump isolation valves is also required by SR 3.6.6.3. A single surveillance may be used to satisfy both requirements.

SR 3.6.6.5

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Because of the passive design of the nozzle, a test at [the first refueling and at] 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.

2. FSAR, Section [6.2].

3. 10 CFR 50.49.

4. 10 CFR 50, Appendix K.

5. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~5~~6. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS

A.1

If one QS train is inoperable, it must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal and iodine removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

If the Required Action and associated Completion Time are not met, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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 3.6.6D.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the QS System provides assurance that the proper flow path exists for QS System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6D.2

Verifying that each QS pump's developed head at the flow test point is greater than or equal to the required developed head ensures that QS pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 45). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on bypass 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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6D.3 and SR 3.6.6D.4

These SRs ensure that each QS automatic valve actuates to its correct position and each QS pump starts upon receipt of an actual or simulated containment spray actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is 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 an [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.6D.5

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzle, a test at [the first refueling and at] 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. FSAR, Section [6.2].

2. 10 CFR 50.49.

3. 10 CFR 50, Appendix K.

4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

54. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

ACTIONS (continued)

[C.1
With two inside RS subsystems inoperable, at least one of the inoperable subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.]

[D.1

With two outside RS subsystems inoperable, at least one of the inoperable subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.]

[E.1

With the casing cooling tank inoperable, the NPSH available to the outside RS subsystem pumps may not be sufficient. The inoperable casing cooling tank must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.]

F.1 and F.2

If the inoperable RS subsystem(s) [or the casing cooling tank] cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ 54 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action F.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of

LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows 48 hours to restore the RS subsystem(s) [or casing cooling tank] to OPERABLE status in MODE 3. This ~~additional time and~~ is reasonable considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6E.4

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the RS System and casing cooling tank provides assurance that the proper flow path exists for operation of the RS System. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified as being in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.6E.5

Verifying that each RS [and casing cooling] pump's developed head at the flow test point is greater than or equal to the required developed head ensures that these pumps' performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by the ASME Code (Ref. 45). Since the QS System pumps cannot be tested with flow through the spray headers, they are tested on bypass 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 3.6.6E.6

These SRs ensure that each automatic valve actuates and that the RS System and casing cooling pumps start upon receipt of an actual or simulated High-High containment pressure signal. Start delay times are also verified for the RS System pumps. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was considered to be acceptable from a reliability standpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6E.7

This SR ensures that each spray nozzle is unobstructed and that spray coverage of the containment will meet its design bases objective. An air or smoke test is performed through each spray header. Due to the passive design of the spray header and its normally dry state, a test at [the first refueling and at] 10 year intervals is considered adequate for detecting obstruction of the nozzles.

REFERENCES

1. FSAR, Section [6.2].

2. 10 CFR 50.49.

3. 10 CFR 50, Appendix K.

4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

45. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

LCO (continued)

spray flow until the Containment Spray System suction path is switched from the RWST to the containment sump, and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between [7.2 and 11.0]. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ 54hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows 48 hours to restore ~~for restoration of~~ the Spray

BASES

ACTIONS (continued)

Additive System to OPERABLE status in MODE 3 ~~and 36 hours to reach MODE 5~~. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.7.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.5

To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, the flow rate in the Spray Additive System is verified once every 5 years. This SR provides assurance that the correct amount of NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow rate.

REFERENCES

1. FSAR, Chapter [15].
2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS

A.1

With one ICS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as:

a. The availability of the OPERABLE redundant ICS train,

b. The fact that, even with no ICS train in operation, almost the same amount of iodine would be removed from the containment atmosphere through absorption by the Containment Spray System, and

c. The fact that the Completion Time is adequate to make most repairs.

B.1 and B.2

If the ICS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.11.2

This SR verifies that the required ICS filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.11.3

The automatic startup test verifies that both trains of equipment start upon receipt of an actual or simulated test signal. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. Furthermore, the Frequency was developed considering that the system equipment OPERABILITY is demonstrated at a 31 day Frequency by SR 3.6.11.1.

[SR 3.6.11.4

The ICS filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The [18] month Frequency is considered to be acceptable based on the damper reliability and design, the mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.]

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41, GDC 42, and GDC 43.
2. FSAR, Section [6.5].
3. Regulatory Guide 1.52, Revision [2].
4. FSAR, Chapter [15].

5. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

LCO

The LCO establishes the minimum equipment required to accomplish the vacuum relief function following the inadvertent actuation of containment cooling features. Two 100% vacuum relief lines are required to be OPERABLE to ensure that at least one is available, assuming one or both valves in the other line fail to open.

APPLICABILITY

In MODES 1, 2, 3, and 4, the containment cooling features, such as the Containment Spray System, are required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside containment could occur whenever these systems are required to be OPERABLE due to inadvertent actuation of these systems. Therefore, the vacuum relief lines are required to be OPERABLE in MODES 1, 2, 3, and 4 to mitigate the effects of inadvertent actuation of the Containment Spray System, Quench Spray (QS) System, or Containment Cooling System.

In MODES 5 and 6, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations of these MODES. The Containment Spray System, QS System, and Containment Cooling System are not required to be OPERABLE in MODES 5 and 6. Therefore, maintaining OPERABLE vacuum relief valves is not required in MODE 5 or 6.

ACTIONS

A.1

When one of the required vacuum relief lines is inoperable, the inoperable line must be restored to OPERABLE status within 72 hours. The specified time period is consistent with other LCOs for the loss of one train of a system required to mitigate the consequences of a LOCA or other DBA.

B.1 and B.2

If the vacuum relief line cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met.

However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.12.1 This SR cites the Inservice Testing Program, which establishes the requirement that inservice testing of the ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with the ASME Code (Ref. ~~2~~3). Therefore, SR Frequency is governed by the Inservice Testing Program.
REFERENCES	1. FSAR, Section [6.2]. 2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE]. ~~2~~3. ASME Code for Operation and Maintenance of Nuclear Power Plants.

BASES

LCO

In the event of a DBA, one SBACS train is required to provide the minimum particulate iodine removal assumed in the safety analysis. Two trains of the SBACS must be OPERABLE to ensure that at least one train will operate, assuming that the other train is disabled by a single active failure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could lead to fission product release to containment that leaks to the shield building. The large break LOCA, on which this system's design is based, is a full power event. Less severe LOCAs and leakage still require the system to be OPERABLE throughout these MODES. The probability and severity of a LOCA decrease as core power and Reactor Coolant System pressure decrease. With the reactor shut down, the probability of release of radioactivity resulting from such an accident is low.

In MODES 5 and 6, the probability and consequences of a DBA are low due to the pressure and temperature limitations in these MODES. Under these conditions, the Filtration System is not required to be OPERABLE (although one or more trains may be operating for other reasons, such as habitability during maintenance in the shield building annulus).

ACTIONS

A.1

With one SBACS train inoperable, the inoperable train must be restored to OPERABLE status within 7 days. The components in this degraded condition are capable of providing 100% of the iodine removal needs after a DBA. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SBACS train and the low probability of a DBA occurring during this period. The Completion Time is adequate to make most repairs.

B.1 and B.2

If the SBACS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of

LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.6.13.4

The SBACS filter bypass dampers are tested to verify OPERABILITY. The dampers are in the bypass position during normal operation and must reposition for accident operation to draw air through the filters. The [18] month Frequency is considered to be acceptable based on damper reliability and design, mild environmental conditions in the vicinity of the dampers, and the fact that operating experience has shown that the dampers usually pass the Surveillance when performed at the [18] month Frequency.]

SR 3.6.13.5

The proper functioning of the fans, dampers, filters, adsorbers, etc., as a system is verified by the ability of each train to produce the required system flow rate. The [18] month Frequency on a STAGGERED TEST BASIS is consistent with Regulatory Guide 1.52 (Ref. 45) guidance for functional testing.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. FSAR, Section [6.5].
3. FSAR, Chapter [15].
4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].
45. Regulatory Guide 1.52, Revision [2].

BASES

ACTIONS (continued)

B.1 and B.2

If the ARS train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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 3.6.14.1

Verifying that each ARS fan starts on an actual or simulated actuation signal, after a delay ≥ [9.0] minutes and ≤ [11.0] minutes, and operates for ≥ 15 minutes is sufficient to ensure that all fans are OPERABLE and that all associated controls and time delays are functioning properly. It also ensures that blockage, fan and/or motor failure, or excessive vibration can be detected for corrective action. The [92] day Frequency was developed considering the known reliability of fan motors and controls and the two train redundancy available.

SR 3.6.14.2

Verifying ARS fan motor current to be at rated speed with the return air dampers closed confirms one operating condition of the fan. This test is

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.6.14.4

Verifying the OPERABILITY of the motor operated valve in the Hydrogen Skimmer System hydrogen collection header to the lower containment compartment provides assurance that the proper flow path will exist when the valve receives an actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. This Surveillance also confirms that the time delay to open is within specified tolerances. The 92 day Frequency was developed considering the known reliability of the motor operated valves and controls and the two train redundancy available. Operating experience has also shown this Frequency to be acceptable.]

REFERENCES

1. FSAR, Section [6.2].
2. 10 CFR 50, Appendix K.
3. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

APPLICABILITY (continued)

The probability and consequences of these events in MODES 5 and 6 are low due to the pressure and temperature limitations of these MODES. As such, the containment recirculation drains are not required to be OPERABLE in these MODES.

ACTIONS

A.1

If one ice condenser floor drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1

If one refueling canal drain is inoperable, 1 hour is allowed to restore the drain to OPERABLE status. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status in 1 hour.

C.1 and C.2

If the affected drain(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.18.1 Verifying the OPERABILITY of the refueling canal drains ensures that they will be able to perform their functions in the event of a DBA. This Surveillance confirms that the refueling canal drain plugs have been removed and that the drains are clear of any obstructions that could impair their functioning. In addition to debris near the drains, attention must be given to any debris that is located where it could be moved to the drains in the event that the Containment Spray System is in operation and water is flowing to the drains. SR 3.6.18.1 must be performed before entering MODE 4 from MODE 5 after every filling of the canal to ensure that the plugs have been removed and that no debris that could impair the drains was deposited during the time the canal was filled. The 92 day Frequency was developed considering such factors as the inaccessibility of the drains, the absence of traffic in the vicinity of the drains, and the redundancy of the drains. SR 3.6.18.2 Verifying the OPERABILITY of the ice condenser floor drains ensures that they will be able to perform their functions in the event of a DBA. Inspecting the drain valve disk ensures that the valve is performing its function of sealing the drain line from warm air leakage into the ice condenser during normal operation, yet will open if melted ice fills the line following a DBA. Verifying that the drain lines are not obstructed ensures their readiness to drain water from the ice condenser. The [18] month Frequency was developed considering such factors as the inaccessibility of the drains during power operation; the design of the ice condenser, which precludes melting and refreezing of the ice; and operating experience that has confirmed that the drains are found to be acceptable when the Surveillance is performed at an [18] month Frequency. Because of high radiation in the vicinity of the drains during power operation, this Surveillance is normally done during a shutdown.
REFERENCES	1. FSAR, Section [6.2]. 2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

APPLICABLE SAFETY ANALYSES (continued)

ESFAS logic may not detect the affected steam generator if the backflow check valve to the affected MFW header worked properly. One motor driven AFW pump would deliver to the broken MFW header at the pump runout flow until the problem was detected, and flow terminated by the operator. Sufficient flow would be delivered to the intact steam generator by the redundant AFW pump.

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. DC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

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

LCO

----------------------------------REVIEWER'S NOTE----------------------------------

[Implementation of WCAP-16294-NP-A, Rev. 1 "Risk Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," provides the allowance to remain in MODE 4 instead of transitioning to MODE 5 for certain Required Actions. In order to utilize this allowance, a requirement must be established to maintain the TDAFW train available in MODE 4. The requirement to maintain the TDAFW train available in order to remain in MODE 4 will be addressed by a licensing commitment to include a requirement that will be located in the Technical Specification Bases, a Licensee Controlled Document, or implementing procedures. If a sufficient steam supply is unavailable during the application of the MODE 4 end state, the unit's configuration risk management program will direct the appropriate actions to be taken. The risk impact is managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This program will determine the safest course of action for an emergent condition in MODE 4 that renders the TDAFW train unavailable, and could include proceeding to a MODE 5 endstate in the affected Specifications.

The additional requirement to maintain the TDAFW train available in MODE 4 in accordance with WCAP-16294-NP-A, Rev. 1, may be included in this Bases or a reference to the location of this requirement (e.g., a specific location in the Technical Requirements Manual, other licensee controlled document, or the implementing procedure may be included in this Bases.]

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

The AFW System is configured into [three] trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE in [two] diverse paths, each supplying AFW to separate steam generators. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each of [two] main steam lines upstream of the MSIVs, and shall be capable of supplying AFW to any of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The LCO is modified by a Note indicating that one AFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

BASES

APPLICABILITY (continued)

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

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," be entered if an inoperable CCW train results in an inoperable RHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

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

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.2].
2. FSAR, Section [6.2].

3. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

LCO 3.0.6 and ensures the proper actions are taken for these components. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SWS train cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE ~~5~~4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1

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

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.8.2

This SR verifies proper automatic operation of the SWS valves on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.8.3

This SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.1].
2. FSAR, Section [6.2].
3. FSAR, Section [5.4.7].
4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

The 7 day Completion Time is reasonable based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable (in one or more cooling towers), the number of available systems, and the time required to reasonably complete the Required Action.]

[B.1

-----------------------------------REVIEWER'S NOTE-----------------------------------
The [ ]°F is the maximum allowed UHS temperature value and is based on temperature limitations of the equipment that is relied upon for accident mitigation and safe shutdown of the unit.

With water temperature of the UHS > [90]°F, the design basis assumption associated with initial UHS temperature are bounded provided the temperature of the UHS averaged over the previous 24 hour period is ≤ [90]°F. With the water temperature of the UHS > [90]°F, long term cooling capability of the ECCS loads and DGs may be affected. Therefore, to ensure long term cooling capability is provided to the ECCS loads when water temperature of the UHS is > [90]°F, Required Action B.1 is provided to more frequently monitor the water temperature of the UHS and verify the temperature is ≤ [90]°F when averaged over the previous 24 hour period. The once per hour Completion Time takes into consideration UHS temperature variations and the increased monitoring frequency needed to ensure design basis assumptions and equipment limitations are not exceeded in this condition. If the water temperature of the UHS exceeds [90]°F when averaged over the previous 24 hour period or the water temperature of the UHS exceeds [ ]°F, Condition C must be entered immediately.]

[C.1 and C.2

If the Required Actions and Completion Times of Condition [A or B] are not met, or the UHS is inoperable for reasons other than Condition A [or B], the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS

[SR 3.7.9.1

This SR verifies that adequate long term (30 day) cooling can be maintained. The specified level also ensures that sufficient NPSH is available to operate the SWS pumps. The [24] hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the UHS water level is ≥ [562] ft [mean sea level].]

[SR 3.7.9.2

This SR verifies that the SWS is available to cool the CCW System to at least its maximum design temperature with the maximum accident or normal design heat loads for 30 days following a Design Basis Accident. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES. This SR verifies that the average water temperature of the UHS is ≤ [90°F].]

[SR 3.7.9.3

Operating each cooling tower fan for ≥[15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, the known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.]

[SR 3.7.9.4

This SR verifies that each cooling tower fan starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with the typical refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.]

REFERENCES

1. FSAR, Section [9.2.5].

2. Regulatory Guide 1.27.

3. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room boundary.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CREFS train or control room boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE ~~that minimizes accident risk~~in which the overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 3). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

D.1 and D.2

[In MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies, if the inoperable CREFS train cannot be restored to OPERABLE status within the required Completion Time, action must be

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.10.3

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. The Frequency of [18] months is specified in Regulatory Guide 1.52 (Ref. ~~3~~4).

SR 3.7.10.4

This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the CREFS. During the emergency mode of operation, the CREFS is designed to pressurize the control room ≥[0.125] inches water gauge positive pressure with respect to adjacent areas in order to prevent unfiltered inleakage. The CREFS is designed to maintain this positive pressure with one train at a makeup flow rate of [3000] cfm. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with the guidance provided in NUREG-0800 (Ref. ~~4~~5).

REFERENCES

1. FSAR, Section [6.4].

2. FSAR, Chapter [15].

3. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~3~~4. Regulatory Guide 1.52; Rev. [2].

~~4~~5. NUREG-0800, Section 6.4, Rev. 2, July 1981.

BASES

LCO (continued)

The CREATCS is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the heating and cooling coils and associated temperature control instrumentation. In addition, the CREATCS must be operable to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, 4, [5, and 6,] and during movement of [recently] irradiated fuel assemblies, the CREATCS must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room. [The CREATCS is only required to be OPERABLE during fuel handling involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous [X] days), due to radioactive decay.]

[In MODE 5 or 6,] CREATCS may not be required for those facilities that do not require automatic control room isolation.

ACTIONS

A.1

With one CREATCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREATCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CREATCS train could result in loss of CREATCS function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE ~~that minimizes the risk~~in which the overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 2). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

ACTIONS (continued)

C.1 and C.2

[In MODE 5 or 6, or] during movement of [recently] irradiated fuel, if the inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREATCS train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

[In MODE 5 or 6, or] during movement of [recently] irradiated fuel assemblies, with two CREATCS trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREATCS trains are inoperable in MODE 1, 2, 3, or 4, the control room CREATCS may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.11.1

This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses] in the control room. This SR consists of a combination of testing and calculations. The [18] month Frequency is appropriate since significant degradation of the CREATCS is slow and is not expected over this time period.

REFERENCES

1. FSAR, Section [6.4].
2. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

If the ECCS pump room boundary is inoperable, the ECCS PREACS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE ECCS pump room boundary within 24 hours. During the period that the ECCS pump room boundary is inoperable, appropriate compensatory measures [consistent with the intent, as applicable, of GDC 19, 60, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the ECCS pump room boundary.

C.1 and C.2

If the ECCS PREACS train or ECCS pump room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 6). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.12.2

This SR verifies that the required ECCS PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal adsorbers efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test Frequencies and additional information are discussed in detail in the [VFTP].

SR 3.7.12.3

This SR verifies that each ECCS PREACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference 4.

SR 3.7.12.4

This SR verifies the integrity of the ECCS pump room enclosure. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the ECCS PREACS. During the [post accident] mode of operation, the ECCS PREACS is designed to maintain a slight negative pressure in the ECCS pump room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The ECCS PREACS is designed to maintain a ≤[-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm from the ECCS pump room. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. ~~6~~7).

This test is conducted with the tests for filter penetration; thus, an [18] month Frequency on a STAGGERED TEST BASIS is consistent with that specified in Reference 4.

[SR 3.7.12.5

Operating the ECCS PREACS bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the ECCS PREACS bypass damper is verified if it can be specified in Reference 4.]

BASES

REFERENCES

1. FSAR, Section [6.5.1].
2. FSAR, Section [9.4.5].
3. FSAR, Section [15.6.5].
4. Regulatory Guide 1.52 (Rev. 2).
5. 10 CFR 100.11.
6. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].
~~6~~7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

ACTIONS (continued)

applicable, of GDC 19, 60, 61, 63, 64 and 10 CFR Part 100] should be utilized to protect plant personnel from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the fuel building boundary.

[C.1 and C.2

In MODE 1, 2, 3, or 4, when Required Action A.1 or B.1 cannot be completed within the associated Completion Time, or when both FBACS trains are inoperable for reasons other than an inoperable fuel building boundary (i.e., Condition B), the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be placed in MODE 3 within 6 hours, and in MODE ~~5~~4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 6). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

D.1 and D.2

BASES

ACTIONS (continued)

E.1

When two trains of the FBACS are inoperable during movement of [recently] irradiated fuel assemblies in the fuel building, action must be taken to place the unit in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of [recently] irradiated fuel assemblies in the fuel building. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system.

Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

[SR 3.7.13.2

This SR verifies that the required FBACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].]

[SR 3.7.13.3

This SR verifies that each FBACS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with Reference ~~6~~7.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.13.4

This SR verifies the integrity of the fuel building enclosure. The ability of the fuel building to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the FBACS. During the [post accident] mode of operation, the FBACS is designed to maintain a slight negative pressure in the fuel building, to prevent unfiltered LEAKAGE. The FBACS is designed to maintain a ≤[-0.125] inches water gauge with respect to atmospheric pressure at a flow rate of [20,000] cfm to the fuel building. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800, Section 6.5.1 (Ref. ~~7~~8).

An [18] month Frequency (on a STAGGERED TEST BASIS) is consistent with Reference ~~6~~7.

[SR 3.7.13.5

Operating the FBACS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FBACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with Reference ~~6~~7.]

REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

3. FSAR, Section [15.7.4].

4. Regulatory Guide 1.25.

5. 10 CFR 100.

6. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~6~~7. Regulatory Guide 1.52, Rev. [2].

~~7~~8. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

BASES

ACTIONS (continued)

C.1 and C.2

If the inoperable train or penetration room boundary cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 5). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action C.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

Standby systems should be checked periodically to ensure that they function properly. As the environmental and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture that may have accumulated in the charcoal as a result of humidity in the ambient air. [Systems with heaters must be operated for ≥10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of equipment and the two train redundancy available.

SR 3.7.14.2

This SR verifies that the required PREACS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the [VFTP].

[SR 3.7.14.3

This SR verifies that each PREACS starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with that specified in Reference ~~5~~6.]

BASES

SURVEILLANCE REQUIREMENTS (continued)

[SR 3.7.14.4

This SR verifies the integrity of the penetration room enclosure. The ability of the penetration room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of PREACS. During the [post accident] mode of operation, the PREACS is designed to maintain a ≤[-0.125] inches water gauge relative to atmospheric pressure at a flow rate of [3000] cfm in the penetration room, with respect to adjacent areas, to prevent unfiltered LEAKAGE. The Frequency of [18] months is consistent with the guidance provided in NUREG-0800 (Ref. ~~6~~7).

The minimum system flow rate maintains a slight negative pressure in the penetration room area, and provides sufficient air velocity to transport particulate contaminants, assuming only one filter train is operating. The number of filter elements is selected to limit the flow rate through any individual element to about [3000] cfm. This may vary based on filter housing geometry. The maximum limit ensures that the flow through, and pressure drop across, each filter element are not excessive.

The number and depth of the adsorber elements ensure that, at the maximum flow rate, the residence time of the air stream in the charcoal bed achieves the desired adsorption rate. At least a [0.125] second residence time is necessary for an assumed [99]% efficiency.

The filters have a certain pressure drop at the design flow rate when clean. The magnitude of the pressure drop indicates acceptable performance, and is based on manufacturers' recommendations for the filter and adsorber elements at the design flow rate. An increase in pressure drop or a decrease in flow indicates that the filter is being loaded or that there are other problems with the system.

This test is conducted along with the tests for filter penetration; thus, the [18] month Frequency is consistent with that specified in Reference ~~5~~6.]

[SR 3.7.14.5

It is necessary to operate the PREACS filter bypass damper to ensure that the system functions properly. The OPERABILITY of the PREACS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference ~~5~~6.]

BASES

REFERENCES

1. FSAR, Section [6.5.1].

2. FSAR, Section [9.4.5].

3. FSAR, Section [15.6.5].

4. 10 CFR 100.

5. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~5~~6. Regulatory Guide 1.52, Rev. [2].

~~6~~7. NUREG-0800, Section 6.5.1, Rev. 2, July 1981.

ACTIONS (continued)

G.1 and G.2

If the inoperable AC electric power sources cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action G.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

H.1

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

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. ~~8~~9). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. ~~9~~10), and Regulatory Guide 1.137 (Ref. ~~10~~11), as addressed in the FSAR.

Where the SRs discussed herein specify voltage and frequency tolerances, the following is applicable. The minimum steady state output voltage of [3740] V is 90% of the nominal 4160 V output voltage. This value, which is specified in ANSI C84.1 (Ref. ~~11~~12), allows for voltage drop to the terminals of 4000 V motors whose minimum operating voltage is specified as 90% or 3600 V. It also allows for voltage drops to motors and other equipment down through the 120 V level where minimum operating voltage is also usually specified as 90% of name plate rating. The specified maximum steady state output voltage of [4756] V is equal to the maximum operating voltage specified for 4000 V motors. It ensures that for a lightly loaded distribution system, the voltage at the terminals of 4000 V motors is no more than the maximum rated operating

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by four Notes. Note 1 indicates that diesel engine runs for this Surveillance may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. Note 2 states that momentary transients, because of changing bus loads, do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 3 indicates that this Surveillance should be conducted on only one DG at a time in order to avoid common cause failures that might result from offsite circuit or grid perturbations. Note 4 stipulates a prerequisite requirement for performance of this SR. A successful DG start must precede this test to credit satisfactory performance.

SR 3.8.1.4

This SR provides verification that the level of fuel oil in the day tank [and engine mounted tank] is at or above the level at which fuel oil is automatically added. The level is expressed as an equivalent volume in gallons, and is selected to ensure adequate fuel oil for a minimum of 1 hour of DG operation at full load plus 10%.

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

SR 3.8.1.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel oil day [and engine mounted] tanks once every [31] days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. ~~10~~11). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from its associated storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

[The Frequency for this SR is variable, depending on individual system design, with up to a [92] day interval. The [92] day Frequency corresponds to the testing requirements for pumps as contained in the ASME Code (Ref. ~~11~~12); however, the design of fuel transfer systems is such that pumps operate automatically or must be started manually in order to maintain an adequate volume of fuel oil in the day [and engine mounted] tanks during or following DG testing. In such a case, a 31 day Frequency is appropriate. Since proper operation of fuel transfer systems is an inherent part of DG OPERABILITY, the Frequency of this SR should be modified to reflect individual designs.]

SR 3.8.1.7

See SR 3.8.1.2.

[SR 3.8.1.8

Transfer of each [4.16 kV ESF bus] power supply from the normal offsite circuit to the alternate offsite circuit demonstrates the OPERABILITY of the alternate circuit distribution network to power the shutdown loads. The [18 month] Frequency of the Surveillance is based on engineering judgment, taking into consideration the unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2

SURVEILLANCE REQUIREMENTS (continued)

is further amplified to allow the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed 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.] Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.9

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip. [For this unit, the single load for each DG and its horsepower rating is as follows:] This Surveillance may be accomplished by:

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus, or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

As required by IEEE-308 (Ref. ~~12~~13), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.

The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 3 seconds specified is equal to 60% of a typical 5 second load sequence interval associated with

SURVEILLANCE REQUIREMENTS (continued)

sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection. The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. ~~9~~10).

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

Note 2 ensures that the DG is tested under load conditions that are as close to design basis conditions as possible. When synchronized with offsite power, testing should be performed at a power factor of ≤ [0.9]. This power factor is representative of the actual inductive loading a DG would see under design basis accident conditions. Under certain conditions, however, Note 2 allows the Surveillance to be conducted at a power factor other than ≤ [0.9]. These conditions occur when grid voltage is high, and the additional field excitation needed to get the power factor to ≤ [0.9] results in voltages on the emergency busses that are too high. Under these conditions, the power factor should be maintained as close as practicable to [0.9] while still maintaining acceptable voltage limits on the emergency busses. In other circumstances, the grid voltage may be such that the DG excitation levels needed to obtain a power factor of [0.9] may not cause unacceptable voltages on the emergency busses, but the excitation levels are in excess of those recommended for the DG. In such cases, the power factor shall be maintained as close as practicable to [0.9] without exceeding the DG excitation limits.

SURVEILLANCE REQUIREMENTS (continued)

---------------------------------REVIEWER'S NOTE---------------------------------
The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable,

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.10

This Surveillance demonstrates the DG capability to reject a full load without overspeed tripping or exceeding the predetermined voltage limits. The DG full load rejection may occur because of a system fault or inadvertent breaker tripping. This Surveillance ensures proper engine generator load response under the simulated test conditions. This test simulates the loss of the total connected load that the DG experiences following a full load rejection and verifies that the DG does not trip upon loss of the load. These acceptance criteria provide for DG damage protection. While the DG is not expected to experience this transient during an event and continues to be available, this response ensures that the DG is not degraded for future application, including reconnection to the bus if the trip initiator can be corrected or isolated.

The [18 month] Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. ~~9~~10) and is intended to be consistent with expected fuel cycle lengths.

This SR has been modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbation to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety

SURVEILLANCE REQUIREMENTS (continued)

a. Performance of the SR will not render any safety system or component inoperable,

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.11

As required by Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(1), this Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of [10] seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems 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.

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(1), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

[SR 3.8.1.12

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time ([10] seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.12.d and SR 3.8.1.12.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

SURVEILLANCE REQUIREMENTS (continued)

insights or deterministic methods may be used for this assessment. Credit may be taken for unplanned events that satisfy this SR.

--------------------------------REVIEWER'S NOTE--------------------------------
The above MODE restrictions may be deleted if it can be demonstrated to the staff, on a plant specific basis, that performing the SR with the reactor in any of the restricted MODES can satisfy the following criteria, as applicable:

a. Performance of the SR will not render any safety system or component inoperable,

b. Performance of the SR will not cause perturbations to any of the electrical distribution systems that could result in a challenge to steady state operation or to plant safety systems, and

c. Performance of the SR, or failure of the SR, will not cause, or result in, an AOO with attendant challenge to plant safety systems.

SR 3.8.1.14

Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(3), requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ [2] hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY.

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(3), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by three Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor

SURVEILLANCE REQUIREMENTS (continued)

The [18 month] Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(5).

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least [2] hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.16

As required by Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(6), this Surveillance ensures that the manual synchronization and automatic load transfer from the DG to the offsite source can be made and the DG can be returned to ready to load status when offsite power is restored. It also ensures that the autostart logic is reset to allow the DG to reload if a subsequent loss of offsite power occurs. The DG is considered to be in ready to load status when the DG is at rated speed and voltage, the output breaker is open and can receive an autoclose signal on bus undervoltage, and the load sequence timers are reset.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(6), and takes into consideration unit conditions required to perform the Surveillance.

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

SURVEILLANCE REQUIREMENTS (continued)

independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment.] Credit may be taken for unplanned events that satisfy this SR.

SR 3.8.1.18

Under accident [and loss of offsite power] conditions loads are sequentially connected to the bus by the [automatic load sequencer]. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The [10]% load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~9~~10), paragraph 2.a.(2), takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

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

SURVEILLANCE REQUIREMENTS (continued)

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

SR 3.8.1.20

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

The 10 year Frequency is consistent with the recommendations of Regulatory Guide 1.108 (Ref. ~~9~~10).

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

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. FSAR, Chapter [8].
3. Regulatory Guide 1.9, Rev. 3.
4. FSAR, Chapter [6].
5. FSAR, Chapter [15].
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
8. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

~~8~~9. 10 CFR 50, Appendix A, GDC 18.

~~9~~10. Regulatory Guide 1.108, Rev. 1, August 1977.

~~10~~11. Regulatory Guide 1.137, Rev. [ ], [date].

~~11~~12. ASME Code for Operation and Maintenance of Nuclear Power Plants.

~~12~~13. IEEE Standard 308-1978.

BASES

ACTIONS (continued)

C.1

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

If one of the required DC electrical power subsystems is inoperable for reasons other than Condition A or B (e.g., inoperable battery charger and associated inoperable battery), the remaining DC electrical power subsystem has the capacity to support a safe shutdown and to mitigate an accident condition. Since a subsequent worst- case single failure could, however, result in the loss of minimum necessary DC electrical subsystems to mitigate a worst case accident, continued power operation should not exceed 2 hours. The 2 hour Completion Time is based on Regulatory Guide 1.93 (Ref. 7) and reflects a reasonable time to assess unit status as a function of the inoperable DC electrical power subsystem and, if the DC electrical power subsystem is not restored to OPERABLE status, to prepare to effect an orderly and safe unit shutdown.

D.1 and D.2

If the inoperable DC electrical power subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 8). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge helps to ensure the effectiveness of the battery chargers, which support the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state while supplying the continuous steady state loads of the associated DC subsystem. On float charge, battery cells will receive adequate current to optimally charge the battery. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the minimum float voltage established by the battery manufacturer ([2.20] Vpc or [127.6] V at the battery terminals). This voltage maintains the battery plates in a condition that supports maintaining the grid life (expected to be approximately 20 years). The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. ~~8~~9).

SR 3.8.4.2

This SR verifies the design capacity of the battery chargers. According to Regulatory Guide 1.32 (Ref. ~~9~~10), the battery charger supply is recommended to be based on the largest combined demands of the various steady state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the unit during these demand occurrences. The minimum required amperes and duration ensure that these requirements can be satisfied.

This SR provides two options. One option requires that each battery charger be capable of supplying [400] amps at the minimum established float voltage for [8] hours. The ampere requirements are based on the output rating of the chargers. The voltage requirements are based on the charger voltage level after a response to a loss of AC power. The time period is sufficient for the charger temperature to have stabilized and to have been maintained for at least [2] hours.

The other option requires that each battery charger be capable of recharging the battery after a service test coincident with supplying the largest coincident demands of the various continuous steady state loads (irrespective of the status of the plant during which these demands occur). This level of loading may not normally be available following the

BASES

SURVEILLANCE REQUIREMENTS (continued)

battery service test and will need to be supplemented with additional loads. The duration for this test may be longer than the charger sizing criteria since the battery recharge is affected by float voltage, temperature, and the exponential decay in charging current. The battery is recharged when the measured charging current is ≤ [2] amps.

The Surveillance Frequency is acceptable, given the unit conditions required to perform the test and the other administrative controls existing to ensure adequate charger performance during these [18 month] intervals. In addition, this Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.8.4.3

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

The Surveillance Frequency of [18 months] is consistent with the recommendations of Regulatory Guide 1.32 (Ref. ~~9~~10) and Regulatory Guide 1.129 (Ref. ~~10~~11), which state that the battery service test should be performed during refueling operations, or at some other outage, with intervals between tests not to exceed [18 months].

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

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems. This restriction from normally performing the Surveillance in MODE 1 or 2 is further amplified to allow portions of the Surveillance to be performed for the purpose of reestablishing OPERABILITY (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated OPERABILITY concerns) provided an assessment determines plant safety is maintained or enhanced. This assessment shall, as a minimum, consider the potential outcomes and transients associated with a failed partial

BASES

SURVEILLANCE REQUIREMENTS (continued)

Surveillance, a successful partial Surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the partial Surveillance; as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when portions of the Surveillance are performed in MODE 1 or 2. Risk insights or deterministic methods may be used for the assessment. Credit may be taken for unplanned events that satisfy this SR.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. Regulatory Guide 1.6, March 10, 1971.
3. IEEE-308-[1978].
4. FSAR, Chapter [8].
5. FSAR, Chapter [6].
6. FSAR, Chapter [15].
7. Regulatory Guide 1.93, December 1974.
8. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].
~~8~~9. IEEE-450-[1995].
~~9~~10. Regulatory Guide 1.32, February 1977.
~~10~~11. Regulatory Guide 1.129, December 1974.

BASES

ACTIONS

A.1

With a required inverter inoperable, its associated AC vital bus becomes inoperable until it is [manually] re-energized from its [Class 1E constant voltage source transformer or inverter using internal AC source].

For this reason a Note has been included in Condition A requiring the entry into the Conditions and Required Actions of LCO 3.8.9, "Distribution Systems - Operating." This ensures that the vital bus is re-energized within 2 hours.

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

B.1 and B.2

If the inoperable devices or components cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action B.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or

other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.7.1 This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC vital buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation of the RPS and ESFAS connected to the AC vital buses. The 7 day Frequency takes into account the redundant capability of the inverters and other indications available in the control room that alert the operator to inverter malfunctions.
REFERENCES	1. FSAR, Chapter [8]. 2. FSAR, Chapter [6]. 3. FSAR, Chapter [15]. 4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

BASES

ACTIONS (continued)

Condition C represents one or more DC buses or distribution panels without adequate DC power; potentially both with the battery significantly degraded and the associated charger nonfunctioning. In this situation, the unit is significantly more vulnerable to a complete loss of all DC power. It is, therefore, imperative that the operator's attention focus on stabilizing the unit, minimizing the potential for loss of power to the remaining trains and restoring power to the affected train.

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

a. The potential for decreased safety by requiring a change in unit conditions (i.e., requiring a shutdown) while allowing stable operations to continue,

b. The potential for decreased safety by requiring entry into numerous applicable Conditions and Required Actions for components without DC power and not providing sufficient time for the operators to perform the necessary evaluations and actions for restoring power to the affected train, and

c. The potential for an event in conjunction with a single failure of a redundant component.

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

D.1 and D.2

If the inoperable distribution subsystem cannot be restored to OPERABLE status within the required Completion Time, the unit must be brought to a MODE in which ~~the LCO does not apply~~overall plant risk is reduced. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~36~~ 12 hours.

Remaining within the Applicability of the LCO is acceptable because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. 4). In MODE 4 there are two means of decay heat removal, which provides diversity and defense in depth. However, voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

Required Action D.2 is modified by a Note that states that LCO 3.0.4.a is not applicable when entering MODE 4. This Note prohibits the use of LCO 3.0.4.a to enter MODE 4 during startup with the LCO not met. However, there is no restriction on the use of LCO 3.0.4.b, if applicable, because LCO 3.0.4.b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 4, and establishment of risk management actions, if appropriate. LCO 3.0.4 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

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

BASES

ACTIONS (continued)

E.1

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

SURVEILLANCE REQUIREMENTS

SR 3.8.9.1

This Surveillance verifies that the [required] AC, DC, and AC vital bus electrical power distribution systems are functioning properly, with the correct circuit breaker alignment. The correct breaker alignment ensures the appropriate separation and independence of the electrical divisions is maintained, and the appropriate voltage is available to each required bus. The verification of proper voltage availability on the buses ensures that the required voltage is readily available for motive as well as control functions for critical system loads connected to these buses. The 7 day Frequency takes into account the redundant capability of the AC, DC, and AC vital bus electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.

REFERENCES

1. FSAR, Chapter [6].

2. FSAR, Chapter [15].

3. Regulatory Guide 1.93, December 1974.

4. WCAP-16294-NP-A, Rev. 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," [DATE].

APPENDIX C RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION, DATED DECEMBER 12, 2007

NUCLEAR ENERGY INSTITUTE

Biff Bradley
DIRECTOR
RISK ASSESSMENT
NUCLEAR GENERATION DIVISION

December 12, 2007

Mr. Jon H. Thompson
Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to NRC Request for Additional Information Regarding PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134)

Project Number: 689

Dear Mr. Thompson:

Enclosed for NRC review are the responses to the NRC's June 13, 2007 Request for Additional Information (RAI) regarding WCAP-16294-NP, which addresses the endstates risk-informed technical specifications initiative for Westinghouse plants. Attachment 1 provides the RAI responses, and attachment 2 provides proposed markups to WCAP-16294-NP associated with certain RAI responses.

The changes to WCAP-16294-NP contained in attachment 2 will be incorporated into the approved version of WCAP-16294-NP, which will be issued as Revision 1, following receipt of the final Safety Evaluation Report.

Additionally, the technical specification and bases changes justified in WCAP-16294-NP that are included in Appendix B of attachment 2 will be removed from the approved WCAP and submitted to the NRC in a TSTF.

Please contact me if you have any questions.

Sincerely,

Biff Bradley

Attachments

c: Mr. Carl S. Schulten, NRC
NRC Document Control Desk

By letter dated September 9, 2005, the Nuclear Energy Institute (NEI) submitted topical report (TR) WCAP-16294-NP (August 2005), "Risk Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," requesting U.S. Nuclear Regulatory Commission (NRC) staff review and approval. WCAP-16294 includes the technical justification supporting Risk Management Technical Specification (TS) Initiative 1 for Westinghouse Nuclear Steam Supply System (NSSS) pressurized water reactors (PWRs).

NRC letter to NEI dated June 13, 2007, Project Numbers 689 and 700 transmitted the NRC staff Request for Additional Information (RAI) questions on TR WCAP-16294. Listed below are the NRC RAI questions and the responses.

Probabilistic and Risk Assessment Branch Questions

NRC RAI 1

For the Limiting Condition For Operation (LCO) governing containment isolation functions, the proposed changes allow a Mode 4 end state when the condition represents unavailability of the containment barrier (i.e., LC0 3.6.1, or conditions with both airlock doors open (action 3.6.2.c), or a penetration open and unisolable (action 3.6.3.b)). In such conditions, it is not clear that the justification based on diversity of core cooling mechanisms is an adequate basis for the change, since core cooling is not directly relevant to the containment fission product barrier. A cold shutdown endstate is more appropriate for such conditions due to the complete unavailability of the containment barrier, which should be reflected in the TR and in the markup TS. Provide justification for a Mode 4 end state or do not apply a hot shutdown endstate to these containment LCOs (Note: CE TSTF-422, "Change in Technical Specifications End States, CE-NPSD-1186" does not apply a hot shutdown endstate in these cases).

Response to RAI 1

The text of the WCAP will be revised to delete the TS Required Action endstate changes for TS 3.6.1 Required Action B.2, TS 3.6.2 Required Action D.2, and TS 3.6.3 Required Action F.2. By removing these Technical Specification changes from the WCAP, there will be no Required Action endstate changes directly related to containment integrity.

Note that the Technical Specification pages with the finalized changes marked up will be provided separately (from the revised WCAP pages) in the TSTF associated with this WCAP.

NRC RAI 2

The assessment of the relative risks of operation in Mode 4 compared to Mode 5 assumes that the turbine-driven auxiliary feedwater (AFW) pump is available (assures heat removal for station blackout), and that the reactor trip breakers are open (rod withdrawal accidents not credible). These assumptions are not TS requirements, and tier 2 restrictions have not been identified for any of the LCOs. The TR should justify these assumptions on equipment availability and propose appropriate requirements or other controls for those LCOs for which these assumptions support the changed end state. Similarly, there appear to be assumptions made in the risk analyses with regards to the unavailability of equipment in Mode 5 which are then factored into the qualitative and quantitative risk analyses conclusions, but no specific basis could be identified to justify these assumptions:

- Unavailability of the AFW start signal.
- Unavailability of one Emergency Core Cooling System (ECCS) train.
- Unavailability of safety injection (SI) pumps (note that quantitative analysis uses combined charging/SI pumps, so at least one SI pump should be available in Mode 5).
- Unavailability of alternating current (AC) sources.
- Unavailability of containment isolation and cooling systems.

Although these systems and functions may not be required by TS, the report states that equipment is assumed available unless operating procedures direct isolation or lockout. It is not apparent that the isolation or lockout of the above functions is so directed at all plants. For example, plants with high pressure charging SI pumps would not be required to lock out all SI pumps, and there is no requirement to remove sources of AC power in Mode 5. The report should clarify differences in assumed equipment availability to mitigate initiating events between Modes 4 and 5, and provide a specific basis as to why equipment is not credited in Mode 5 for both the qualitative and quantitative risk analyses.

Response to RAI 2

The Bases for the Technical Specifications addressed in the WCAP will include a requirement that the auxiliary feedwater turbine-driven pump be available for decay heat removal in Mode 4. This is an important assumption that has been modeled in the Mode 4 (POS 3) PRA model. From a review of plant operating procedures, the reactor trip breakers are expected to be open when the plant is in Mode 4 and rod withdrawal accidents would not be expected to be important contributors to risk in Mode 4. The more important assumption of the availability of the auxiliary feedwater turbine-driven pump will be specifically addressed by the changes to the Technical Specification Bases.

The following discusses the bulleted items in the RAI.

- Unavailability of the AFW start signal – The ESFAS Technical Specification (LCO 3.3.2) Applicability for AFW automatic start instrumentation only requires this instrumentation to be operable in Modes 1, 2, and 3. Therefore, it is reasonable to assume the automatic start feature of the AFW system will be unavailable in Modes 4 and 5. Given the results of the analyses performed, it is not expected that a realistic modeling would change the conclusion that the risk is lower for a cooldown to Mode 4 than it is to Mode 5.
- Unavailability of one Emergency Core Cooling System (ECCS) train – the POS 4 (Mode 5) PRA model assumes both high pressure safety injection trains are available if needed to mitigate an accident in Mode 5. This will be clarified in the

text in Section 6.3.1 under the POS 4: Plant Response Model heading (see Attachment 2, Insert A).

- Unavailability of safety injection (SI) pumps – the text on WCAP-16294 page 6-5 and in Table 6-4 is based on a review of plant operating procedures and indicates that the SI pumps are disabled. The POS 4 (Mode 5) PRA model assumes that both trains of SI are available with 3 charging pumps. This will be clarified in the text in Section 6.3.1 under the POS 4: Plant Response Model heading (see Attachment 2, Insert A).
- Unavailability of alternating current (AC) sources – the full compliment of offsite power was modeled in both the POS 3 (Mode 4) and POS 4 (Mode 5) PRA models. Table 6-2 indicates that less than the full complement of diesel generators may available depending on the equipment required. The POS 4 (Mode 5) PRA model assumes that both diesel generators in the PRA model are available. This will be clarified in the text in Section 6.3.1 under the POS 4: Plant Response Model heading (see Attachment 2, Insert A).
- Unavailability of containment isolation and cooling systems – a LERF model was not developed for this program. The qualitative evaluation is based on lower CDP risk for cooldown to Mode 4 and systems likely to be available in Mode 4. The POS 3 PRA model does credit containment cooling systems for small loss of coolant accidents, but their effect is minor as can be seen from the results presented in WCAP-16294 Table 6-13. Also note that the text of the WCAP will be revised to delete the endstate changes for the Technical Specifications associated with containment integrity (TS 3.6.1 Required Action B.2, TS 3.6.2 Required Action D.2, and TS 3.6.3 Required Action F.2) (see the response to RAI 1). Note that the Technical Specification pages with the finalized changes marked up will be provided separately (from the revised WCAP pages) in the TSTF associated with this WCAP.

NRC RAI 3

LC0 3.0.4.a of the Standard TSs would allow entry into Mode 4 from Mode 5 for each of the TS within the scope of the proposed changes, due to the change in action requirements which would now allow indefinite operations in Mode 4. This is based on TSTF-422 implementation guidance of WCAP-16364-NP (Section 2.5). However, the TR has not justified such operations, and therefore the proposed TS changes should include a restriction on applicability of 3.0.4.a.

Response to RAI 3

The applicable portion of LCO 3.0.4.a which allows entry into the Mode or other specified condition in the applicability when an LCO is not met states:

> "When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time";

The Bases for LCO 3.0.4.a states the following:

> "Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation".

Therefore, the Basis for LCO 3.0.4.a is that the Required Actions provide "an acceptable level of safety for continued operation." The applicable Required Actions, in this case are the revised endstate actions that now allow continued operation in Mode 4 instead of Mode 5. WCAP-16294-NP has confirmed for each Technical Specification with a revised endstate action that continued operation in Mode 4 provides an acceptable level of safety. For each affected Technical Specification, WCAP-16294 confirmed that the safety margins are not reduced and addressed defense-in-depth to confirm that continued operation in Mode 4 will not adversely affect safe operation.

In addition, it should be noted that LCO 3.0.4 does not provide an exception to the applicable Required Actions once the Mode of applicability for a Technical Specification is entered. In fact, the Bases for LCO 3.0.4 confirm the applicability of the Required Actions as follows:

> "Upon entry into a MODE or other specified condition in the Applicability with the LCO not met, LCO 3.0.1 and LCO 3.0.2 require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the LCO is met, or until the unit is not within the Applicability of the Technical Specification."

LCO 3.0.1 states:

> "LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8".

LCO 3.0.2 states in part:

> Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

Based on the Technical Specifications usage rules, any restoration or remedial actions applicable to the inoperable equipment must be applied at the time the Mode of applicability is entered. Therefore, in the case in question, entry into Mode 4 from Mode 5 would only be made when there is reasonable assurance that the applicable Required Actions (e.g., restoration to operable status) can be accomplished within the specified Completion Time. The requirement to apply Required Actions when the Mode of applicability is entered provides assurance that the entry into Mode 4 is made under conditions that continue to be controlled by the Technical Specifications and that the reason for not meeting the LCO will be addressed in a timely manner per the Required

Actions. In addition, it should be noted that the Bases of LCO 3.0.4 are clear regarding unreasonable Mode changes. The Bases for LCO 3.0.4 state:

> "The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability."

It is also worth noting that the decay heat load is typically much greater when entering Mode 4 from Mode 3, than when entering Mode 4 from Mode 5. Once a plant has entered Mode 5 or Mode 6, a significant amount of time has passed since the reactor was critical. Thus, the decay heat load in Mode 4 when entered from Mode 5 is significantly less when compared to the typical shutdown of a plant from Mode 1 to Mode 4. With a lower decay heat load when entering Mode 4 from Mode 5, the plant is inherently more stable.

In addition to the Technical Specification requirements discussed above, other regulatory requirements would work to prevent arbitrary or unreasonable Mode changes with unavailable safety-related equipment. For example, extended time in the Mode of applicability while the LCO is not met would increase the unavailability of the affected equipment. 10CFR 50.65 (the Maintenance Rule) requires equipment unavailability to be tracked and plant risk evaluated. Increased equipment unavailability and the resulting potential increase in plant risk are adverse to plant performance metrics which are monitored by the NRC.

NRC RAI 4

A review of the plant operating state (POS) 4 cutsets identified seven apparent issues which may be causing a bias in the overall risk analyses results which favor POS 3 over POS 4. These issues need to be investigated and resolved, and if appropriate explored via sensitivity analyses to demonstrate their impact on the overall conclusions of the TR. Address the following seven issues in the justifications and conclusions of the TR.

a. Several cutsets contain events for failure to start emergency feedwater (EFW) or to align component cooling water (CCW) to the residual heat removal (RHR) heat exchangers. However, these actions should not be required due to the timing of events (i.e., RHR cooling not yet established so EFW would still be running, or RHR cooling established and CCW need not be realigned.) See for example cutsets #1, #11.

   Response to RAI 4 a.

   For POS 4, cutset 1, in Revision 0 of WCAP-16294-NP, the event is initiated by the failure of the operator to align CCW to the RHR heat exchangers during the switchover to RHR cooling (basic event OAH_1IE). For this failure, EFW would not be shut off until RHR is in operation. The operator action to start EFW (D-TRANOPSTRTHE) also appears in cutset 1, which means that there was a failure in the operating train of EFW after the failure to establish RHR cooling.

There is no dependence assumed between operator failures OAH_1IE and D-TRANOPSTRTHE. Assuming no dependence results in a lower value for the cutset, which is conservative when comparing Mode 5 (POS 4) risk to Mode 4 (POS 3). This cutset also includes a failure of an operator action to initiate feed and bleed (OAB2-HS). The dependent operator action OAB2-HS-D1 accounts for the failure of the operating EFW train and the dependency of OAB2-HS-D1 on the previous failed operator actions. While investigating this RAI, it was determined that the intended dependent operator action multiplier was not included in the cutset. The model was adjusted and the revised POS 4 cutsets reflect the intended lower value for OAB-HS-D1.

POS 4, cutset 11 in Revision 0 of WCAP-16294-NP, was investigated in response to this RAI. For this cutset, a loss of inventory occurs. Operator action OAH_1 to align CCW to the RHR heat exchanger also appears in the cutset. It is agreed that the CCW would already be aligned and that this operator action should be removed. The POS 4 PRA model was reviewed for other similar occurrences of this operator action and they were removed. The revised POS 4 cutsets do not contain operator action OAH_1.

b. The offsite power nonrecovery factor of 0.5 over two hours seems overly conservative.

Response to RAI 4 b.

The non-recovery factor of 0.5 over two hours is based on power recovery information in EPRI Technical Report 10029987, "Losses of Offsite Power at U.S. Nuclear Power Plants Through 2001," April 2002. The factor also includes the probability of equipment failing to start after the power recovery and operator action failures. A sensitivity case was run assuming that the power non-recovery factor is 0.1. The revised base case POS 4 results are reduced from 4.03E-06 to 3.47E-06. This is not a large change in CDP for a factor of 5 reduction in the power recovery value. The sensitivity case CDP is still greater than the revised POS 3 base case CDP of 5.95E-07 (see the response to RAI 7 for a discussion of the revised POS 3 results). A brief description of the sensitivity case will be added as Section 6.5.3 to WCAP-16294.

c. The assessment of dependencies between human interactions does not account for sequences which have an intervening successful operator action (which breaks dependency), or involve very long times between events.

Response to RAI 4 c.

In general, operator action dependencies for POS 4 are considered to be either moderate or high unless there is a successful operator action completed between the two in question. Certain combinations of operator actions also require cutset adjustments as described in the response to RAI 4 a. In addition, as described in the RAI 4 a. response, operator action OAH_1 was removed from the POS 4 fault tree which removed some of the dependencies in the cutsets. No dependency was assumed for the new operator action that was added to model the recovery of RHR (see response to RAI 4 d.).

d. The loss of RHR cooling event involving multiple failures (i.e., independent failure of RHR pumps) does not consider the time available to repair and restore the first failure nor reduce the mission time of the second component based on the average time of failure of the first component.

Response to RAI 4 d.

The POS 4 model for the loss of RHR cooling event has been changed. The mission time for RHR pump B was reduced from 70 hours to 35 hours based on the average time to failure for pump A of 35 hours (70 hours/2). An operator action is added to the POS 4 fault tree for restoring RHR. For the loss of RHR sequence where EFW and establishing bleed and feed also fail, an operator action to restore RHR was added. In this sequence, the operator must act quickly to restore RHR. The failure probability was set to 0.5 based on a Pressurized Water Reactor Owners Group study of risk models for plant shutdown modes. The revised POS 4 results, shown in Attachment 2, include these changes. The response to RAI 4 e. also discusses an operator action that was added to the POS 4 model.

e. The loss of RHR cooling event mitigation requires high pressure recirculation for some sequences. The failure of the RHR would seem to preclude the availability of RHR cooling for recirculation. Therefore, it is not clear how the initiating event interacts with this mitigation event. If there is an assumption or a recovery of RHR included in high pressure recirculation then the option to restore RHR cooling could also be credited.

Response to RAI 4 e.

The POS 4 model did not explicitly address recovering RHR. For successful HPR, at least one train of RHR must be available. RHR could have failed in a way that would still allow RHR alignment to the containment sump for recirculation, however, the dominant failures for the loss of RHR involve operator actions, pump failures, and electrical failures. Recovery from these failures would allow RHR to be used for normal cooling. Given that RHR must be recovered to support HPR, and RHR would provide sufficient cooling, only the recovery of RHR is modeled for the cases in which RHR fails, and bleed and feed succeeds. This change was made for other POS 4 initiating events when RHR fails, and bleed and feed cooling succeeds. The failure probability for the recovery of RHR was set to 0.1 based on a Pressurized Water Reactor Owners Group study of risk models for plant shutdown modes. The revised POS 4 results, shown in Attachment 2, include these changes.

f. The loss of RHR cooling event mitigation assumes that the unavailability of high pressure recirculation following successful feed-and-bleed cooling results in core damage. This is overly pessimistic, given that the reactor coolant system (RCS) would be completely filled and subcooled, and there would be substantial time available to restore the RHR system.

Response to RAI 4 f.

The POS 4 model is changed. Refer to the response for RAI 4 e.

g. The loss of inventory event mitigation assumes that if the operator fails to terminate high pressure safety injection flow, then the only core cooling option available is high pressure recirculation (i.e., returning to RHR cooling or secondary cooling is not credited). This is overly pessimistic, given that the RCS would be completely filled and subcooled with the leak isolated, and there would be substantial time available to recognize that the RHR system was available and could be operated.

Response to RAI 4 g.

For the loss of inventory sequences in which the leak is isolated, high pressure injection is successful, but injection is not terminated, the POS 4 model has been revised to include reestablishing RHR or emergency feedwater. This model change is included in the revised results. Revised WCAP-16294 tables with revised POS 4 results are included in Attachment 2.

NRC RAI 5

Section 6.3.1 of the report identifies the system configuration assumptions employed in the quantitative model and in some cases provides information on how plant-specific designs vary. However, there is no basis provided to justify that plant-specific designs are bounded by this analysis. For example:

- Separate low pressure SI pumps and RHR pumps would result in the availability of low pressure SI in Mode 5 to mitigate inventory losses, making the Mode 5 risk lower.

- Designs with a common non-redundant CCW safety-related supply header increase potential for loss of CCW and RHR cooling in Mode 5

- Service water (SW) configurations less robust than assumed (common headers, etc.) similarly would increase loss of RHR cooling in Mode 5, while more robust designs may reduce Mode 5 risk.

- Separate high pressure SI pumps are typically required to be disabled in Mode 5, while combined charging/SI pumps would assure availability of one pump in Mode 5.

- Additional redundancy in emergency diesel generators (EDGs) would enhance mitigation of loss of offsite power, which would improve the risk of Mode 5 preferentially to Mode 4.

- Ability of containment spray or coolers to provide backup cooling during recirculation for core damage mitigation would improve the risk of Mode 5 preferentially to Mode 4.

- A more complete discussion of design variations and assessment of their impact on the quantitative risk analysis, possibly using sensitivity studies, is needed to assure applicability to all Westinghouse plants; otherwise, plant-specific justifications will be required.

Response to RAI 5

To address the plant configurations discussed in this RAI, the following text will replace the last five paragraphs on page 6-18 and the first two paragraphs on page 6-19 of WCAP-16294.

The transition risk PRA model is based on a single unit, therefore, it does not take credit for the availability of shared systems. The AFW system is similar to many Westinghouse NSSS plants. The AFW system does not include a diesel-driven pump that would provide diverse mitigation for loss of offsite power events. Based on the results presented in Table 6-10, the loss of offsite power event is a larger percent contributor to the POS 3 risk than it is to the POS 4 risk and this design difference would be more beneficial for Mode 4 (POS 3).

The ECCS design is also similar to many Westinghouse NSSS plants. The centrifugal charging pumps are used for high pressure safety injection. High pressure safety injection does not include a set of separate safety injection pumps, but the general success criteria of requiring one train is common for Westinghouse NSSS PRA models. Plants with Intermediate Head SI (IHSI) pumps typically require the IHSI pumps to be disabled in Mode 5. The POS 4 PRA model assumes that two trains of charging (SI) pumps are available and the model includes an available swing pump. Therefore, high pressure SI is modeled as being available in the POS 4 PRA model, when it may not be available based on plant procedures. This is a conservative approach because it reduces the risk in Mode 5 when, in fact, the pump may not be available.

Low pressure SI and recirculation is performed by the RHR pumps and heat exchangers. This is similar to many other Westinghouse NSSS plants, although some plants have low pressure SI pumps separate from the RHR system. Separate low pressure SI pumps are not modeled in any of the POS PRA models. The availability of separate low pressure SI pumps would reduce the risk in Mode 5 if modeled for POS 4. However, the POS 3 PRA model does not credit switching to RHR cooling if AFW cooling fails. This is a conservative approach with respect to the estimated risk for POS 3. Modeling separate low pressure SI pumps for POS 4 and modeling RHR cooling for POS 3 would result in lower risk for both plant operating states. Therefore, the conclusions of the WCAP would not change.

The reactor protection system for the transition risk PRA model is based on a solid state protection system. While many Westinghouse NSSS plants have relay protection systems, the reliability of the two systems is not significantly different, therefore, the model is applicable to both protection systems.

The modeled CCW system consists of two separate trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants. Plants with a common CCW header would be expected to have a higher failure probability for the CCW system. This would affect the risk for both the POS 3 and POS 4 PRA models, however, it would have a greater effect on the POS 4 model, because it would directly contribute to the loss of RHR cooling, whereas it does not directly contribute to the loss of AFW cooling for POS 3. Some insight to the relative effect of different CCW designs can be gained by comparing the POS 3 and POS 4 CDPs for the two cases presented in Table 6-14. The table provides the results from modeling one CCW train unavailable. The difference between the two cases is the number of CCW pumps assumed to be unavailable. The POS 4 results for the two cases do not differ by a large amount, and the POS 3 results differ by even less. However, the POS 4 CDP is more than 20 times greater than the POS 3 CDP for these cases. Plant design differences in the CCW system will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The modeled SW system consists of two trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants. Plants with a common SW header would be expected to have a higher failure probability for the SW system. This would affect the risk for both the POS 3 and POS 4 PRA models, however, it would have a greater effect on the POS 4 model because it would contribute to the loss of RHR cooling, whereas it does not directly contribute to the loss of AFW cooling for POS 3. Some insight to the relative effect of different SW designs can be gained by comparing the POS 3 and POS 4 CDPs for the two cases presented in Table 6-15. The table provides the results from modeling one SW train unavailable. The difference between the two cases is the number of SW pumps assumed to be unavailable. The POS 4 results for the two cases do not differ by a large amount and the POS 3 results differ by even less. However, the POS 4 CDP is approximately 10 times greater than the POS 3 CDP for these cases. Plant design differences in the CCW system will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The electric power system modeled is a two train system with one diesel generator per train. This is a common design among the Westinghouse NSSS plants. Some plants have more redundancy in their design, including shared diesels between units. Based on the results presented in Table 6-10, the loss of offsite power event is a larger percent contributor to the POS 3 risk, than it is to the POS 4 risk and a more redundant design would be more beneficial for Mode 4 (POS 3).

The use containment spray or the containment coolers for backup cooling during recirculation has been modeled only to a limited extent for POS 3 for small LOCAs, and does not have a very large effect on the results (see Table 6-13). More detailed modeling to take credit for these systems may result in a larger risk decrease for POS 4 than it would for POS 3 because the probability of losing RHR cooling is greater than the probability of losing AFW cooling. However, the risk for POS 3 would also decrease and the POS 3 risk is approximately a factor of 7 lower than the risk for POS 4 (see Table 6-

9). More detailed modeling will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The model chosen includes many safety system features and support system features that are common among many of the Westinghouse NSSS plants. The evaluation of design differences indicates that while the Westinghouse NSSS plant designs vary for the systems modeled, the model used provides representative results whose conclusions are applicable to all Westinghouse NSSS plants.

NRC RAI 6

In Section 6.3.1.1 for POS 3 plant response model, the bulleted items state that the event mitigation of loss of coolant accidents (LOCAs) is identical to the at-power PRA model "except for the availability of accumulators." This is not explained in the TR and is not understood as to the meaning. Explain and justify this statement.

Response to RAI 6

In Section 6.3.1.1, the bulleted items under "POS 3 and POS 5: Plant Response Model" will be revised, as shown below, to clarify the intent of the statements.

For Large, Medium, and Small LOCA, Loss of Offsite Power, SG Tube Rupture, and Secondary Side Breaks, the notes will be changed to:

> The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

NRC RAI 7

Table 6-9 provides core damage probability (CDP) results for POS 3 and POS 4. A per-hour frequency is also provided by dividing the total probability by the time assumed. It is not stated whether the time-independent contribution of transition risk has been deleted from the results in order to obtain a time-dependent CDP. The first bullet after the table states the time-adjusted CDP is 4x higher in Mode 5 compared to Mode 4, consistent with Table 6-9. Similarly, the results of Table 6-10 do not identify if transition risk for loss of shutdown cooling and loss of inventory include the transition risks. Please clarify the basis for the CDPs in this table.

Response to RAI 7

Table 6-9 will be revised as shown in Attachment 2. The CDP/hour calculation is removed and the core damage probabilities associated with a plant cooldown to POS 3 (Mode 4) and POS 4 (Mode 5) are presented. The approach in Revision 0 of WCAP-16294 has been revised to model a 3 hour POS 3 duration when the cooldown is to POS

4, and a 49 hour POS 3 duration when the cooldown is to POS 3. Table 6-7 is also revised as shown in Attachment 2 to be consistent with the changes.

NRC RAI 8

Section 6.3.2 fourth bullet states that loss of offsite power is a larger risk contributor in Mode 5 than in Mode 4, but Table 6-10 shows these contributions as 10.1 percent in Mode 5, and 12.7 percent in Mode 4. Explain this apparent inconsistency.

Response to RAI 8

The text of the fourth bullet following Table 6-9 will be deleted based on the revised results. Table 6-9 will be replaced as shown in Attachment 2.

NRC RAI 9

In Section 6.4 for the individual LC0 assessments, each individual evaluation includes statements regarding the availability of equipment which is also in the scope of the application. However, there are not constraints in the TS, nor tier 2 restrictions recommended, to assure availability of this equipment. It is not stated directly whether any of the risk analyses to support the individual LCOs are sensitive to the availability of this equipment identified in each individual LCO evaluation, and if so, what measures are appropriate to assure the plant configuration remains bounded by the assessment. Where assessments include statements regarding availability of equipment, include tier 2 restrictions or justify not doing so.

Response to RAI 9

The Basis for Proposed Change and Defense-in-Depth Consideration sections were reviewed for each proposed Technical Specification change and revised to eliminate wording that could be interpreted to require a Tier 2 restriction. See Attachment 2 for the text changes. In addition, the Bases for the Technical Specifications addressed in the WCAP will include a requirement that the auxiliary feedwater turbine-driven pump be available for decay heat removal in Mode 4. This is an important assumption that has been modeled in the Mode 4 (POS 3) PRA model. Note that final Technical Specification markups will be provided separately in the associated TSTF.

Containment and Ventilation Branch Questions

NRC RAI 1

WCAP-16294-NP, Page 6-57/58, Section 6.4.9, TS 3.6.1 B Containment (Atmospheric, Subatmospheric, Ice Condenser, and Dual).

At the top of Page 6-58, first paragraph, the fourth line states "due to the limited time in the shutdown modes". This does not appear to add much to the basis for changing the required action endstate for an inoperable containment from Mode 5 to Mode 4. This TS change would allow for an extended stay in Mode 4 and not necessarily ensure a limited time (defined in Section 6.3.1.2 as a normal or average time from historical data). Inoperability of the containment ranges from a condition where leakage after an accident at full power probably would be slightly higher than that assumed in the accident/dose analysis up to gross leakage potential or containment structural integrity not being reasonably assured even for an accident in Mode 4. The rationale for similarly changing the endstate required in the other TSs appears to rely in part on containment being operable. Although passive, the containment structure is not redundant and allowing an extended stay in Mode 4 for the entire range of containment inoperability vice proceeding to Mode 5 in an orderly fashion as is currently required does not appear to afford the defense-in-depth that would make avoiding a transition to Mode 5 risk/safety beneficial.

Justify the change in required endstate from Mode 5 to Mode 4. It appears to be in a different category than similar changes to the other TSs.

Response to RAI 1

The text of the WCAP will be revised to delete the TS Required Action endstate changes for TS 3.6.1 Required Action B.2, TS 3.6.2 Required Action D.2, and TS 3.6.3 Required Action F.2. By removing these Technical Specification changes from the WCAP, there will be no Required Action endstate changes directly related to containment integrity.

Note that the Technical Specification pages with the finalized changes marked up will be provided separately (from the revised WCAP pages) in the TSTF associated with this WCAP.

NRC RAI 2

WCAP-16294-NP Page 6-99/100 Section 6.4.35, TS 3.7.10, Control Room Emergency Filtration System (CREFS).

At the top of Page 6-100, first paragraph, second sentence states, "If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event must occur along with core damage and containment isolation failure for filtration to be required." The assertion that containment isolation failure would also have to occur for CREFS to be required to maintain Control Room occupant doses less than required limits may not be accurate for many plants.

This qualitative evaluation contains an assertion in its basis rationale that would not appear accurate for many plants. Explain this assertion and its general applicability would be needed for this TS change to be acceptable.

Response to RAI 2

The first paragraph on Page 6-100 will be revised to the text discussed below to make the justification applicable to all Westinghouse NSSS design plants.

> If one CREFS train is inoperable, the other train provides the necessary filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event with a radioactive release must occur for radioactive filtration to be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2 of the WCAP. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event was to occur.
>
> CREFS also provides protection from chemical releases, toxic gas releases, or radiation releases from other sources on-site and offsite. The likelihood of these events occurring are independent of the unit operating Mode.

In addition, the paragraph under Defense-in-Depth Considerations will be revised to:

> If one CREFS train is inoperable, the other train remains available to provide control room filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event and radioactive release must occur for filtration to be required. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Reactor Systems Branch Questions

NRC RAI 1

Pages 5-1, Section 5.1, first bullet states, "A reasonable balance among prevention of core damage, prevention of containment failure, and consequences mitigation is preserved."

Section 2.2.1 of Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," states that, "A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved, i.e., the proposed change in a TS has not significantly changed the balance among these principles of prevention and mitigation, to the extent that such balance is needed to meet the acceptance criteria of the specific design basis accidents and transients, consistent with 10 CFR 50.36. TS change requests should consider whether the anticipated operational changes associated with a TS change could introduce new accidents or transients or could increase the likelihood of an accident or transient (as is required by 10 CFR 50.92)."

Consistent with 10 CFR 50.36, "Technical Specifications" and as required by 10 CFR 50.92, "Issuance of Amendment," please include the TS change (TSTF) required section on "No Significant Hazards Consideration" as listed in 10 CFR 50.92(c)(1),(2), and (3).

Response to RAI 1

The TSTF associated with this WCAP will include a No Significant Hazards Consideration in accordance with 10CFR50.92.

NRC RAI 2

There are missing pages in the TS markups (both in hardcopy and in ADAMS) for TS 3.8.1, AC Sources Operating (Page 3.8.1 -4). Please provide the missing page.

Response to RAI 2

The missing page is included in Attachment 2 and will be included in the TSTF associated with this WCAP.

NRC RAI 3

Page 6-107, Section 6.4.40, TS 3.8.1 - AC Sources - Operating. In 'Condition A and Condition C,' a transformer and a transformer common cause basic event were respectively selected to model the respective conditions. Please state the reason for selecting transformers (and not other components such as breakers, etc.) in the risk models.

Response to RAI 3

The transformer was selected based on a review of a simplified electrical schematic for the electrical system modeled in the PRA. Failing the transformer chosen (setting the basic event to TRUE) failed one offsite power circuit. Other component failures could have been chosen to achieve the same effect in the PRA model. For two offsite power circuits unavailable, the appropriate transformer common cause basic event was selected using the simplified electrical schematic and reviewing the plant PRA model. Similar to the one offsite power circuit case, other basic events in the PRA model could have been chosen to achieve the same effect of the loss of two offsite power circuits.

NRC RAI 4

Page 6-109, Table 6-18, states "The total CDP decreases by greater than a factor of 6 when the unit cooled down to Mode 4 (POS 3) instead of Mode 5 (POS 4)," and it should refer instead to a "factor of 60." It is an apparent editorial error.

Response to RAI 4

Table 6-18 was revised as shown in Attachment 2. The revised format of the table should make the differences between the Mode 4 (POS 3) and Mode 5 (POS4) core damage probabilities more obvious.

APPENDIX D
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION, DATED NOVEMBER 26, 2008

NEI

NUCLEAR ENERGY INSTITUTE

Biff Bradley
DIRECTOR
RISK ASSESSMENT
NUCLEAR GENERATION DIVISION

November 26, 2008

Ms. Tanya M. Mensah
Senior Project Manager
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to NRC Request for Additional Information Regarding PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134).

Project Number: 689

Dear Ms. Mensah:

Attached for NRC review are the responses to the NRC's October 23, 2008, Request for Additional Information (RAI) regarding WCAP-16294-NP, which addresses the endstates risk-informed technical specifications initiative (Initiative 1) for Westinghouse plants. Attachment 1 provides the RAI responses, and Attachment 2 provides proposed markups to WCAP-16294-NP associated with certain RAI responses. A response to previous NRC RAIs on this topical report was provided by NEI's letter to NRC of December 12, 2007.

Please note that the changes in Attachment 2, to WCAP-16294-NP pages 2-1, 6-61, 6-63, 6-64, 6-66, 6-67, 6-69, 6-81, 6-87, 6-90, and 6-92, were made to the markups of those WCAP pages that were transmitted in Attachment 2 of the NEI letter dated December 12, 2007.

The changes to WCAP-16294-NP contained in Attachment 2 of this letter will also be incorporated into the approved version of WCAP-16294-NP, which will be issued as Revision 1, following receipt of the NRC's Final Safety Evaluation. In addition, please note that the final approved version of the WCAP will include related editorial changes such as changes to the table of contents and section number changes due to deleted sections of the WCAP.

Additionally, the Technical Specification and Bases changes justified in WCAP-16294-NP that are included in Attachment B of the WCAP will be deleted from the approved WCAP and will be submitted to the NRC separately via a TSTF traveler.

We appreciate NRC staff's efforts to review this report. Please contact me at 202-739-8083; reb@nei.org should you have any questions.

Sincerely,

Biff Bradley

Attachments

c: Mr. Carl S. Schulten, NRR/ADRO/DIRS/IT, NRC
NRC Document Control Desk

ENCLOSURE 1

NEI letter dated November 25, 2008

Containment and Ventilation Branch Questions

NRC RAI 1

TS 3.6.4A, Containment Pressure (Atmospheric, Dual, and Ice Condenser) and TS 3.6.4B-B, Containment Pressure (Subatmospheric): The basis for the proposed changes for the containment pressure TS use vague language such as "variations in containment pressure are expected to be small, therefore, any increase above the Technical Specification limit is expected to be small...". No discussion is provided for the basis of the statement. There is no indication of the processes that can cause containment pressure to be greater than or lower than the specified range or if (and why) changes from these processes will be rapid or slow. There is no discussion of any automatic system actuations that will occur if the containment pressure is too high or too low. Please provide a more comprehensive basis for the statement "variations in containment pressure are expected to be small, therefore, any increase above the Technical Specification limit is expected to be small..."

Response to RAI 1

The PWROG withdraws the proposed endstate changes to TS 3.6.4A and TS 3.6.4B. WCAP-16294-NP will be revised to delete the discussions associated with these proposed changes. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 2

TS 3.6.4A, Containment Pressure (Atmospheric, Dual, and Ice Condenser) and TS 3.6.4B-B, Containment Pressure (Subatmospheric): The basis provided for the proposed change discussed states that the "minimum technical specification containment pressure is established such that if there was an inadvertent actuation of the containment spray system, the minimum (negative) containment design pressure would not be exceeded. Inadvertent actuation of the containment spray system does not lead to core damage and LERF by itself." Please provide additional information that justifies that inadvertent actuation of containment heat removal systems, with containment pressure less than the minimum TS limit, the containment minimum (negative) design pressure would not be exceeded. Include in the discussion atmospheric containment designs that are not provided with vacuum relief systems.

Response to RAI 2

The PWROG withdraws the proposed endstate changes to TS 3.6.4A and TS 3.6.4B. WCAP-16294-NP will be revised to delete the discussions associated with these proposed changes. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 3

TS 3.6.4A, Containment Pressure (Atmospheric, Dual, and Ice Condenser) and TS 3.6.4B-B, Containment Pressure (Subatmospheric): The minimum TS containment pressure is used as input to determining the performance of the emergency core cooling system (ECCS) pumps. Provide a discussion in the basis for the change that there will be sufficient containment pressure for ECCS operation and that the criteria of 10 CFR 50.46 will be satisfied following a loss-of-coolant (LOCA) in Mode 4 with the containment below the minimum pressure limiting condition for operation (LCO). Include core reflood and ECCS pump net positive suction head available (NPSHa) in the discussion.

Response to RAI 3

The PWROG withdraws the proposed endstate changes to TS 3.6.4A and TS 3.6.4B. WCAP-16294-NP will be revised to delete the discussions associated with these proposed changes. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 4

TS 3.6.15, Ice Bed (Ice Condenser): Please provide a discussion in the basis for the change or the reference to a previously submitted evaluation that provides justification that with Ice Bed inoperable there will be a sufficient source of borated water (via the containment sump) for long-term ECCS operation and that the criteria of 10 CFR 50.46 will be satisfied following a LOCA in Mode 4. Include containment spray operation, ECCS pump vortex, and ECCS pump net positive suction head available (NPSHa) in the discussion.

Response to RAI 4

The PWROG withdraws the proposed endstate changes to TS 3.6.15. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 5

TS 3.6.16, Ice Condenser Doors (Ice Condenser): Please provide a discussion or the reference to a previously submitted evaluation that documents that when the Ice Condenser doors are inoperable-open, there will not be excessive sublimation nor obstruction of flow passages that will render the ice bed inoperable based on TS 3.6.15.

Response to RAI 5

The PWROG withdraws the proposed endstate changes to TS 3.6.16. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 6

TS 3.6.16, Ice Condenser Doors (Ice Condenser): Please provide a discussion that documents that when the Ice Condenser doors are determined to be inoperable in the closed position, for a LOCA while in Mode 4, there will be an adequate source of borated water available to the containment sump for long-term ECCS and containment spray heat removal functions in the recirculation mode. The evaluation should include ECCS pump NPSHa and ECCS pump vortex.

Response to RAI 6

The PWROG withdraws the proposed endstate changes to TS 3.6.16. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 7

TS 3.6.17, Divided Barrier Integrity (Ice Condenser): Please provide a discussion of the basis for the change, or reference to a previously submitted evaluation, which provides justification that with the divided barrier inoperable, in the event of a LOCA while in Mode 4, the pressure in the containment lower compartment will be great enough to open the ice condenser doors permitting the steam air mixture to enter and flow through the ice condenser. If the ice condenser doors will not open, two trains of containment spray will be available for control of containment peak temperature and pressure control. Discuss, or reference, previously submitted documentation that show that containment spray alone without the benefit of the ice condenser is sufficient to control peak containment temperature and pressure. Also discuss, or reference previously submitted documentation that demonstrates that without the melt from the ice condenser, there will be an adequate source of borated water available to the containment sump for long-term ECCS and containment spray heat removal functions in the recirculation mode. The evaluation should include ECCS pump NPSHa and ECCS pump vortex.

Response to RAI 7

The PWROG withdraws the proposed endstate changes to TS 3.6.17. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 8

TS 3.6.5B, Containment Air Temperature (Ice Condenser) and TS 3.6.5C, Containment Air Temperature (Subatmospheric): The bases for proposed changes address temperatures above the maximum temperature LCO. Please provide basis that discusses why it is acceptable to be in Mode 4 with the containment below the minimum temperature LCO.

Response to RAI 8

The PWROG withdraws the proposed endstate changes to TS 3.6.5A, TS 3.6.5B, and TS 3.6.5C. WCAP-16294-NP will be revised to delete the discussions associated with these proposed changes. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 9

TS 3.6.8, Shield Building (Dual and Ice Condenser): Please provide a discussion or the reference to a previously submitted evaluation which documents, with the Shield Building inoperable, both trains Shield Building Air Cleanup System (Dual and Ice Condenser) remains operable.

Response to RAI 9

The PWROG withdraws the proposed endstate changes to TS 3.6.8. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 10

TS 3.6.8, Shield Building (Dual and Ice Condenser): It is not clear if containment vacuum relief systems installed in Westinghouse Dual Containment design plants rely on the shield building for proper operation. If the containment vacuum relief draws air from the annulus between the shield building and the containment the system design may be based on a maximum shield building leakage. Please provide a discussion of any role the shield building may perform in the operation of containment vacuum relief systems in Mode 4.

Response to RAI 10

The PWROG withdraws the proposed endstate changes to TS 3.6.8. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 11

TS 3.6.8, Shield Building (Dual and Ice Condenser): The NEI response to NRC Request for Additional Information Regarding PWROG TR WCAP-16294-NP, Revision 0, dated December 12, 2007, revised the Defense-in-Depth Considerations for TS 3.6.8 (TR page 6-81). The change does not appear to be related to any specific RAI. Provide an explanation for the revision and why the change is acceptable.

Response to RAI 11

The PWROG withdraws the proposed endstate changes to TS 3.6.8. WCAP-16294-NP will be revised to delete the discussion associated with this proposed change. Attachment 2 contains the changes to WCAP-16294-NP that reflect these changes.

NRC RAI 12

TR WCAP-16294-NP, Revision 0, states that when in Mode 4 the secondary side steam pressure will be at normal operating pressure. Please verify that when the reactor coolant system (RCS) average temperature is decreased from approximately 560°F in Mode 1 to less than 350°F in Mode 4 the pressure in the secondary side remains at normal operating pressure. If secondary side pressure is not at normal operating pressure in Mode 4 please provide verification that there will be sufficient pressure to operate the turbine driven auxiliary feedwater pump. If secondary side steam pressure in Mode 4 will be less than normal operating pressure in Mode 1 please update all references in TR WCAP-16294-NP, Revision 0, and in the RAI responses.

Response to RAI 12

The secondary side steam pressure decreases as the RCS Tavg is reduced from the hot full power RCS Tavg in Mode 1, to the RCS Tavg in Mode 4. Therefore the secondary side steam pressure in Mode 4 will be less than the normal operating secondary side steam pressure in Mode 1. The turbine driven auxiliary feedwater (TDAFW) pump is designed to operate with a steam pressure of approximately 92 psia for some plants. It should be noted that this change does not impact the evaluations and conclusions contained in WCAP-16294-NP. WCAP-16294-NP will be revised as identified in the mark-ups in Attachment 2 to reflect a reduced secondary side steam pressure in Mode 4.

The requirement to maintain the TDAFW train available to remain in Mode 4 will be addressed by a licensing commitment to include a Tier 2 requirement that will be located in the Technical Specification Bases, a Licensee Controlled Document, or implementing procedures. The addition of this Tier 2 requirement would be contained in the Implementation Guidance that will be issued for implementing this change. If a sufficient steam supply is unavailable during the application of the Mode 4 end state, the unit's configuration risk management program will direct the appropriate actions to be taken. The risk impact is managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." Regulatory Guide 1.182 endorses the guidance in Section 11 of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." This program will determine the safest course of action for an emergent condition in Mode 4 that renders the TDAFW train unavailable, and could include proceeding to a Mode 5 endstate in the affected Specifications.

Attachment 2

WCAP-16294-NP Markups to Address NRC RAIs Dated 10/23/08

2 TECHNICAL SPECIFICATIONS AND CHANGE REQUEST

The Technical Specification Required Action endstates evaluated for the endstate change are contained in NUREG-1431, "Standard Technical Specifications for Westinghouse Plants" (Reference 3). Technical Specification number, title, Condition, current endstate, and the proposed endstate are provided in Table 2-1.

Table 2-1 Proposed Changes to Endstates

Technical Specification/ Condition	Title	Current Endstate	Proposed Endstate
3.3.2-B	ESFAS Instrumentation	5	4
3.3.2-C	ESFAS Instrumentation	5	4
3.3.2-K	ESFAS Instrumentation	5	4
3.3.7-C	Control Room Emergency Filtration System Actuation Instrumentation	5	4
3.3.8-D	Fuel Building Air Cleanup System Actuation Instrumentation	5	4
3.4.13-B	RCS Operational Leakage	5	4
3.4.14-B	RCS Pressure Isolation Valve Leakage	5	4
3.4.15-E	RCS Leakage Detection Instrumentation	5	4
3.5.3-C	Emergency Core Cooling System - Shutdown	5	4
3.5.4-C	Refueling Water Storage Tank	5	4
~~3.6.1-B~~	~~Containment (Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~	~~5~~	~~4~~
~~3.6.2-D~~	~~Containment Air Locks~~	~~5~~	~~4~~
~~3.6.3-F~~	~~Containment Isolation Valves~~	~~5~~	~~4~~
~~3.6.4A-B~~	~~Containment Pressure (Atmospheric, Dual and Ice Condenser)~~	~~5~~	~~4~~
~~3.6.4B-B~~	~~Containment Pressure (Subatmospheric)~~	~~5~~	~~4~~
~~3.6.5A-B~~	~~Containment Air Temperature (Atmospheric and Dual)~~	~~5~~	~~4~~
~~3.6.5B-B~~	~~Containment Air Temperature (Ice Condenser)~~	~~5~~	~~4~~
~~3.6.5C-B~~	~~Containment Air Temperature (Subatmospheric)~~	~~5~~	~~4~~
3.6.6A-B	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6A-E	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6B-F	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4

[NEI Letter, 12/12/07]

Table 2-1 (cont.) Proposed Changes to Endstates

Technical Specification/ Condition	Title	Current Endstate	Proposed Endstate
3.6.6C-B	Containment Spray System (Ice Condenser)	5	4
3.6.6D-B	Quench Spray System (Subatmospheric)	5	4
3.6.6E-F	Recirculation Spray System (subatmospheric)	5	4
3.6.7-B	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)	5	4
~~3.6.8-B~~	~~Shield Building (Dual and Ice Condenser)~~	~~5~~	~~4~~
3.6.11-B	Iodine Cleanup System (Atmospheric and Subatmospheric)	5	4
3.6.12-B	Vacuum Relief Valves (Atmospheric and Ice Condenser)	5	4
3.6.13-B	Shield Building Air Cleanup System (Dual and Ice Condenser)	5	4
3.6.14-B	Air Return System (Ice Condenser)	5	4
~~3.6.15-B~~	~~Ice Bed (Ice Condenser)~~	~~5~~	~~4~~
~~3.6.16-D~~	~~Ice Condenser Doors (Ice Condenser)~~	~~5~~	~~4~~
~~3.6.17-C~~	~~Divided Barrier Integrity (Ice Condenser)~~	~~5~~	~~4~~
3.6.18-C	Containment Recirculation Drains (Ice Condenser)	5	4
3.7.7-B	Component Cooling Water System	5	4
3.7.8-B	Service Water System	5	4
3.7.9-C	Ultimate Heat Sink	5	4
3.7.10-C	Control Room Emergency Filtration System	5	4
3.7.11-B	Control Room Emergency Air Temperature Control System	5	4
3.7.12-C	ECCS Pump Room Exhaust Air Cleanup System	5	4
3.7.13-C	Fuel Building Air Cleanup System	5	4
3.7.14-C	Penetration Room Exhaust Air Cleanup System	5	4
3.8.1-G	AC Sources – Operating	5	4
3.8.4-D	DC Sources – Operating	5	4
3.8.7-B	Inverters – Operating	5	4
3.8.9-D	Distribution Systems – Operating	5	4

Table 6-1 Key Plant Parameters by Technical Specification Mode

Parameter	Mode 5 to Mode 4	Mode 4 to Mode 3	Mode 3 to Mode 2	Mode 2 to Mode 1	Mode 1
Technical Specification RCS Average Temperature	≤200°F (Mode 5, Cold Shutdown)	>200°F to <350°F (Mode 4, Hot Shutdown)	≥350°F (Mode 3, Hot Standby)	NA (Mode 2, Startup)	NA (Mode 1, Power)
Technical Specification Reactor Power Level	NA (Mode 5)	NA (Mode 4)	NA (Mode 3)	≤5% (Mode 2)	>5% (Mode 1)
RCS Average Temperature	~185°F to ~330°F	~330°F to ~557°F	~557°F	~557°F	~557°F
RCS Pressure	~340 psig	~340 psig to ~2235 psig	~2235 psig	~2235 psig	~2235 psig
Pressurizer Status	Water solid to bubble	Bubble	Bubble	Bubble	Bubble
Secondary Side Pressure	0 psig	Less than Normal operating pressure	Normal operating pressure	Normal operating pressure	Normal operating pressure

The following discussion centers around several plant operating states (POS). The POS approach is used as the basis for the qualitative discussion to be consistent with the quantitative transition PRA model, that is discussed in Section 6.3.1. A POS is a unique plant configuration defined by a set of parameters. For each POS, a unique set of initiating events, plant conditions, and systems available for event mitigation can be identified. Very often a plant will be in a POS for a relatively short period of time, because switching the plant configuration (system re-alignments) is required to reach the desired endstate. To identify the POSs, an understanding of the typical key activities that are in progress as the plant shuts down and restarts is required. The following provides a general summary of these activities:

Modes 1-2

- Decrease power (Mode 1)
- Take the turbine off-line (Mode 1)
- Transfer from MFW (main feedwater) to AFW (note that some plants may continue on MFW depending on their MFW design and approach to plant shutdown) (Mode 1)
- Mode 2 (startup) when power level is ≤ 5%

Table 6-3 Important Parameters for Mode Target Conditions

Parameter	Mode 1	Mode 2	Mode 3	Mode 4	Mode 5
RCS Temperature	557°F	557°F	557°F	~330°F	170°F to 190°F
RCS Pressure	2235 psig	2235 psig	2235 psig	340 psig	250 psig
Secondary Side Status	Normal operating pressure	Normal operating pressure	Normal operating pressure	Normal operating pressure	Low pressure
PZR Status	Bubble	Bubble	Bubble	Bubble	Bubble
(Decay) Heat Removal Mode	MFW	AFW	AFW	AFW	RHR
Power Level	100%	5%	0%	0%	0%

Less than

Table 6-4 Plant Operating States (Power Operation to Cold Shutdown)

State	POS 1	POS 2	POS 3	POS 4
Plant Mode	1 (transition only) 2 3 (upper part)	3 (middle part)	3 (lower part) 4 (upper part)	4 (lower part) 5 (upper part)
RCS Temperature	557°F	557°F to XX°F[1]	XX°F[1] to 340°F	340°F to 180°F
RCS Pressure	2235 psig	2235 psig to 950 psig	950 psig to 365 psig	365 psig to 250 psig
Pressurizer	Bubble	Bubble Less than	Bubble	Bubble
Secondary Side	Normal operating pressure	Normal operating pressure	Normal operating pressure	Low pressure (shutdown)
Activities	• AFW for decay heat removal • Reduce power • Switch from MFW to AFW • Borate • Insert control rods • Take turbine off-line	• AFW for decay heat removal and cooldown • Open trip breakers • Reduced operating RCPs • Block SI and SLI • RCS cooldown	• AFW for decay heat removal and cooldown • Reduced operating RCPs • Isolate accumulators • RCS cooldown • Start secondary side cooldown	• RHR for decay heat removal and cooldown • Switch to RHR cooling • Disable SI pumps • Defeat AFW start signals • Cold overpressure protection (COP) in service
System Status	• AFW operating • All systems available	• AFW operating • All systems available • SI and SLI signals blocked • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Accumulators isolated • Reactor trip breakers open	• RHR operating • SI, SLI, and AFW signals blocked • Accumulators isolated • SI pumps disabled • COP in service • Reactor trip breakers open

Note:

1. A defined temperature is not important to this analysis.

Table 6-5 Plant Operating States (Cold Shutdown to Power Operation)

State	POS 4	POS 5	POS 6	POS 7
Plant Mode	4 (lower part) 5 (upper part)	3 (lower part) 4 (upper part)	3 (middle part)	1 (transition only) 2 3 (upper part)
RCS Temperature	180°F to 340°F	340°F to XX°F[1]	XX°F[1] to 557°F	557°F
RCS Pressure	250 psig to 365 psig	365 psig to 950 psig	950 psig to 2235 psig	2235 psig
Pressurizer	Bubble	Bubble Less than	Bubble	Bubble
Secondary Side	Low pressure (shutdown)	Normal operating pressure	Normal operating pressure	Normal operating pressure
Activities	• RHR for decay heat removal • Switch to AFW cooling • Establish AFW actuation signals • RCS heatup	• AFW for decay heat removal • One RCP running[2] • RCS heatup • Start secondary side heatup	• AFW for decay heat removal • One RCP running[2] • RCS heatup • Establish SI and SLI signals • Un-isolate accumulators	• AFW for decay heat removal • Switch from AFW to MFW • Withdraw shutdown and control rods • Bring turbine on-line • Close trip breakers • All RCPs running • Increase power
System Status	• RHR operating • SI, SLI, and AFW signals blocked • Accumulators isolated • SI pumps disabled • COP in service • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Accumulators isolated • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Reactor trip breakers open	• AFW to MFW • All systems available

Notes:

1. A defined temperature is not important to this analysis.
2. If the rods are not capable of withdrawal.

6.2.2 Comparison of Endstates

To achieve Mode 4 as an endstate, the plant will need to transition through POS 1, POS 2, and into POS 3. To achieve Mode 5 as an endstate, the plant will need to transition through POS 1, POS 2, POS 3, and into POS 4. With either endstate, the plant will need to transition through POS 1, POS 2 and into POS 3. To determine the appropriate endstate (Mode 4 vs. Mode 5), the additional risk for the transition through POS 3 and into POS 4 needs to be considered, as well as the risk of remaining in POS 3 (Mode 4) as opposed to POS 4 (Mode 5).

Several of the key differences between POS 3 and 4 are:

- The frequency of loss of decay heat removal is at an increased level with POS 4 as the endstate due to the system re-alignments required. Loss of decay heat removal events in POS 3 can be addressed with AFW (all pumps available) or the RHR system following depressurization of the RCS. In POS 4, the TD AFW pump will not be available to address similar events. In addition, the automatic AFW start signal is available in POS 3, but not POS 4. Therefore, additional options are available in POS 3 for decay heat removal.

- The frequencies of loss of inventory (LOCA) events can be at an increased level with POS 4 as the endstate due to the system re-alignments required. Loss of inventory events in POS 3 can be addressed with the available train of ECCS. In POS 4 a full train of ECCS is not available. The SI pumps are out of service. Inventory control is dependent on the charging system. Therefore, additional options are available in POS 3 for inventory control.

- Mitigation of loss of offsite power (LOSP)/station blackout (SBO) events in POS 3 can be provided by the AFW system including the turbine-driven pump. Availability of the turbine-driven pump is particularly important in case the event degrades to a station blackout. In POS 4, the AFW system turbine-driven pump will not be available for decay heat removal, and the plant will be dependent on restoring electric power to the RHR system. Again, additional options are available in POS 3 for event mitigation.

- The cold overpressurization event needs to be considered in POS 4, but not in POS 3. Although not a large risk contributor, this event is addressed in POS 4.

- Secondary side breaks are considered in POS 3, but not in POS 4. In POS 3 the secondary side will be less than ~~may be near the~~ normal operating pressure, but in POS 4 this pressure is greatly reduced, reducing the likelihood of a secondary side break. Secondary side breaks are not typically large contributors to risk, therefore, this assumption for POS 4 has a small risk impact.

- Risk in the shutdown modes is very dependent on electric power availability. There are more required independent sources of electrical power in POS 3 than in POS 4 and there are more potential activities in POS 4 that could cause a loss of offsite power.

- In POS 3, there is more redundancy and diversity of mitigating and support systems required to be available than there is in POS 4.

- Loss of Decay Heat Removal: This event is the failure of the decay heat removal source which is either the auxiliary feedwater system or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: Event mitigation is identical to that modeled in the at-power PRA model.

- SG Tube Rupture: Event mitigation is identical to that modeled in the at-power PRA model.

- Secondary Side Breaks: Event mitigation is identical to that modeled in the at-power PRA model.

POS 3 and POS 5: Plant Response Model

In POS 3, the RCS pressures and temperatures are significantly reduced compared to POS 1 and POS 2. Therefore, the LOCA events remain applicable, but at a reduced frequency. The secondary side ~~remains~~ is less than ~~at~~ normal operating pressure. Similar to POS 2, the reactor trip breakers are open so rod withdrawal is no longer a possible event. Also, like POS 2, signals for safety injection and steamline isolation are blocked, therefore, operator actions are required to start equipment to mitigate a number of the potential events. Accumulators are isolated. Again, loss of feedwater control is no longer an issue. The at-power plant PRA model is applicable, with several modifications, to model this POS. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: Event mitigation is identical to that modeled in the at-power PRA model except for the availability of accumulators.

- Medium LOCA: Event mitigation is identical to that modeled in the at-power PRA model except for the availability of accumulators.

- Small LOCA: Event mitigation is identical to that modeled in the at-power PRA model except for the availability of accumulators.

- Loss of Decay Heat Removal: This event is the failure of the decay heat removal source which is either the auxiliary feedwater system or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: Event mitigation is identical to that modeled in the at-power PRA model.

- SG Tube Rupture: Event mitigation is identical to that modeled in the at-power PRA model.

- Secondary Side Breaks: Event mitigation is identical to that modeled in the at-power PRA model.

[NEI Letter, 12/12/07]

Initiating events during the shutdown process are less severe for containment than those at-power because of the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and the reduced likelihood of a LOCA or secondary side break due to the limited time in the shutdown modes and less severe thermal-hydraulic conditions. Some of the containment penetration lines have a small enough diameter such that they would not contribute to LERF even if all isolation capability for the line is inoperable. A unit shutdown to Mode 5 requires switching to RHR cooling which introduces the potential for increased risks including LOCAs both inside and outside containment. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

DELETE

The proposed change to the Required Action F.2 endstate does not change the operability requirement for the containment isolation valves. The valves must still be operable in Mode 4. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Most containment penetration lines have two isolation valves and it is unlikely that both would be inoperable. In the unlikely event that the actions cannot be completed in time and Condition F is entered, placing the unit in Mode 5 does not increase the equipment available for event mitigation. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are required to be operable. In addition, some of the containment penetration lines have a small enough diameter such that their contribution to LERP would be insignificant. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.12 Technical Specification 3.6.4A – Containment Pressure (Atmospheric, Dual, and Ice Condenser)

Description

Containment pressure is a process variable that is monitored and controlled. The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a LOCA or steam line break. These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the containment spray system.

DELETE

Limiting Condition for Operation

Containment pressure shall be ≥ [-0.3] psig and ≤ [+1.5] psig.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

DELETE

Containment pressure is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment pressure. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The upper containment pressure limit is based on the Mode 1 design basis analyses. These analyses verify that the containment design pressure is not exceeded for a double-ended guillotine break of either the RCS or main steam piping. The containment design pressure is typically a factor of 2 or more below the containment failure pressure. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, containment loadings will be less than in Mode 1 and well below the design pressure and there will be significant margin to the failure pressure. Variations in containment pressure are expected to be small, therefore, any increase above the Technical Specification limit is expected to be small and it is concluded that there will still be sufficient margin to the design basis pressure and significant margin to the failure pressure.

The minimum Technical Specification containment pressure is established such that if there was an inadvertent actuation of the containment spray system, the minimum (negative) containment design pressure would not be exceeded. Inadvertent actuation of the containment spray system does not lead to core damage and LERF by itself, another event needs to occur to cause the core damage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

Defense-in-depth is maintained by the margin to containment failure in Mode 4. The containment pressure limit is based on Mode 1 design basis analyses that include higher energy releases than would occur in Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are available ~~required to be operable.~~ In addition, containment vacuum relief valves and the containment purge system could be used to mitigate containment pressure being outside of the Technical Specification limits. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.13 Technical Specification 3.6.4B – Containment Pressure (Subatmospheric)

Description

Containment air partial pressure is a process variable that is monitored and controlled. The containment air partial pressure is maintained as a function of refueling water storage tank temperature and service water temperature to ensure that, following a design basis accident, the containment would depressurize in less than 60 minutes to subatmospheric conditions. Controlling containment partial pressure within prescribed limits also prevents the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of an inadvertent actuation of the quench spray system.

DELETE

Limiting Condition for Operation

Containment air partial pressure shall be ≥ [9.0] psia and within the acceptable operation range shown on Figure 3.6.4B-1.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

Containment air partial pressure is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air partial pressure. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The upper containment pressure limit is based on the Mode 1 design basis analyses. These analyses verify that the containment design pressure is not exceeded for a double-ended guillotine break of either the RCS or main steam piping. The containment design pressure is typically a factor of 2 or more below the containment failure pressure. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, containment loadings will be less than in Mode 1 and well below the design pressure and there will be significant margin to the failure pressure. Variations in containment pressure are expected to be small, therefore, any increase above the Technical Specification limit is expected to be small and it is concluded that there will still be sufficient margin to the design basis pressure and significant margin to the failure pressure.

The minimum Technical Specification containment pressure is established such that if there was an inadvertent actuation of the quench spray system, the minimum (negative) containment design pressure would not be exceeded. Inadvertent actuation of the quench spray system does not lead to core damage and LERF by itself, another event needs to occur to cause the core damage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

DELETE

Defense-in-depth is maintained by the margin to containment failure in Mode 4. The containment pressure limit is based on Mode 1 design basis analyses that include higher energy releases than would occur in Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., quench spray, containment cooling) are ~~required to be operable~~ available. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.14 Technical Specification 3.6.5A – Containment Air Temperature (Atmospheric and Dual)

Description

The containment structure serves to contain radioactive material that may be released from the reactor core following a design basis accident. The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for LOCA or steam line break. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis.

Limiting Condition for Operation

Containment average air temperature shall be ≤ [120]°F.

Applicability

DELETE

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

Containment air temperature is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air temperature. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The containment air temperature limit is based on the Mode 1 design basis analyses and containment equipment qualification requirements. The containment air temperature may exceed the design limit for a short period of time, however, the equipment surface temperatures remain below the design limit. The containment air temperature is also an important initial assumption for calculating the peak containment pressure during a LOCA or main steam line break. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, the containment temperature will be well below the design temperature and there will be significant margin to the design temperature. In the shutdown modes, the containment air temperature is not expected to be high because of lower RCS and steam generators temperatures. Variations in containment air temperature are expected to be small, therefore, any change that exceeds the Technical Specification limit is expected to be small and it is concluded that there will still be sufficient margin to the design basis temperature. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

DELETE

Defense-in-Depth Considerations

Defense-in-depth is maintained by the margin to the containment design air temperature limit that is available in Mode 4. The containment design air temperature limit is based on the Mode 1 design basis analyses that include higher energy releases than would occur for Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are available ~~required to be operable~~. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.15 Technical Specification 3.6.5B – Containment Air Temperature (Ice Condenser)

Description

The containment structure serves to contain radioactive material that may be released from the reactor core following a design basis accident. The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for LOCA or steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used.

Limiting Condition for Operation

Containment average air temperature shall be ≥ [85]°F and ≤ [110]°F for the containment upper compartment and ≥ [100]°F and ≤ [120]°F for the containment lower compartment.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

DELETE

Basis for Proposed Change

Containment air temperature is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air temperature. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The containment temperature limits are based on the Mode 1 design basis analyses and containment equipment qualification requirements. The containment air temperature may exceed the design limit for a short period of time, however, the equipment surface temperatures remain below the design limit. The containment temperature is also an important initial assumption for calculating the peak containment pressure during a LOCA or main steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, the containment temperature will be well below the design temperature and there will be significant margin to the design temperature. In the shutdown modes, the containment temperature is not expected to be high because of lower RCS and steam generators temperatures. Variations in containment temperature are expected to be small, therefore, any change that falls outside of the Technical Specification limits is expected to be small and it is concluded that there will still be sufficient margin to the design basis temperature limit. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

Defense-in-depth is maintained by the margin to the containment design air temperature limit that is available in Mode 4. The containment air temperature limit is based on the Mode 1 design basis analyses that include higher energy releases than would occur for Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray) are available ~~required to be operable.~~ Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.16 Technical Specification 3.6.5C – Containment Air Temperature (Subatmospheric)

Description

The containment structure serves to contain radioactive material that may be released from the reactor core following a design basis accident. The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for LOCA or steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used.

Limiting Condition for Operation

Containment average air temperature shall be ≥ [86]°F and ≤ [120]°F.

Applicability

DELETE

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

Containment air temperature is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air temperature. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The containment temperature limits are based on the Mode 1 design basis analyses and containment equipment qualification requirements. The containment air temperature may exceed the design limit for a short period of time, however, the equipment surface temperatures remain below the design limit. The containment temperature is also an important initial assumption for calculating the peak containment pressure during a LOCA or main steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, the containment temperature will be well below the design temperature and there will be significant margin to the design temperature. In the shutdown modes, the containment temperature is not expected to be high because of lower RCS and steam generators temperatures. Variations in containment temperature are expected to be small, therefore, any change that falls outside of the Technical Specification limits is expected to be small and it is concluded that there will still be sufficient margin to the design basis temperature limit. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations DELETE

Defense-in-depth is maintained by the margin to the containment design air temperature limit that is available in Mode 4. The containment air temperature limit is based on the Mode 1 design basis analyses that include higher energy releases than would occur for Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., quench spray, recirculation spray) are ~~required~~ available ~~to be operable~~. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.17 Technical Specification 3.6.6A – Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

Description

The containment spray and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits.

The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of SW. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units are operating. The fans are normally operated at high speed with SW supplied to the cooling coils. In post accident operation following an actuation signal, the containment cooling system fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere.

Limiting Condition for Operation

Two containment spray trains and [two] containment cooling trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Defense-in-Depth Considerations

The spray additive system is designed for accident conditions initiated at power. The containment spray system will remove some iodine from the containment atmosphere without the additive system and two trains of containment spray are required to be operable. The spray additive system also serves to provide the proper pH in the containment sump. For most containments, a backup system for containment sump pH is not available, but proceeding to Mode 5 does not increase the protection available. Note that the ice condenser containments have ice that is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere and minimizes the occurrence of the chloride and caustic stress corrosion on mechanical systems and components. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.22a Recirculation Fluid pH Control System

Some Westinghouse NSSS plants have replaced the spray additive system with a passive ECCS recirculation fluid pH control system. Although the Technical Specification for this system is not contained in NUREG-1431, the endstate is Mode 5 if the system is inoperable, and the Required Action and associated Completion Time are not met. The system consists of baskets in the containment sump with a specified amount of trisodium phosphate in each basket. The trisodium phosphate dissolves when the containment sump level increases to the level of the baskets.

It is highly unlikely that all of the baskets would be empty, therefore, an inoperable recirculation fluid pH control system would still provide some pH control. The justification for changing the endstate to Mode 4 for Technical Specification 3.6.7, "Spray Additive System," is also applicable to the recirculation fluid pH control system, since they perform the same function.

The recirculation fluid pH control system Technical Specification currently requires the unit to be in Mode 3 in 6 hours and Mode 5 in 84 hours if the system is inoperable, and the Required Action and associated Completion Time are not met. The current Mode 5 endstate is proposed to be changed to require the unit to be in Mode 4 in 60 hours if the Required Action and associated Completion Time are not met.

6.4.23 Technical Specification 3.6.8 – Shield Building (Dual and Ice Condenser)

Description

DELETE

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annulus that collects containment leakage than may occur following a loss of coolant accident, steam line break, or control rod ejection.

Limiting Condition for Operation

The shield building shall be operable.

DELETE

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The shield building Technical Specification has no impact on CDP. Shield building integrity is not modeled in the risk models described in Section 6.3.1, therefore, a qualitative evaluation is performed for this proposed endstate change.

Significant leakage from the containment vessel and the shield building in Mode 4 is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and the reduced likelihood of a LOCA or secondary side break due to reduced thermal-hydraulic conditions. In Level 2 PRA models, shield building and containment vessel leakage is not considered to contribute to LERF. A unit shutdown to Mode 5 requires transferring to RHR cooling which introduces the potential for increased risks including LOCAs both inside and outside containment. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

~~The proposed change to the Required Action B.2 endstate does not change the operability requirement for the shield building. The shield building must still be operable in Mode 4.~~ In Mode 4, two trains of containment spray are ~~required to be operable~~ available to mitigate radioactive releases after an event. Significant leakage from the containment vessel and the shield building is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, the reduced likelihood of a LOCA in the shutdown modes, and reduced thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

[NEI Letter, 12/12/07]

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the air return system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one air return train is inoperable, the other train is available to assist in cooling the containment atmosphere. In addition, two trains of containment spray are available ~~required to be operable~~. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

[NEI Letter, 12/12/07]

Defense-in-Depth Considerations

If one air return train is inoperable, the other train is available to assist in cooling the containment atmosphere. Containment cooling is still available from the containment ice condenser and from two trains of containment spray ~~that are required to be operable~~. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.28 Technical Specification 3.6.15 – Ice Bed (Ice Condenser)

Description

DELETE

The ice absorbs energy and limits containment peak pressure and temperature during a design basis accident. The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The ice bed consists of over [2,721,600] lb of ice stored in [1944] baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a design basis accident in containment.

The ice baskets contain the ice within the ice condenser. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from the steam to the ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a design basis accident.

In the event of a design basis accident, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condenser limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a design basis accident and the additional heat loads that would exist in containment during several hours following the initial blowdown. As ice melts, the water passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to serve as a large source of borated water (via the containment sump) for long term ECCS and containment spray system heat removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission product iodine that may be released from the core during a design basis accident. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the containment spray system. The ice is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere. The alkaline pH also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and containment spray system fluids in the recirculation mode of operation.

Limiting Condition for Operation

The ice bed shall be operable.

DELETE

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the ice bed because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. Two phenomena that can degrade the ice bed during power operation are the loss of ice by melting or sublimation and the obstruction of flow passages through the ice bed due to buildup of frost or ice. Both of these degrading phenomena are reduced by minimizing air leakage into and out of the ice condenser. Due to the very large mass of stored ice, if the ice bed is inoperable, it would still retain cooling capability and the containment spray system and the air return system would also provide cooling for accident conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations DELETE

Due to the very large mass of stored ice, if the ice bed is inoperable, it is highly unlikely that it would have no contribution to containment cooling if an event occurred. In addition, two trains of the containment spray system and two trains of the air return system provide cooling capability for accident conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.29 Technical Specification 3.6.16 – Ice Condenser Doors (Ice Condenser)

Description

The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to seal the ice condenser from air leakage and open in the event of a design basis accident to direct the hot steam air mixture into the ice bed, where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient. The ice baskets contained in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice.

In the event of a design basis accident, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condensers limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

Limiting Condition for Operation

The ice condenser inlet doors, intermediate deck doors, and top deck [doors] shall be operable and closed.

DELETE

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A or C not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A or C not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the ice condenser because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. There are a series of ice condenser doors at each level (lower, intermediate, and upper). It is highly unlikely that a sufficient number of doors would be inoperable to the extent that they would render the ice condenser ineffective for cooling containment during accident conditions. In addition, the containment spray system and the air return system provide cooling capability. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. A cooldown to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one or more ice condenser doors are inoperable such that the ice condenser is degraded for cooling containment during accident conditions, two trains of the containment spray system and two trains of the air return system are available ~~required~~ to ~~be operable and would~~ provide cooling capability. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event that would challenge containment integrity occurring in Mode 4 is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.30 Technical Specification 3.6.17 – Divider Barrier Integrity (Ice Condenser)

DELETE

Description

The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a design basis accident. This ensures that most of the gases pass through the ice bed, which condenses the steam and limits pressure and temperature during the accident transient.

Limiting Condition for Operation

Divider barrier integrity shall be maintained.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the divider barrier because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. It is unlikely that the divider barrier would create a large enough bypass of the ice condenser to render the ice condenser completely ineffective during accident conditions. During accident conditions, the containment spray system and the air return system will also provide cooling capability. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

DELETE

It is unlikely that the divider barrier would create a large enough bypass of the ice condenser to render the ice condenser completely ineffective during accident conditions. Two trains of the containment spray system and two trains of the air return system are ~~required to be operable and would~~ available to provide cooling capability during accident conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.31 Technical Specification 3.6.18 – Containment Recirculation Drains (Ice Condenser)

Description

The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. [Twenty of the 24] ice condenser bays have a floor drain at the bottom to drain the melted ice into the lower compartment (in the [4] bays that do not have drains, the water drains through the floor drains in the adjacent bays). A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, the check valve opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it drains to the floor and provides a source of borated water at the containment sump for long term use by the ECCS and the containment spray system during the recirculation mode of operation.

The two refueling canal drains are at low points in the refueling canal. In the event of a design basis accident, the refueling canal drains are the main return path to the lower compartment for containment spray system water sprayed into the upper compartment.

Limiting Condition for Operation

The ice condenser floor drains and the refueling canal drains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

APPENDIX E
DRAFT REVISIONS TO PWROG TOPICAL REPORT WCAP-16294-NP, REVISION 0, DATED MARCH 13, 2009

NUCLEAR ENERGY INSTITUTE

Biff Bradley
DIRECTOR
RISK ASSESSMENT
NUCLEAR GENERATION DIVISION

March 13, 2009

Ms. Tanya M. Mensah
Senior Project Manager
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Draft Revisions to PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134)

Project Number: 689

Dear Ms. Mensah:

Attached for NRC review is a copy of WCAP-16294-NP that reflects revisions made as a result of discussions between the NRC and industry during a February 24, 2009 teleconference to discuss responses to the NRC's October 23, 2008 Request for Additional Information. The changes to WCAP-16294-NP contained in the attachment to this letter will be incorporated into the approved version of WCAP-16294-NP, which will be issued as Revision 1, following receipt of the NRC's Final Safety Evaluation.

We appreciate NRC staff's efforts to review this report. Please contact me if you should have any questions.

Sincerely,

Biff Bradley

Biff Bradley

Enclosure

c: Mr. Carl S. Schulten, U.S. Nuclear Regulatory Commission
NRC Document Control Desk

Westinghouse Non-Proprietary Class 3

WCAP-16294-NP August 2005

Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-16294-NP

Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs

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Work Performed Under Shop Order MUHP-3015

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Westinghouse Owners Group
Member Participation* for WOG Project/Task MUHP-3015

Utility Member	Plant Site(s)	Participant	
		Yes	No
AmerenUE	Callaway (W)	✓	
American Electric Power	D.C. Cook 1 & 2 (W)	✓	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)		✓
Constellation Energy Group	Calvert Cliffs 1 & 2 (CE)		✓
Constellation Energy Group	Ginna (W)	✓	
Dominion Connecticut	Millstone 2 (CE)		✓
Dominion Connecticut	Millstone 3 (W)	✓	
Dominion VA	North Anna 1 & 2, Surry 1 & 2 (W)	✓	
Duke Energy	Catawba 1 & 2, McGuire 1 & 2 (W)	✓	
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	✓	
Entergy Operations South	Arkansas 2, Waterford 3 (CE)		✓
Exelon Generation Co. LLC	Braidwood 1 & 2, Byron 1 & 2 (W)	✓	
FirstEnergy Nuclear Operating Co	Beaver Valley 1 & 2 (W)	✓	
Florida Power & Light Group	St. Lucie 1 & 2 (CE)		✓
Florida Power & Light Group	Turkey Point 3 & 4, Seabrook (W)	✓	
Nuclear Management Company	Prairie Island 1 & 2, Point Beach 1 & 2, Kewaunee (W)	✓	
Nuclear Management Company	Palisades (CE)		✓
Omaha Public Power District	Fort Calhoun (CE)		✓
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	✓	
Progress Energy	Robinson 2, Shearon Harris (W)	✓	
PSEG – Nuclear	Salem 1 & 2 (W)	✓	
Southern California Edison	SONGS 2 & 3 (CE)		✓
South Carolina Electric & Gas	V.C. Summer (W)	✓	
South Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)	✓	
Southern Nuclear Operating Co.	Farley 1 & 2, Vogtle 1 & 2 (W)	✓	
Tennessee Valley Authority	Sequoyah 1 & 2, Watts Bar (W)	✓	
TXU Power	Comanche Peak 1 & 2 (W)	✓	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	✓	

Note:
* Project participants as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the WOG Program Management Office to verify participation before sending this document to participants not listed above.

Westinghouse Owners Group
International Member Participation* for WOG Project/Task MUHP-3015

Utility Member	Plant Site(s)	Participant	
		Yes	No
British Energy	Sizewell B	✓	
Electrabel (Belgian Utilities)	Doel 1, 2 & 4, Tihange 1 & 3	✓	
Kansai Electric Co., LTD	Mihama 1, Ohi 1 & 2, Takahama 1	✓	
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 Yonggwang 1 & 2 (W)	✓	
Korea Hydro & Nuclear Power Corp.	Yonggwang 3, 4, 5 & 6 Ulchin 3, 4 & 5 (CE)		✓
Nuklearna Electrarna KRSKO	Krsko	✓	
Nordostschweizerische Kraftwerke AG (NOK)	Beznau 1 & 2 (W)	✓	
Ringhals AB	Ringhals 2, 3 & 4	✓	
Spanish Utilities	Asco 1 & 2, Vandellos 2, Almaraz 1 & 2	✓	
Taiwan Power Co.	Maanshan 1 & 2	✓	
Electricite de France	54 Units	✓	

Note:

* This is a list of participants in this project as of the date the final deliverable was completed. On occasion, additional members will join a project. Please contact the WOG Program Management Office to verify participation before sending documents to participants not listed above.

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LIST OF ACRONYMS

AFW	Auxiliary Feedwater
CCP	Centrifugal Charging Pump
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDP	Core Damage Probability
CI	Containment Isolation
CREATCS	Control Room Emergency Air Temperature Control System
CREFS	Control Room Emergency Filtration System
CRMP	Configuration Risk Management Program
CS	Containment Spray
CVCS	Chemical and Volume Control System
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generators
EFW	Emergency Feedwater
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FBACS	Fuel Building Air Cleanup System
FW	Feedwater
HELB	High Energy Line Break
HEPA	High Efficiency Particulate Air
ICS	Iodine Cleanup System
IE	Initiating Event
LCO	Limiting Condition for Operation
LERF	Large Early Release Frequency
LERP	Large Early Release Probability
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
MD	Motor-driven
MFW	Main Feedwater
NSSS	Nuclear Steam Supply System
PIV	Pressure Isolation Valve
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PREACS	Pump Room Exhaust Air Cleanup System (or Penetration Room Exhaust Air Cleanup System)
QS	Quench Spray
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RS	Recirculation Spray
RWST	Refueling Water Storage Tank

LIST OF ACRONYMS (cont.)

SBACS	Shield Building Air Cleanup System
SBO	Station Blackout
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SSPS	Solid State Protection System
SW	Service Water
SWS	Service Water System
TD	Turbine-driven
UHS	Ultimate Heat Sink
WOG	Westinghouse Owners Group

EXECUTIVE SUMMARY

The Westinghouse Owners Group (WOG) has undertaken a risk-informed program to evaluate the endstates that the Technical Specification Actions require the unit to be placed in if the Required Action and associated Completion Time are not met. The Technical Specifications contained in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3, were reviewed to determine the Actions for which changes to the endstates are proposed. The endstates are currently defined based on transitioning the unit to a Mode or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. Mode 5 is the current endstate for LCOs that are applicable in Modes 1 through 4.

The risk of the transition from Mode 1 to Mode 4 or Mode 5 depends on the equipment that is operable. For example, the transition from Mode 4 to Mode 5 can introduce additional risk since it is required to re-align the unit from steam generator cooling to residual heat removal, or shutdown, cooling. During this re-alignment, there is an increased potential for loss of shutdown cooling and loss of inventory events. In addition, decay heat removal following a loss of offsite power event in Mode 5 is dependent on AC power for shutdown cooling, whereas, in Mode 4 the turbine-driven auxiliary feedwater pump will be available. Therefore, transitioning to Mode 5 may not always be the appropriate endstate from a risk perspective.

The purpose of this program is to evaluate and identify the appropriate endstate for a number of Technical Specification Required Actions based on the risk of transitioning the unit from Mode 1 to the lower Modes. Mode 4 is justified as an acceptable alternate endstate to Mode 5.

The proposed changes to the Technical Specification endstates will also reduce the amount of time a unit is shutdown to restore inoperable equipment. The time reduction comes from not requiring a cooldown to Mode 5 and subsequent heat up from Mode 5.

A risk-informed approach, consistent with Regulatory Guides 1.174 and 1.177 (References 1 and 2, respectively) is used in this evaluation. The risk associated with the transition from Mode 1 to Modes 4 and 5, and then returning to Mode 1 operation, is assessed both qualitatively and quantitatively. In addition to assessing the risk impact, the impacts on defense-in-depth and safety margins are also considered.

The Required Actions in the Technical Specifications listed in Table 2-1 are been evaluated for a change in endstate from Mode 5 to Mode 4. The general qualitative evaluation of the plant operating states concludes that there are advantages in risk and in defense-in-depth when the unit remains in Mode 4 rather than continuing to cool down to Mode 5. Technical Specification-specific evaluations demonstrate that there is less risk for the unit if the endstates for these Technical Specifications are changed from Mode 5 to Mode 4. The probabilistic risk assessment model used for the quantitative evaluations is described and shown to be representative of all Westinghouse NSSS units. The results of sensitivity cases support the conclusion that there is less risk associated with a cooldown to Mode 4 than there is for a cooldown to Mode 5. These conclusions are also supported by the Tier 2 assessment and the qualitative assessment of external events.

The evaluations presented and their associated conclusions support changing the endstate from Mode 5 to Mode 4 for the Technical Specifications listed in Table 2-1. Markups of the changes to the NUREG-1431 Technical Specifications and Bases are presented in Appendix B.

1 INTRODUCTION

The Westinghouse Owners Group (WOG) has undertaken a risk-informed program to evaluate the endstates that the Technical Specification Actions require the unit to be placed in if the Required Action and associated Completion Time are not met. The Technical Specifications contained in NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3, were reviewed to determine the Actions for which changes to the endstates are proposed. The endstates are currently defined based on transitioning the unit to a Mode or condition in which the Technical Specification Limiting Condition for Operation (LCO) is not applicable. Mode 5 is the current endstate for LCOs that are applicable in Modes 1 through 4.

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A risk-informed approach, consistent with Regulatory Guides 1.174 and 1.177 (References 1 and 2, respectively) is used in this evaluation. The risk associated with the transition from Mode 1 to Modes 4 and 5, and then returning to Mode 1 operation, is assessed both qualitatively and quantitatively. In addition to assessing the risk impact, the impacts on defense-in-depth and safety margins are also considered.

2 TECHNICAL SPECIFICATIONS AND CHANGE REQUEST

The Technical Specification Required Action endstates evaluated for the endstate change are contained in NUREG-1431, "Standard Technical Specifications for Westinghouse Plants" (Reference 3). Technical Specification number, title, Condition, current endstate, and the proposed endstate are provided in Table 2-1.

Table 2-1 Proposed Changes to Endstates

Technical Specification/ Condition	Title	Current Endstate	Proposed Endstate
3.3.2-B	ESFAS Instrumentation	5	4
3.3.2-C	ESFAS Instrumentation	5	4
3.3.2-K	ESFAS Instrumentation	5	4
3.3.7-C	Control Room Emergency Filtration System Actuation Instrumentation	5	4
3.3.8-D	Fuel Building Air Cleanup System Actuation Instrumentation	5	4
3.4.13-B	RCS Operational Leakage	5	4
3.4.14-B	RCS Pressure Isolation Valve Leakage	5	4
3.4.15-F	RCS Leakage Detection Instrumentation	5	4
3.5.3-C	Emergency Core Cooling System - Shutdown	5	4
3.5.4-C	Refueling Water Storage Tank	5	4
~~3.6.1-B~~	~~Containment (Atmospheric, Subatmospheric, Ice Condenser, and Dual)~~	~~5~~	~~4~~
~~3.6.2-D~~	~~Containment Air Locks~~	~~5~~	~~4~~
~~3.6.3-F~~	~~Containment Isolation Valves~~	~~5~~	~~4~~
~~3.6.4A-B~~	~~Containment Pressure (Atmospheric, Dual and Ice Condenser)~~	~~5~~	~~4~~
~~3.6.4B-B~~	~~Containment Pressure (Subatmospheric)~~	~~5~~	~~4~~
~~3.6.5A-B~~	~~Containment Air Temperature (Atmospheric and Dual)~~	~~5~~	~~4~~
~~3.6.5B-B~~	~~Containment Air Temperature (Ice Condenser)~~	~~5~~	~~4~~
~~3.6.5C-B~~	~~Containment Air Temperature (Subatmospheric)~~	~~5~~	~~4~~
3.6.6A-B	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6A-E	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4
3.6.6B-F	Containment Spray and Cooling Systems (Atmospheric and Dual)	5	4

Table 2-1 (cont.) **Proposed Changes to Endstates**

Technical Specification/ Condition	Title	Current Endstate	Proposed Endstate
3.6.6C-B	Containment Spray System (Ice Condenser)	5	4
3.6.6D-B	Quench Spray System (Subatmospheric)	5	4
3.6.6E-F	Recirculation Spray System (subatmospheric)	5	4
3.6.7-B	Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)	5	4
~~3.6.8-B~~	~~Shield Building (Dual and Ice Condenser)~~	~~5~~	~~4~~
3.6.11-B	Iodine Cleanup System (Atmospheric and Subatmospheric)	5	4
3.6.12-B	Vacuum Relief Valves (Atmospheric and Ice Condenser)	5	4
3.6.13-B	Shield Building Air Cleanup System (Dual and Ice Condenser)	5	4
3.6.14-B	Air Return System (Ice Condenser)	5	4
~~3.6.15-B~~	~~Ice Bed (Ice Condenser)~~	~~5~~	~~4~~
~~3.6.16-D~~	~~Ice Condenser Doors (Ice Condenser)~~	~~5~~	~~4~~
~~3.6.17-C~~	~~Divided Barrier Integrity (Ice Condenser)~~	~~5~~	~~4~~
3.6.18-C	Containment Recirculation Drains (Ice Condenser)	5	4
3.7.7-B	Component Cooling Water System	5	4
3.7.8-B	Service Water System	5	4
3.7.9-C	Ultimate Heat Sink	5	4
3.7.10-C	Control Room Emergency Filtration System	5	4
3.7.11-B	Control Room Emergency Air Temperature Control System	5	4
3.7.12-C	ECCS Pump Room Exhaust Air Cleanup System	5	4
3.7.13-C	Fuel Building Air Cleanup System	5	4
3.7.14-C	Penetration Room Exhaust Air Cleanup System	5	4
3.8.1-G	AC Sources – Operating	5	4
3.8.4-D	DC Sources – Operating	5	4
3.8.7-B	Inverters – Operating	5	4
3.8.9-D	Distribution Systems – Operating	5	4

3 NEED FOR TECHNICAL SPECIFICATION CHANGE

As discussed in Regulatory Guide 1.177 acceptable reasons for requesting Technical Specification changes fall into one or more of the following categories:

- Improvement to Operational Safety: A change to the Technical Specifications can be made due to reductions in the plant risk or a reduction in the occupational exposure of plant personnel in complying with the Technical Specification requirements.

- Consistency with Risk Basis in Regulatory Requirements: Technical Specification requirements can be changed to reflect improved design features in a plant or to reflect equipment reliability improvements that make a previous requirement unnecessarily stringent or ineffective. Technical Specifications may be changed to establish consistently based requirements across the industry or across an industry group.

- Reduce Unnecessary Burdens: The change may be requested to reduce unnecessary burdens in complying with current Technical Specification requirements, based on operating history of the plant or industry in general. This includes extending completion times 1) that are too short to complete repairs when components fail with the plant at-power, 2) to complete additional maintenance activities at-power to reduce plant down time, and 3) to provide increased flexibility to plant operators.

The benefits of changing the Technical Specification Required Action endstates are related primarily to the first two categories.

With regard to operational safety, the risk of the transition from Mode 1 to Mode 4 is lower than the risk of the transition from Mode 1 to Mode 5. The additional mode transition (Mode 4 to Mode 5) involves re-aligning the unit from steam generator cooling to shutdown cooling (residual heat removal (RHR)). This activity requires system alignment changes that can lead to loss of inventory events and loss of shutdown cooling. In addition, in Mode 4, as opposed to Mode 5, additional systems are available for event mitigation that provide a reduced risk once the unit has transitioned to the required endstate. As an example, for a loss of offsite power/station blackout (LOSP/SBO) event, the turbine driven auxiliary feedwater pumps will be available for decay heat removal in Mode 4. In Mode 5, this capability is not available.

Changing the required endstate will also result in increasing unit availability by decreasing the time shutdown. The additional time required to transition to Mode 5 from Mode 4 when shutting down and also to Mode 4 from Mode 5 when restarting can be eliminated with the endstate change. As noted in Section 6.3.1.2, a typical ~~total~~ time for the transition from Mode 4 to Mode 5 during shutdown and from Mode 5 to 4 during startup is ~~70~~ 24 hours. This change will allow a ~~total~~ time reduction of ~~70~~ 24 hours.

4 DESIGN BASES REQUIREMENTS AND IMPACT

The requested change to the Technical Specification Required Action endstate does not impact the design basis for any unit. As discussed in Sections 5.1 and 5.2 defense-in-depth and safety margins will not be impacted. The FSAR accident analyses, events considered and input assumptions will not be changed.

The current Required Action for the unit to be in Mode 5 in 36 hours if inoperable equipment is not restored to operable status, or the Required Actions and associated Completion Times are not met, is being revised to be in Mode 4 in 12 hours.

5 IMPACT ON DEFENSE-IN-DEPTH AND SAFETY MARGINS

In addition to discussing the impact of the changes on plant risk, as presented in Section 6, the traditional engineering considerations also need to be addressed. These include defense-in-depth and safety margins. The fundamental safety principles on which the plant design is based cannot be compromised. Design basis accidents are used to develop the plant design. These are a combination of postulated challenges and failure events that are used in the plant design to demonstrate a safe plant response. Defense-in-depth and adequate safety margins may be impacted by the proposed change and consideration needs to be given to these elements.

5.1 DEFENSE-IN-DEPTH

The proposed change needs to meet the defense-in-depth principle that consists of a number of elements. These elements follow:

- A reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is preserved.
- Over-reliance on programmatic activities to compensate for weaknesses in plant design.
- System redundancy, independence, and diversity are maintained commensurate with the expected frequency and consequences of challenges to the system.
- Defenses against potential common cause failures are maintained and the potential for introduction of new common cause failure mechanisms is assessed.
- Independence of barriers is not degraded.
- Defenses against human errors are maintained.
- The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained.

Operation in Mode 4 offers additional system availability over Mode 5. The additional systems, that offer increased defense-in-depth, are available to mitigate events that can occur. For example:

- <u>Core Cooling</u>: In Mode 4, core cooling can be maintained by the SGs in conjunction with AFW, and the RHR system can provide a backup function. In Mode 5, core cooling is dependent on the RHR system and AFW is not required to be operable.
- <u>Inventory Makeup</u>: In Mode 4, one train of ECCS is available to mitigate loss of coolant events. The ECCS is not required to be operable in Mode 5. Mitigation of loss of coolant events is dependent on the availability of a high head injection pump/train.
- <u>Electrical Power Sources</u>: In Mode 4, all emergency DGs are operable, while in Mode 5 not all are required to be operable, depending on the other equipment required to be operable.

In addition, the endstate changes eliminate the need to transition from Mode 4 to Mode 5. This transition requires re-alignment of systems to transfer to shutdown cooling, which is cooling provided by the RHR system for the Westinghouse NSSS plants. This transition can lead to an increased probability of loss of shutdown cooling or loss of inventory events. Remaining in Mode 4 eliminates the need to make this transition.

Therefore, the system redundancy, dependence, and diversity are increased by remaining in Mode 4. The reasonable balance among prevention of core damage, prevention of containment failure, and consequence mitigation is not impacted since there is no change in plant configuration, design, or operation in the Modes of interest. In addition, the other elements of defense-in-depth are not impacted by this change since the plant configuration (systems, structures, and components) and operation within each Mode remains the same.

The impact on defense-in-depth is further discussed in each individual requested endstate change in Section 6.

5.2 IMPACT ON SAFETY MARGINS

The safety analysis acceptance criteria in the licensing basis is not impacted by this change. The codes and standards approved for use by the NRC continue to be met. Availability of redundant and diverse systems will be maintained or improved with the proposed endstate changes. The proposed changes will not allow plant operation in a configuration outside the design basis. For the proposed changes, Mode 4 offers additional defense-in-depth and a lower risk level, for the requested changes, than Mode 5. There is no impact on safety margins.

6 ASSESSMENT OF THE IMPACT OF THE ENDSTATE CHANGE

This section provides the discussion of the impact of each proposed endstate change on defense-in-depth, safety margins, and risk. The risk impact is discussed qualitatively and quantitatively. The quantitative risk discussion is primarily directed at the impact on core damage frequency. The impact with respect to large early release frequency is provided qualitatively. Also included in this section is a discussion of the transition risk model used in the quantitative evaluations.

The evaluation was completed consistent with the requirements in Regulatory Guides 1.174 and 1.177.

6.1 APPROACH TO THE EVALUATION

The evaluation supporting these Technical Specification changes is divided into three parts: defense-in-depth, safety margins, and risk assessment. The impact on defense-in-depth is discussed, in general terms, in Section 5.1. The impact on safety margins is discussed, in general terms, in Section 5.2. The specific impact of each proposed change is discussed, with regard to defense-in-depth, in Section 6.4.

The risk assessment supporting these changes is divided into two parts. The first part discusses the impact of the changes qualitatively and the second assesses the impact from the quantitative perspective. The qualitative and quantitative risk impacts are discussed in general terms in Sections 6.2 and 6.3. The specific impact of each proposed change is discussed in Section 6.4.

The Technical Specification change being considered is the Technical Specification Required Action endstate. A change from a Mode 5 endstate to a Mode 4 endstate is justified for a number of Technical Specifications. Each proposed endstate change listed in Table 2-1 is evaluated and discussed separately in Section 6.4.

The analysis approach identifies the appropriate endstate from a risk perspective, that is, which endstate is associated with the lowest risk. The evaluation considers the risk associated with:

- The transition from operating in Mode 1 at 100% power to Mode 4 or Mode 5 with the inoperable equipment assumed to be unavailable.
- Operation in the endstate of interest with the inoperable equipment assumed to be unavailable.
- The transition from Mode 4 or Mode 5 to Mode 1 at 100% power with all equipment operable.

To perform the quantitative evaluation a transition risk model is used. This model has the capability to evaluate the risk of changing plant operating modes during plant shutdowns and startups. Transitions between modes is specifically modeled. The risk analysis considers more than design basis events. The model is discussed in Section 6.3. Core damage probability (CDP) is the risk metric used in this model. Large early release probability (LERP) is not addressed quantitatively. A qualitative evaluation is performed for LERP.

6.2 QUALITATIVE RISK ASSESSMENT

The qualitative risk assessment is directed at the risk associated with plant operation in Mode 4 and Mode 5, and transitioning to and from these modes. Consistent with the Technical Specification mode definitions, this evaluation assumes that the unit has entered the mode of interest when the conditions associated with the upper end of mode are met. The following sections provide a qualitative assessment of the risk associated with transitioning a plant to, from, and operating in Modes 4 and 5. All equipment is assumed to be available, unless operating procedures direct that equipment be isolated or locked-out. In Section 6.4, the impact of inoperable equipment is addressed.

6.2.1 Mode Transition Risk Assessment

The following provides the general steps to shutting the plant down (to Mode 5). Plants that are shutting down as required by the Technical Specification or for a refueling or other forced outage will generally be starting at or near 100% power (Mode 1).

- The power reduction will start and eventually the feedwater supply will be switched from the main feedwater system to the auxiliary or startup feedwater system, and the turbine will be tripped. These plant configuration changes provide for an increased probability of loss of feedwater flow to the steam generators. After transitioning to AFW or the startup feedwater system and reducing the power level to less than 5%, the plant will be in Mode 2.

- The plant power level will continue to be reduced to 0% with AFW used to remove decay heat. The reactor coolant system (RCS) temperature remains at ~557°F, or 547°F depending on the no load T_{avg}, and the RCS pressure is at ~2235 psig with the secondary side at normal operating pressure. Mode 3 (hot standby) is achieved when the power level is 0%.

- The next step is to begin RCS cooldown. The shutdown margin (boration) is established and the reactor trip breakers are opened. The number of operating reactor coolant pumps (RCPs) may be reduced. Cooldown is initiated via the steam dump system or SG atmospheric relief valves. The transition to Mode 4 (hot shutdown) occurs when the RCS temperature is less than 350°F.

- The cooldown continues and the safety injection pumps are disabled. RCS cooling can be switched to the RHR system when the RCS temperature is less than 340°F and the RCS pressure is less than 365 psig. The number of operating RCPs can be reduced to one. The switch to Mode 5 (cold shutdown) occurs when the RCS temperature is less than 200°F.

To return to power, similar steps are required in the reverse order. Table 6-1 provides a summary of the important RCS parameters for the different mode transitions. This table also provides the Technical Specification temperatures and power levels specified for the different Modes.

Table 6-1 Key Plant Parameters by Technical Specification Mode

Parameter	Mode 5 to Mode 4	Mode 4 to Mode 3	Mode 3 to Mode 2	Mode 2 to Mode 1	Mode 1
Technical Specification RCS Average Temperature	≤200°F (Mode 5, Cold Shutdown)	>200°F to <350°F (Mode 4, Hot Shutdown)	≥350°F (Mode 3, Hot Standby)	NA (Mode 2, Startup)	NA (Mode 1, Power)
Technical Specification Reactor Power Level	NA (Mode 5)	NA (Mode 4)	NA (Mode 3)	≤5% (Mode 2)	>5% (Mode 1)
RCS Average Temperature	~185°F to ~330°F	~330°F to ~557°F	~557°F	~557°F	~557°F
RCS Pressure	~340 psig	~340 psig to ~2235 psig	~2235 psig	~2235 psig	~2235 psig
Pressurizer Status	Water solid to bubble	Bubble	Bubble	Bubble	Bubble
Secondary Side Pressure	0 psig	Less than Normal operating pressure	Normal operating pressure	Normal operating pressure	Normal operating pressure

The following discussion centers around several plant operating states (POS). The POS approach is used as the basis for the qualitative discussion to be consistent with the quantitative transition PRA model, that is discussed in Section 6.3.1. A POS is a unique plant configuration defined by a set of parameters. For each POS, a unique set of initiating events, plant conditions, and systems available for event mitigation can be identified. Very often a plant will be in a POS for a relatively short period of time, because switching the plant configuration (system re-alignments) is required to reach the desired endstate. To identify the POSs, an understanding of the typical key activities that are in progress as the plant shuts down and restarts is required. The following provides a general summary of these activities:

Modes 1-2

- Decrease power (Mode 1)
- Take the turbine off-line (Mode 1)
- Transfer from MFW (main feedwater) to AFW (note that some plants may continue on MFW depending on their MFW design and approach to plant shutdown) (Mode 1)
- Mode 2 (startup) when power level is ≤ 5%

Modes 2-3

- Reduce power to 0% (Mode 2 to Mode 3)
- Insert control banks (Mode 2)
- Mode 3 (hot standby) when power level is 0% (T_{ave} = 557°F, pressurizer pressure = 2235 psig, pressurizer level = 25% to 40%, SG levels = 65%)

Modes 3-4 (upper part of Mode 4 on AFW, then transfer to RHR)

- Borate the RCS to the required boron concentration to satisfy the shutdown margin requirement (Mode 3)
- If running and not required to feed the SGs, stop the turbine-driven AFW pump (Mode 3)
- If running, shut down the MFW pumps (Mode 3)
- Maintain SG water level (Mode 3)
- If withdrawn, insert the shutdown banks (Mode 3)
- Open the reactor trip breakers (Mode 3)
- Reduce the number of operating RCPs (Mode 3)
- At approximately 2000 psig, block safety injection and steamline isolation (Mode 3)
- At approximately 950 psig, isolate the accumulators (Mode 3)
- If the secondary side is to be cooled down, start "Secondary Plant Shutdown" (Mode 3)
- Place cold overpressure protection in service prior to reaching 350°F (or plant-specific temperature) (Mode 3)
- RCS cooldown is controlled by the condenser steam dump valve and SG atmospheric relief valves
- Decrease the RCS temperature from ~557°F to ~330°F (transition to Mode 4 occurs when the RCS temperature drops below 350°F)
- Decrease the RCS pressure from ~2235 psig to ~340 psig
- When the RCS pressure is less than 365 psig and the RCS temperature is less than 340°F, place one RHR loop in operation (Mode 4)

Modes 4-5 (on RHR system)

- Disable the safety injection pumps (Mode 4)
- Defeat the AFW actuation system (Mode 4)
- Reduce the operating RCPs to one (Mode 4)
- Cooldown the RCS using the RHR system
- Maintain the pressurizer pressure at 250 psig
- At 200°F, Mode 5 (Cold Shutdown) is entered
- Continue to cooldown to 190°F to 170°F
- End state: A bubble in the pressurizer, the RCS temperature is between 190°F and 170°F, the RCS pressure is 250 psig
- Centrifugal charging pumps (CCP) are in standby (Mode 5) (one CCP operating if a RCP is running)
- Solid state protection system (SSPS) is in service only for certain functions depending on whether the control rods are capable of withdrawing (Mode 5)

Modes 5-4 (on RHR system)

- CCPs are in standby (Mode 5) (one CCP operating if a RCP is running)
- SSPS is in service (Mode 5)
- Increase the RCS temperature from ~185°F to ~330°F (transition to Mode 4 occurs when RCS temperature exceeds 200°F)
- Verify that AFW is aligned for startup (Mode 4)
- Maintain the RCS pressure at ~340 psig
- RCS cooling by RHR (Mode 5 and lower end of Mode 4)
- Cold overpressure protection in service (Mode 5 and lower end of Mode 4)

Modes 4-3 (lower end of Mode 4 on RHR, then transfer to AFW)

- Prepare the SG for startup (Mode 4)
- AFW actuation signals and AFW components are available for automatic actuation (Mode 4)
- Place the RHR system in standby (lower end of Mode 4)
- Block the cold overpressure protection system (Mode 4)
- Initiate AFW (note that at some plants, a startup feedwater pump or condensate pumps and MFW may be used for startup instead of AFW) (Mode 4)
- Increase the RCS temperature from ~330°F to ~557°F (transition to Mode 3 occurs when the RCS temperature is ≥ 350°F)
- Increase the RCS pressure from ~340 psig to ~2235 psig
- Start the remaining RCPs (Mode 3)
- Verify pressurizer (PZR) pressure safety injection (SI) and steamline pressure SI and steamline isolation (SLI) auto reset (Mode 3)
- RCS heatup is controlled by the condenser steam dump valves and the SG atmospheric relief valves

Modes 3-2

- Close the reactor trip breakers (Mode 3)
- Withdraw shutdown and control banks (Mode 3)
- Raise power to less than 5% (Mode 3 to Mode 2)

Modes 2-1

- Transfer from AFW to MFW (note that some plants may already be on MFW depending on their MFW design and approach to plant startup) (Mode 1)
- Increase power (Mode 1)
- Bring turbine on-line (Mode 1)

Table 6-2 provides a summary of system status by the Technical Specification mode. Table 6-3 provides the target plant conditions when shutting down as required by the Technical Specifications. The target conditions of interest are for Modes 4 and 5. As noted previously, it is assumed that the plant will stop the shutdown after entering the mode required by the Technical Specifications.

Each POS is defined based on the plant state, available equipment, and potential initiating events. Table 6-4 defines the POSs for the transition from power operation to cold shutdown (Mode 5). Table 6-5 defines the POSs for the transition from cold shutdown (Mode 5) to power operation. These plant operating states are based on the previously discussed information and each POS represents a unique plant configuration that is defined by the plant conditions and parameters provided on these tables.

Table 6-2 System Status by Technical Specification Mode

System	Mode 6	Mode 5	Mode 4	Mode 3	Mode 2	Mode 1
RCS Charging and Letdown[1]	Establish function	In service	In service	In service	In service	In service
Reactor Coolant Pumps	None running	As needed for plant heatup	As needed for plant heatup	All RCPs running	All RCPs running	All RCPs running
Reactor Trip Breakers	Open	Open	Open	Open/Closed	Closed	Closed
Residual Heat Removal	In service	In service	In service or in standby	Standby	Standby	Standby
Auxiliary Feedwater	~~Out of service~~ Not required	~~Out of service~~ Not required	Aligned for startup or in service	In service	In service	In service and then standby after switch to MFW
High Head Injection[1]	Pull to lock	Pull to lock when water solid, standby with bubble	Standby	Standby	Standby	Standby
Cold Overpressure Protection	Establish function	In service	In service[2]	Not required	Not required	Not required
High Flux at Shutdown Alarm (HFASA)	In service	In service	In service	In service	Not required	Not required
Source Range	Two channels in service	Two channels in service	Two channels in service	Two channels in service	Two channels in service below P-6	Not required
Intermediate Range	Not required	Not required	Not required	Not required	Two channels in service	Two channels in service below P-10
Power Range	Not required	Not required	Not required	Not required	Required	Required
Solid State Protection System	Not required	Not required	In service	In service	In service	In service
Emergency Diesel Generators	Less than full complement[3]	Less than full complement[3]	Full complement	Full complement	Full complement	Full complement

Notes:

1. One charging pump is operating to provide RCS charging in Modes 1-6.
2. Cold overpressurization is required in the lower part of Mode 4.
3. Depending on equipment required to be operable.

Table 6-3 Important Parameters for Mode Target Conditions

Parameter	Mode 1	Mode 2	Mode 3	Mode 4	Mode 5
RCS Temperature	557°F	557°F	557°F	~330°F	170°F to 190°F
RCS Pressure	2235 psig	2235 psig	2235 psig	340 psig	250 psig
Secondary Side Status	Normal operating pressure	Normal operating pressure	Normal operating pressure	Normal operating pressure	Low pressure
PZR Status	Bubble	Bubble	Bubble	Bubble	Bubble
(Decay) Heat Removal Mode	MFW	AFW	AFW	AFW	RHR
Power Level	100%	5%	0%	0%	0%

Less than

Table 6-4 Plant Operating States (Power Operation to Cold Shutdown)				
State	**POS 1**	**POS 2**	**POS 3**	**POS 4**
Plant Mode	1 (transition only) 2 3 (upper part)	3 (middle part)	3 (lower part) 4 (upper part)	4 (lower part) 5 (upper part)
RCS Temperature	557°F	557°F to XX°F[1]	XX°F[1] to 340°F	340°F to 180°F
RCS Pressure	2235 psig	2235 psig to 950 psig	950 psig to 365 psig	365 psig to 250 psig
Pressurizer	Bubble	Bubble Less than	Bubble	Bubble
Secondary Side	Normal operating pressure	Normal operating pressure	Normal operating pressure	Low pressure (shutdown)
Activities	• AFW for decay heat removal • Reduce power • Switch from MFW to AFW • Borate • Insert control rods • Take turbine off-line	• AFW for decay heat removal and cooldown • Open trip breakers • Reduced operating RCPs • Block SI and SLI • RCS cooldown	• AFW for decay heat removal and cooldown • Reduced operating RCPs • Isolate accumulators • RCS cooldown • Start secondary side cooldown	• RHR for decay heat removal and cooldown • Switch to RHR cooling • Disable SI pumps • Defeat AFW start signals • Cold overpressure protection (COP) in service
System Status	• AFW operating • All systems available	• AFW operating • All systems available • SI and SLI signals blocked • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Accumulators isolated • Reactor trip breakers open	• RHR operating • SI, SLI, and AFW signals blocked • Accumulators isolated • SI pumps disabled • COP in service • Reactor trip breakers open

Note:

1. A defined temperature is not important to this analysis.

Table 6-5 Plant Operating States (Cold Shutdown to Power Operation)

State	POS 4	POS 5	POS 6	POS 7
Plant Mode	4 (lower part) 5 (upper part)	3 (lower part) 4 (upper part)	3 (middle part)	1 (transition only) 2 3 (upper part)
RCS Temperature	180°F to 340°F	340°F to XX°F[1]	XX°F[1] to 557°F	557°F
RCS Pressure	250 psig to 365 psig	365 psig to 950 psig	950 psig to 2235 psig	2235 psig
Pressurizer	Bubble	Bubble Less than	Bubble	Bubble
Secondary Side	Low pressure (shutdown)	Normal operating pressure	Normal operating pressure	Normal operating pressure
Activities	• RHR for decay heat removal • Switch to AFW cooling • Establish AFW actuation signals • RCS heatup	• AFW for decay heat removal • One RCP running[2] • RCS heatup • Start secondary side heatup	• AFW for decay heat removal • One RCP running[2] • RCS heatup • Establish SI and SLI signals • Un-isolate accumulators	• AFW for decay heat removal • Switch from AFW to MFW • Withdraw shutdown and control rods • Bring turbine on-line • Close trip breakers • All RCPs running • Increase power
System Status	• RHR operating • SI, SLI, and AFW signals blocked • Accumulators isolated • SI pumps disabled • COP in service • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Accumulators isolated • Reactor trip breakers open	• AFW operating • All systems available • SI and SLI signals blocked • Reactor trip breakers open	• AFW to MFW • All systems available

Notes:

1. A defined temperature is not important to this analysis.
2. If the rods are not capable of withdrawal.

Table 6-6 lists the possible internal initiating events for each POS. The following discusses this information. These notes are also provided with Table 6-6.

1. RCS pressure is much lower in POS 3, 4, and 5 than in POS 1, 2, 6, and 7. Large and medium LOCAs are considered in POS 3 and 5, but at a reduced frequency. Large and medium LOCAs are not considered in POS 4.

2. Small LOCAs in POS 1, 2, 6, and 7 are due to random pipe breaks and random RCP seal failures. Small LOCAs in POS 3 and 5 are significantly reduced due to reduced RCS pressure and temperature. Small consequential LOCAs can also occur in POS 1 and 7 due to transient events that lead to the opening of PZR PORVs with failure of the PORVs to reseat. These are not considered in any other POS due to the low plant power level. In POS 4, LOCAs are due to alignment issues and open valves, not pipe breaks or random failures of RCP seals, and are referred to as loss of inventory events.

3. SGTRs are considered in POS 3 and 5, assuming the at-power frequency based on NUREG/CR-6144 (Reference 4), Section 4.7. SGTRs are not considered in POS 4 because the pressure difference across the tubes is much lower and the steam generators are not being used for RCS cooling. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.

4. Secondary side breaks are not considered in POS 4 since the secondary side pressure is much lower than in POS 1, 2, 3, 5, 6, and 7.

5. RCP seal LOCAs due to loss of seal cooling are not considered in POS 3, 4, and 5 since the RCS pressure and temperature are much lower than in POS 1, 2, 6, and 7. In addition, for POS 3, 4, and 5, there is a minimum of 50°F subcooling, which means that a RCP seal pop-open event is not an issue. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.

6. Rod withdrawal is not considered in POS 2, 3, 4, 5, and 6 since the reactor trip breakers are open.

7. Loss of decay heat removal is due to a loss of AFW in POS 1, 2, 3, 5, 6, and 7 and a loss of RHR in POS 4.

8. With regard to loss of feedwater control, there is an increased probability for loss of feedwater due to feedwater control problems related to the switch from MFW to AFW during the transition down in power and related to the switch from AFW to MFW during the transition up in power. This is only applicable in POS 1 and 7.

Table 6-6 Initiating Events by Plant Operating State

Initiating Event	POS 1	POS 2	POS 3	POS 4	POS 5	POS 6	POS 7
Large LOCA[1]	X	X	X		X	X	X
Medium LOCA[1]	X	X	X		X	X	X
Small LOCA[2]	X	X	X		X	X	X
Loss of Inventory[2]				X			
RCP Seal LOCAs (Loss of Seal Cooling)[5]	X	X				X	X
Loss of Feedwater Control[8]	X						X
Loss of Decay Heat Removal[7]	X	X	X	X	X	X	X
Loss of Offsite Power	X	X	X	X	X	X	X
Cold Overpressurization				X			
SG Tube Rupture[3]	X	X	X		X	X	X
Secondary Side Breaks[4]	X	X	X		X	X	X
ATWS							
Boron Dilution	X			X			X
Rod Withdrawal[6]	X						X

Notes:

1. RCS pressure is much lower in POS 3, 4, and 5 than in POS 1, 2, 6, and 7. Large and medium LOCAs are considered in POS 3 and 5, but at a reduced frequency. Large and medium LOCAs are not considered in POS 4.
2. Small LOCAs in POS 1, 2, 6, and 7 are due to random pipe breaks and random RCP seal failures. Small LOCAs in POS 3 and 5 are significantly reduced due to reduced RCS pressure and temperature. Small consequential LOCAs can also occur in POS 1 and 7 due to transient events that lead to the opening of PZR PORVs with failure of the PORVs to reseat. These are not considered in any other POS due to the low plant power level. In POS 4, LOCAs are due to alignment issues and open valves, not pipe breaks or random failures of RCP seals, and are referred to as loss of inventory events.
3. SGTRs are considered in POS 1, 2, 3, 5, 6, and 7. Even though in POS 3 and POS 5 the delta P across the tubes (P_{RCS} - $P_{secondary\ side}$) is much lower than in POS 1, 2, 6, and 7, the frequency was not reduced. SGTRs are not considered in POS 4 because the pressure difference across the tubes is much lower and the steam generators are not being used for RCS cooling. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.
4. Secondary side breaks are not considered in POS 4 since the secondary side pressure is much lower than in POS 1, 2, 3, 5, 6, and 7.
5. RCP seal LOCAs due to loss of seal cooling are not considered in POS 3, 4, and 5 since the RCS pressure and temperature are much lower than in POS 1, 2, 6, and 7. In addition, for POS 3, 4, and 5, there is a minimum of 50°F subcooling, which means that a RCP seal pop-open event is not an issue. In POS 2 and 6 the RCS pressure is conservatively assumed to be at its higher value (2235 psig) the majority of the time.
6. Rod withdrawal is not considered in POS 2, 3, 4, 5, and 6 since the reactor trip breakers are open.
7. Loss of decay heat removal is due to a loss of AFW in POS 1, 2, 3, 5, 6, and 7 and a loss of RHR in POS 4.
8. With regard to loss of feedwater control, there is an increased probability for loss of feedwater due to feedwater control problems related to the switch from MFW to AFW during the transition down in power and related to the switch from AFW to MFW during the transition up in power. This is only applicable in POS 1 and 7.
9. Boron dilution events have not been found to be important to plant risk, and are excluded from further consideration in POS 2, 3, 5, and 6.

POS 1 and POS 7

This state is defined as Technical Specification Mode 1 (only for transitioning from power operation to Mode 2), Mode 2, and the upper part of Mode 3. In Mode 1, only the additional risk associated with the transition from power operation to Mode 2 is included in POS 1. This additional risk is due to potential transients cause by feedwater control issues when shutting the plant down. Other risks associated with Mode 1 operation are not considered part of the risk of transitioning to lower modes, but are considered part of the at-power risk.

Mode 2 and the upper part of Mode 3 are very similar except for the power level (5% vs. 0%). All other key plant parameters are the same. The same initiating events are applicable in Modes 2 and the upper part of 3. The ATWS event is not considered since the plant is at very low power and the high RCS pressures cannot be reached. Most transients cannot occur due to equipment status, for example, the turbine and generator are not operating. Since the plant will be shutting down from 100% power, decay heat levels will be high.

The key difference in POS 1 and POS 7 is the decay heat level. With the lower decay heat levels when returning to power, due to the time the reactor was shut down, operators have a longer time to respond to events. In addition, feedwater control concerns and issues bringing the turbine on-line can lead to a different risk level when starting up than when shutting down.

In these POSs, all systems are available for event mitigation, that is, none have been removed from service related to the mode changes.

POS 2 and POS 6

This state is defined as the middle of Mode 3. The RCS pressure and temperature are being reduced, but are still relatively high. Signals for safety injection and steamline isolation are blocked and the reactor trip breakers are open. The same events in POS 1 are applicable except for rod withdrawal. Loss of feedwater control is no longer an issue either.

POS 2 and POS 6 are very similar except for the direction of transition. In POS 2 the plant is transitioning down (toward shutdown) and in POS 6 the plant is transitioning up (returning to power). The key difference in POS 2 and POS 6 is the decay heat level. With the lower decay heat levels when returning to power, due to the additional time being shut down, the operators have a longer time to respond to events.

POS 3 and POS 5

This state is defined as the lower part of Mode 3 and the upper part of Mode 4. The RCS pressure and temperature are significantly reduced from power operation, therefore, many of the events associated with the high RCS pressure (LOCAs/pipe breaks) have a reduced frequency. In addition, accumulators are isolated.

POS 3 and POS 5 are very similar except for the direction of transition. In POS 3 the plant is transitioning down (toward shutdown) and in POS 5 the plant is transitioning up (returning to power).

The key difference in POS 3 and POS 5 is the decay heat level. With the lower decay heat levels when returning to power, due to the additional time being shut down, the operators have a longer time to respond to events.

POS 4

This state is defined as the lower part of Mode 4 and the upper part of Mode 5. The transition from AFW cooling to RHR cooling occurs in this POS. The RCS pressure and temperature are significantly reduced from power operation, therefore, the LOCA events and SG tube rupture event are no longer applicable. The secondary side pressure is also reduced eliminating the secondary side break events. Loss of inventory related to the RCS cooling switch from AFW to RHR is an event that is added. This can occur when transitioning down or up. Cold overpressurization is also added.

Key Differences Between POS 1 and POS 2 (and POS 7 and POS 6)

- No switchover from MFW to AFW in POS 2
- Reactor trip breakers are open in POS 2
- SI and SLI signals are blocked in POS 2 (operator actuations required for some events)
- Most initiating events are the same, but rod withdrawal and many transient events (such as, loss of MFW and turbine trip) cannot occur in POS 2

Key Differences Between POS 2 and POS 3 (and POS 6 and POS 5)

- Reduced RCS pressure and temperature results in a reduced event frequency for pipe rupture type LOCAs and RCP seal LOCAs in POS 3
- Accumulators are isolated in POS 3

Key Differences Between POS 3 and POS 4 (and POS 5 and POS 4)

- Reduced RCS pressure and temperature in POS 4
- Reduced secondary side pressure and temperature in POS 4
- Loss of inventory and loss of decay heat removal important in POS 4 due to a transfer from AFW/SG cooling to RHR cooling
- Secondary side breaks cannot occur in POS 4 due to a reduced secondary side pressure
- Cold overpressurization needs to be addressed in POS 4

6.2.2 Comparison of Endstates

To achieve Mode 4 as an endstate, the plant will need to transition through POS 1, POS 2, and into POS 3. To achieve Mode 5 as an endstate, the plant will need to transition through POS 1, POS 2, POS 3, and into POS 4. With either endstate, the plant will need to transition through POS 1, POS 2 and into POS 3. To determine the appropriate endstate (Mode 4 vs. Mode 5), the additional risk for the transition through POS 3 and into POS 4 needs to be considered, as well as the risk of remaining in POS 3 (Mode 4) as opposed to POS 4 (Mode 5).

Several of the key differences between POS 3 and 4 are:

- The frequency of loss of decay heat removal is at an increased level with POS 4 as the endstate due to the system re-alignments required. Loss of decay heat removal events in POS 3 can be addressed with AFW (all pumps available) or the RHR system following depressurization of the RCS. In POS 4, the TD AFW pump will not be available to address similar events. In addition, the automatic AFW start signal is available in POS 3, but not POS 4. Therefore, additional options are available in POS 3 for decay heat removal.

- The frequencies of loss of inventory (LOCA) events can be at an increased level with POS 4 as the endstate due to the system re-alignments required. Loss of inventory events in POS 3 can be addressed with the available train of ECCS. In POS 4 a full train of ECCS is not available. The SI pumps are out of service. Inventory control is dependent on the charging system. Therefore, additional options are available in POS 3 for inventory control.

- Mitigation of loss of offsite power (LOSP)/station blackout (SBO) events in POS 3 can be provided by the AFW system including the turbine-driven pump. Availability of the turbine-driven pump is particularly important in case the event degrades to a station blackout. In POS 4, the AFW system turbine-driven pump will not be available for decay heat removal, and the plant will be dependent on restoring electric power to the RHR system. Again, additional options are available in POS 3 for event mitigation.

- The cold overpressurization event needs to be considered in POS 4, but not in POS 3. Although not a large risk contributor, this event is addressed in POS 4.

- Secondary side breaks are considered in POS 3, but not in POS 4. In POS 3 the secondary side will be less than ~~may be near the~~ normal operating pressure, but in POS 4 this pressure is greatly reduced, reducing the likelihood of a secondary side break. Secondary side breaks are not typically large contributors to risk, therefore, this assumption for POS 4 has a small risk impact.

- Risk in the shutdown modes is very dependent on electric power availability. There are more required independent sources of electrical power in POS 3 than in POS 4 and there are more potential activities in POS 4 that could cause a loss of offsite power.

- In POS 3, there is more redundancy and diversity of mitigating and support systems required to be available than there is in POS 4.

6.2.3 Containment Considerations

In Modes 1-4, the containment, containment isolation valves, containment sprays, and containment cooling systems are required to be operable. The Bases for the Technical Specifications state that in these modes a design basis accident could cause a release of radioactive material to containment, and an increase in containment pressure and temperature that would require the use of these systems for accident mitigation. The Bases also indicate that in Modes 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature in these modes. Additional operability requirements are specified for Mode 6.

The risk analysis considers more than design basis events. Events can occur in Mode 5 that may lead to core damage and may cause releases outside containment. With the reduced RCS pressure and temperature in Mode 5, as opposed to Modes 1-4, the need for containment cooling systems and containment spray is not as important. Mode 4 does require the containment, containment isolation, and containment cooling and spray systems to be operable.

6.2.4 Conclusions from the Qualitative Risk Assessment

Based on Sections 6.2.3 and 6.2.4, there are advantages, from the risk perspective and also the defense-in-depth perspective, to remain in POS 3 (Mode 4) as opposed to POS 4 (Mode 5) for shutdowns required by the Technical Specifications. In POS 3, the initiators are generally at a reduced probability of occurrence compared to power operation and there are additional systems available for event mitigation compared to POS 4 (Mode 5).

6.3 QUANTITATIVE RISK ASSESSMENT

As with the qualitative risk assessment, the quantitative assessment is also directed at the risk associated with plant operation in Mode 4 and Mode 5, and transitions to and from these modes. Evaluating the risk requires an assessment of the likelihood of applicable initiating events. Consistent with the Technical Specification mode definitions, this evaluation assumes that the unit has entered the mode of interest when conditions associated with the upper end of the mode are met. The following sections provide a quantitative assessment of the risk associated with transitioning a plant to, from, and remaining in Modes 4 and 5. All equipment is assumed to be available, unless operating procedures direct equipment to be isolated or locked-out.

6.3.1 Transition Risk Model

A generic transition risk model that is representative of Westinghouse Nuclear Steam Supply System (NSSS) plants is presented in this section. Only high level information of the model is presented as well as the model quantification results. The risk metric to be determined is core damage probability (CDP). A separate risk model, similar to a level 1, internal event, at-power PRA model, is developed for each POS. Only internal initiating events are included in the transition risk model.

This model is based on the information identified in the qualitative risk discussion in Section 6.2. The POSs are defined on Table 6-4 and 6-5, and the associated initiating events that are addressed are provided on Table 6-6. Equipment availability is discussed in Section 6.2.1.

The plant response or event trees for each initiating event in each POS, the time spent in each POS, the initiating event frequencies, mitigation equipment availability, and human error probabilities for operator actions are developed and discussed in the following sections. This is followed by the model quantification that provides the CDP values for each POS, along with the initiating event CDP distribution for POS 3 and POS 4. This quantification assumes all equipment is available except for equipment removed from service, as the plant transitions though the modes, following plant procedures.

The model is based on a single unit site with standard two train systems. Several key characteristics of this model include:

- AFW system – two MD pumps and one TD pump
- Emergency core cooling system (ECCS) – two train system, each train includes a high pressure and low pressure subsystem
- Reactor protection system (RPS) – SSPS
- Service water (SW) system – two train system
- Component cooling water (CCW) system – two train system
- Electrical power system – a two train system with one emergency diesel generator (EDG) per train

INSERT A

~~The transition risk PRA model is based on a single unit, therefore, it does not take credit for the availability of shared systems. The AFW system is similar to many Westinghouse NSSS plants. The AFW system does not include a diesel-driven pump that would provide diverse mitigation for loss of offsite power events.~~

~~The ECCS design is also similar to many Westinghouse NSSS plants. The centrifugal charging pumps are used for high pressure safety injection. High pressure safety injection does not include a set of separate safety injection pumps, but the general success criteria of requiring one train is common for Westinghouse NSSS PRA models. Low pressure injection and recirculation is performed by the RHR pumps and heat exchangers. This is similar to many other Westinghouse NSSS plants, although some plants do have low pressure safety injection pumps separate from the RHR system.~~

~~The reactor protection system for the transition risk PRA model is based on a solid state protection system. While many Westinghouse NSSS plants have relay protection systems, the reliability of the two systems is not significantly different, therefore, the model is applicable to both protection systems.~~

~~The modeled CCW system consists of two trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants.~~

~~The modeled SW system consists of two trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants.~~

~~The electric power system modeled is a two train system with one diesel generator per train. This is a common design among the Westinghouse NSSS plants. Some plants have more redundancy in their design, including shared diesels between units.~~ INSERT A

~~The model chosen includes many safety system features and support system features that are common among many of the Westinghouse NSSS plants. While the plant designs vary for the systems modeled, the model used provides representative results whose conclusions are applicable to all Westinghouse NSSS plants.~~

6.3.1.1 Plant Response Model

The following discusses the response of the plant to the initiating events in each POS. This model is based on a 3-loop Westinghouse NSSS plant at-power PRA model that has undergone the Owners Group Peer Review Process and has been updated in response to Peer Review comments. As discussed in Section 6.2, many of the events that can occur in the upper modes (Modes 2 and 3), or POSs, are mitigated similarly to the events that occur when at-power. Therefore, much of the modeling used in the at-power PRA model is also used in the modeling for events that occur in the transition modes. The events included in the baseline at-power model are:

Loss of Coolant Accidents

- Small LOCA
- Medium LOCA
- Large LOCA
- Interfacing Systems LOCA
- Reactor Vessel Rupture

Secondary Side Breaks

- Inside Containment
- Outside Containment

Transients

- Loss of Main Feedwater
- Partial Loss of Main Feedwater
- Loss of Condenser
- Positive Reactivity Insertion
- Primary System Transient
- Loss of Reactor Coolant Flow
- Reactor Trip
- Inadvertent Safety Injection Signal
- Turbine Trip
- Inadvertent Opening of a Steam Valve

INSERT A

The transition risk PRA model is based on a single unit, therefore, it does not take credit for the availability of shared systems. The AFW system is similar to many Westinghouse NSSS plants. The AFW system does not include a diesel-driven pump that would provide diverse mitigation for loss of offsite power events. Based on the results presented in Table 6-10, the loss of offsite power event is a larger percent contributor to the POS 3 risk than it is to the POS 4 risk and this design difference would be more beneficial for Mode 4 (POS 3).

The ECCS design is also similar to many Westinghouse NSSS plants. The centrifugal charging pumps are used for high pressure safety injection. High pressure safety injection does not include a set of separate safety injection pumps, but the general success criteria of requiring one train is common for Westinghouse NSSS PRA models. Plants with Intermediate Head SI (IHSI) pumps typically require the IHSI pumps to be disabled in Mode 5. The POS 4 PRA model assumes that two trains of charging (SI) pumps are available and the model includes an available swing pump. Therefore, high pressure SI is modeled as being available in the POS 4 PRA model, when it may not be available based on plant procedures. This is a conservative approach because it reduces the risk in Mode 5 when, in fact, the pump may not be available.

Low pressure SI and recirculation is performed by the RHR pumps and heat exchangers. This is similar to many other Westinghouse NSSS plants, although some plants have low pressure SI pumps separate from the RHR system. Separate low pressure SI pumps are not modeled in any of the POS PRA models. The availability of separate low pressure SI pumps would reduce the risk in Mode 5 if modeled for POS 4. However, the POS 3 PRA model does not credit switching to RHR cooling if AFW cooling fails. This is a conservative approach with respect to the estimated risk for POS 3. Modeling separate low pressure SI pumps for POS 4 and modeling RHR cooling for POS 3 would result in lower risk for both plant operating states. Therefore, the conclusions of the WCAP would not change.

The reactor protection system for the transition risk PRA model is based on a solid state protection system. While many Westinghouse NSSS plants have relay protection systems, the reliability of the two systems is not significantly different, therefore, the model is applicable to both protection systems.

The modeled CCW system consists of two separate trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants. Plants with a common CCW header would be expected to have a higher failure probability for the CCW system. This would affect the risk for both the POS 3 and POS 4 PRA models, however, it would have a greater effect on the POS 4 model, because it would directly contribute to the loss of RHR cooling, whereas it does not directly contribute to the loss of AFW cooling for POS 3. Some insight to the relative effect of different CCW designs can be gained by comparing the POS 3 and POS 4 CDPs for the two cases presented in Table 6-14. The table provides the results from modeling one CCW train unavailable. The difference between the two cases is the number of CCW pumps assumed to be unavailable. The POS 4 results for the two cases do not differ by a large amount, and the POS 3 results differ by even less. However, the POS 4 CDP is more than 20 times greater than the POS 3 CDP for these cases. Plant design differences in the CCW system will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The modeled SW system consists of two trains with one pump per train and one pump that can be aligned to either train. This is a common design among the Westinghouse NSSS plants. Plants with a common SW header would be expected to have a higher failure probability for the

INSERT A

SW system. This would affect the risk for both the POS 3 and POS 4 PRA models, however, it would have a greater effect on the POS 4 model because it would contribute to the loss of RHR cooling, whereas it does not directly contribute to the loss of AFW cooling for POS 3. Some insight to the relative effect of different SW designs can be gained by comparing the POS 3 and POS 4 CDPs for the two cases presented in Table 6-15. The table provides the results from modeling one SW train unavailable. The difference between the two cases is the number of SW pumps assumed to be unavailable. The POS 4 results for the two cases do not differ by a large amount and the POS 3 results differ by even less. However, the POS 4 CDP is approximately 10 times greater than the POS 3 CDP for these cases. Plant design differences in the CCW system will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The electric power system modeled is a two train system with one diesel generator per train. This is a common design among the Westinghouse NSSS plants. Some plants have more redundancy in their design, including shared diesels between units. Based on the results presented in Table 6-10, the loss of offsite power event is a larger percent contributor to the POS 3 risk, than it is to the POS 4 risk and a more redundant design would be more beneficial for Mode 4 (POS 3).

The use containment spray or the containment coolers for backup cooling during recirculation has been modeled only to a limited extent for POS 3 for small LOCAs, and does not have a very large effect on the results (see Table 6-13). More detailed modeling to take credit for these systems may result in a larger risk decrease for POS 4 than it would for POS 3 because the probability of losing RHR cooling is greater than the probability of losing AFW cooling. However, the risk for POS 3 would also decrease and the POS 3 risk is approximately a factor of 7 lower than the risk for POS 4 (see Table 6-9). More detailed modeling will not change the conclusion that there is less risk is associated with a cooldown to Mode 4 (POS 3) than a cooldown to Mode 5 (POS 4).

The model chosen includes many safety system features and support system features that are common among many of the Westinghouse NSSS plants. The evaluation of design differences indicates that while the Westinghouse NSSS plant designs vary for the systems modeled, the model used provides representative results whose conclusions are applicable to all Westinghouse NSSS plants.

Special Initiators

- Loss of CCW
- Loss of SW
- Loss of one DC Bus
- Loss of one AC Bus
- Loss of Instrument Air

Internal Flooding

- CCW Pipe Breaks
- SW Pipe Breaks

Others

- Loss of Offsite Power
- Steam Generator Tube Rupture
- Anticipated Transient Without Scram

POS 1 and POS 7: Plant Response Model

In POS 1, the plant conditions, in terms of RCS and secondary side pressures and temperatures, and system availabilities, are very similar to at-power plant conditions. Therefore, the at-power plant PRA model is used with several modifications. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: Event mitigation is identical to that modeled in the at-power PRA model.

- Medium LOCA: Event mitigation is identical to that modeled in the at-power PRA model.

- Small LOCA: Event mitigation is identical to that modeled in the at-power PRA model.

- RCP Seal LOCAs: A number of events can lead to a loss of seal cooling. These are:

  - Loss of CCW
  - Loss of SW
  - Loss of Offsite Power
  - Internal Flooding Events

  Event mitigation is identical to that modeled in the at-power PRA model.

- Loss of Feedwater Control: This leads to a loss of decay heat removal event which is modeled as a loss of main feedwater. Event mitigation is identical to that modeled in the at-power PRA model.

- Loss of Decay Heat Removal: This can occur when the plant has already transitioned to the auxiliary feedwater or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: Event mitigation is identical to that modeled in the at-power PRA model.
- SG Tube Rupture: Event mitigation is identical to that modeled in the at-power PRA model.
- Secondary Side Breaks: Event mitigation is identical to that modeled in the at-power PRA model.
- Boron Dilution: The positive reactivity insertion event mitigation is identical to that modeled in the at-power model.
- Rod Withdrawal: The positive reactivity insertion event mitigation is identical to that modeled in the at-power model.

The decay heat level in POS 7 is lower than that for POS 1. The lower decay heat level in POS 7 provides additional response time for the operator, however, no credit is taken for lower human error probabilities in POS 7.

POS 2 and POS 6: Plant Response Model

In POS 2, the RCS pressures and temperatures are very similar to at-power plant conditions, although both are being reduced. It is assumed that the plant is at the at-power pressure and temperature conditions for the RCS. The secondary side pressure is at the normal operating pressure. The reactor trip breakers are open so rod withdrawal is no longer a possible event. Signals for safety injection and steamline isolation are blocked. Operator actions are required to start equipment to mitigate a number of the potential events. The switchover from main feedwater to auxiliary feedwater (or startup feedwater) has occurred, therefore, loss of feedwater control is no longer an issue either. The at-power plant PRA model is applicable with several modifications. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- Medium LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- Small LOCA: Event mitigation is identical to that modeled in the at-power PRA model.
- RCP Seal LOCAs: A number of events can lead to loss of seal cooling. These are:
  - Loss of CCW
  - Loss of SW
  - Loss of Offsite Power
  - Internal Flooding Events

  Event mitigation is identical to that modeled in the at-power PRA model.

- Loss of Decay Heat Removal: This event is the failure of the decay heat removal source which is either the auxiliary feedwater system or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: Event mitigation is identical to that modeled in the at-power PRA model.

- SG Tube Rupture: Event mitigation is identical to that modeled in the at-power PRA model.

- Secondary Side Breaks: Event mitigation is identical to that modeled in the at-power PRA model.

POS 3 and POS 5: Plant Response Model

is less than

In POS 3, the RCS pressures and temperatures are significantly reduced compared to POS 1 and POS 2. Therefore, the LOCA events remain applicable, but at a reduced frequency. The secondary side ~~remains at~~ normal operating pressure. Similar to POS 2, the reactor trip breakers are open so rod withdrawal is no longer a possible event. Also, like POS 2, signals for safety injection and steamline isolation are blocked, therefore, operator actions are required to start equipment to mitigate a number of the potential events. Accumulators are isolated. Again, loss of feedwater control is no longer an issue. The at-power plant PRA model is applicable, with several modifications, to model this POS. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Large LOCA: ~~Event mitigation is identical to that modeled in the at-power PRA model except for the availability of accumulators.~~ INSERT B

- Medium LOCA: ~~Event mitigation is identical to that modeled in the at-power PRA model except for the availability of accumulators.~~ INSERT B

- Small LOCA: ~~Event mitigation is identical to that modeled in the at-power PRA model except for the availability of accumulators.~~ INSERT B

- Loss of Decay Heat Removal: This event is the failure of the decay heat removal source which is either the auxiliary feedwater system or startup feedwater system. The event is failure of the system to continue to operate. Mitigation of the event is identical to the loss of main feedwater event, as modeled in the at-power PRA model, except for removing credit for the auxiliary or startup feedwater pump that failed and initiated the event.

- Loss of Offsite Power: ~~Event mitigation is identical to that modeled in the at-power PRA model.~~ INSERT B

- SG Tube Rupture: ~~Event mitigation is identical to that modeled in the at-power PRA model.~~ INSERT B

- Secondary Side Breaks: ~~Event mitigation is identical to that modeled in the at-power PRA model.~~ INSERT B

INSERT B

The safety injection system included in the at-power PRA model is used for the POS 3 and POS 5 models, modified to remove the accumulators which have been isolated, and the automatic start of the system in response to the event disabled. Operator action is required to manually start the safety injection system from the control room in these plant operating states.

POS 4: Plant Response Model

In POS 4, the RCS pressures and temperatures are significantly reduced compared to POS 3. The secondary side conditions have also been significantly reduced. Therefore, the LOCA events, including RCP seal LOCAs, SGTR, and secondary side breaks are no longer applicable. The reactor trip breakers remain open so rod withdrawal is not a possible event. Core cooling has been switched to the RHR system (shutdown cooling). Due to this switchover, loss of inventory events are important. In addition, the loss of decay heat removal event is now related to the RHR system. Also, like POS 2 and POS 3, signals for safety injection and steamline isolation are blocked, therefore, operator actions are required to start equipment to mitigate a number of the potential events. Finally, cold overpressurization needs to be considered.

The at-power plant PRA model is no longer applicable due to the significant changes in the operating temperature and pressure on both the primary and secondary sides, available systems, and the events that can occur. The initiating events that need to be considered are listed on Table 6-6. Each is listed below with explanatory notes.

- Loss of Inventory: Loss of inventory events are associated with alignment issues when transferring to RHR (shutdown) cooling. These events can be divided between isolable events and non-isolable events. If the leak is not isolable or if isolation fails, then high pressure injection, via SI/charging, is required followed by recirculation. If isolation is successful, then high pressure injection is required to make up for the inventory lost prior to the leak being isolated. Decay heat removal is also required.

- Loss of Decay Heat Removal: Loss of decay heat removal events are primarily loss of RHR events. These can occur during the switchover from AFW cooling to RHR cooling or from failure of the RHR system. Mitigation of the event depends on the availability of the AFW system. SG cooling credits the MD AFW pumps, but not the TD AFW pump due to the reduced secondary side pressure. If AFW is not available, then feed and bleed ~~followed by high pressure recirculation provides a path to success for cooling.~~ is required. While on feed and bleed, recovery of RHR is modeled.

- Loss of Offsite Power: The causes of loss of offsite power events in POS 4 are similar to these events that occur when at-power. The consequences are different in that RCP seal LOCAs are no longer an issue, but the TD AFW pump is not available for decay heat removal. In addition, the lower decay heat level provides additional time for recovery from the event. For event mitigation, recovery of offsite power is initially addressed, and if successful, is followed by decay heat removal equipment. If offsite power recovery fails, then the availability of decay heat removal mitigating systems with power from the EDGs is addressed. Decay heat removal can be provided by the RHR system, AFW systems (MD pumps only), or by feed and bleed~~, with follow-up recirculation~~. While on feed and bleed, recovery of RHR is modeled.

- Cold Overpressurization: Mitigation of cold overpressure events requires the operators to control charging and letdown. If this is not successful, then pressure relief via a pressurizer PORV or a RHR relief valve is necessary. Following successful pressure relief, the pressurizer PORV or a RHR relief valve is required to re-close. If it does not reclose, a loss of inventory event occurs. This model does not address mitigation of the possible loss of inventory event, but ~~conservatively~~ assumes the event leads to core damage.

- Boron Dilution: The high flux at shutdown alarm will indicate the event. The operators are required to identify and terminate the dilution source. If the dilution is not terminated, the operator must initiate boration.

6.3.1.2 Time in Each Plant Operating State

The time spent in each POS is important to the calculation of initiating event probabilities. When shutting the plant down due to Technical Specification requirements, the Technical Specifications control the maximum length of time allowed in each mode.

If inoperable equipment is not restored to operable status, or the Required Actions and associated Completion Times are not met, the Required Actions require that the unit be in Mode 3 in 6 hours and Mode 5 in 36 hours. Based on this, the time spent in the POSs while shutting the plant down are provided in Table 6-7.

The time spent in each POS when returning to power is not controlled by the Technical Specifications, but rather on the reason for the forced outage and satisfying the applicable Technical Specifications prior to returning to power operation. Information was collected from several Westinghouse NSSS plants for startup times following a forced non-refueling outage. Table 6-7 provides a summary of the time in each POS based on this information. Note that the times begin from the time that the decision is made to shutdown and do not include any time prior to the shutdown because the Completion Times associated with the Required Actions to restore the equipment to operable status are not changing.

6.3.1.3 Initiating Event Probabilities

Table 6-8 provides a summary of the initiating event probabilities. Event probabilities are used, not frequencies, since the risk metric being used (core damage) is determined on a shutdown basis, not a yearly basis. The core damage probability (CDP) can be converted to a CDF by multiplying the CDP by the number of shutdowns per year. The event probabilities for each POS are determined by multiplying the event frequencies by the time in the POS. Additional information on each initiating event probability is provided in the following:

- Large LOCA: In POS 1, POS 2, POS 6, and POS 7 the at-power frequency is used. In POS 3 and POS 5 the RCS pressure is significantly reduced compared to at-power, therefore, the frequency is reduced by a factor of 20, based on Reference 4. The at-power initiating event frequency is 5.0E-06/yr.

Table 6-7 Summary of the Time Spent in the Plant Operating States

Plant Operating State	Time in the Plant Operating State	Justification
1	7 hours	POS 1 covers the transition out of Mode 1, through Mode 2, and into Mode 3. The Technical Specification Required Actions require that this be completed in 6 hours. An additional 1 hour was added to prepare for the shutdown.
2	3 hours	POS 2 covers the mid part of Mode 3 (also viewed as transitioning from the upper end of Mode 3 to the lower end of Mode 3). It is assumed that Mode 4 will be entered in 13 hours. This results in 6 hours (13 hours - 7 hours) to go from the upper end of Mode 3 to the upper end of Mode 4. Assuming that half this time is used to transition from the upper end of Mode 3 to the lower end of Mode 3, the time in POS 2 in 3 hours.
3 (Shutdown to Mode 5)	3 hours	POS 3 covers the lower end of Mode 3 to the upper end of Mode 4. Following the logic presented in POS 2, half the time to go from Mode 3 to Mode 4 (6 hours) is assigned to POS 3.
4 (shutdown)	24 hours	POS 4 (shutdown) covers the transition through Mode 4 to Mode 5. Since the Technical Specification Required Actions require Mode 5 to be entered in 36 hours, and an additional 1 hour was added to prepare for the shutdown, the time in this POS is 37 - 7 - 3 - 3 = 24 hrs.
4 (startup)	46 hours	POS 4 (startup) covers the transition from Mode 5 to Mode 4. As discussed in Section 6.3.1.2, 46 hrs is based on plant operating experience.
5	19 hours	POS 5 covers the transition from Mode 4 to the lower end of Mode 3. As discussed in Section 6.3.1.2, 19 hrs is based on plant operating experience.
6	52 hours	POS 6 covers the middle part of Mode 3 (also viewed as moving from the lower end of Mode 3 to the upper end of Mode 3). As discussed in Section 6.3.1.2, 52 hrs is based on plant operating experience.
7	39 hours	POS 7 covers the upper end of Mode 3, through Mode 2, and the transition through Mode 1 to power operation. As discussed in Section 6.3.1.2, 39 hrs is based on plant operating experience.
3 (shutdown and repair in Mode 4)	49 hours	For a shutdown and repair in Mode 4, the 3 hours to go from Mode 3 to Mode 4 are added to the 46 hours for repair assumed for POS 4. This conservatively includes the startup time assigned to POS 4. See the discussion for POS 4 (startup).

Table 6-8 Summary of Initiating Event Probabilities for Each POS (3 hr.)

Initiating Event	Initiating Event Probabilities							
	POS 1	**POS 2**	**POS 3**	**POS 4**	**POS 5**	**POS 6**	**POS 7**	POS 3 (49 hr.)
Large LOCA	4.0E-09	1.7E-09	8.6E-11	NA	5.4E-10	3.0E-08	2.2E-08	1.4E-09
Medium LOCA	3.2E-08	1.4E-08	6.9E-10	NA	4.3E-09	2.4E-07	1.8E-07	1.1E-08
Small LOCA	2.4E-06	1.0E-06	5.1E-08	NA	3.3E-07	1.8E-05	1.3E-05	8.4E-07
Interfacing Systems LOCA	1.1E-09	4.7E-10	4.7E-10	NA	3.0E-09	8.1E-09	6.1E-09	7.6E-09
Reactor Vessel Rupture	8.0E-11	3.4E-11	3.4E-11	NA	2.2E-10	5.9E-10	4.5E-10	5.6E-10
Loss of Inventory	NA	NA	NA	1.4E-04	NA	NA	NA	NA
RCP Seal LOCAs[1]	9.6E-07 + FT	4.1E-07 + FT	NA	NA	NA	7.1E-06 + FT	5.3E-06 + FT	NA
Loss of Feedwater Control[3]	6.8E-02 + FT	NA	NA	NA	NA	NA	8.8E-02 + FT	NA
Loss of Decay Heat Removal	NA	FT[2]	FT[2]	FT[2]	FT[2]	FT[2]	NA	FT[2]
Loss of Offsite Power	2.0E-05	8.5E-06	8.5E-05	2.0E-04	5.4E-05	1.5E-04	1.1E-04	1.4E-04
Cold Overpressurization	NA	NA	NA	1.5E-03	NA	NA	NA	NA
SG Tube Rupture	2.2E-06	9.3E-07	9.3E-07	NA	5.9E-06	1.6E-05	1.2E-05	1.5E-05
Secondary Side Breaks								
• Inside containment	5.7E-06	2.5E-06	2.5E-06	NA	1.6E-05	4.3E-05	3.2E-05	4.0E-05
• Outside containment	5.7E-06	2.5E-06	2.5E-06	NA	1.6E-05	4.3E-05	3.2E-05	4.0E-05
Boron Dilution	Note 4	NA	NA	4.2E-03	NA	NA	Note 4	NA
Rod Withdrawal	7.9E-05	NA	NA	NA	NA	NA	4.4E-04	NA

Notes:

1. RCP Seal LOCAs are caused by loss of CCW and loss of SW events, in addition to several flooding events that lead to degraded CCW and SW. The numerical values are the IE probability due to flooding events. The FT indicates that fault tree evaluations are used to determine the IE probability for the loss of CCW and loss of SW events. This evaluation is done as part of the model quantification.
2. The FT indicates that fault tree evaluations are used to determine the IE probability for the loss of decay heat removal events. This evaluation is done as part of the model quantification.
3. The loss of AFW is modeled as an initiating event for the time in the POS after AFW has been initiated. Fault tree evaluations are used to determine the IE probability.
4. Boron dilution IE probability is included in the rod withdrawal probability.

- Medium LOCA: In POS 1, POS 2, POS 6, and POS 7 the at-power frequency is used. In POS 3 and POS 5 the RCS pressure is significantly reduced compared to at-power, therefore, the frequency is reduced by a factor of 20, based on Reference 4. The at-power initiating event frequency is 4.0E-05/yr.

- Small LOCA: In POS 1, POS 2, POS 6, and POS 7 the at-power frequency is used. In POS 3 and POS 5 the RCS pressure is significantly reduced compared to at-power, therefore, the frequency is reduced by a factor of 20, based on Reference 4. The at-power initiating event frequency is 3.0E-03/yr.

- Interfacing Systems LOCA: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 1.36E-06/yr.

- Reactor Vessel Rupture: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 1.0E-07/yr.

- Loss of Inventory: This event is only applicable in POS 4. Based on Reference 4, the loss of inventory events are divided into three categories with the noted initiating event frequencies:

  - HLOCA – an event that results from an inadvertent transfer of reactor coolant out of the RCS
    IE frequency = 7.0E-03/yr

  - JLOCA – an event that occurs in a system connected to the RCS
    IE Frequency = 8.0E-03/yr

  - KLOCA – an event that results from a maintenance activity
    IE frequency = 3.0E-03/yr

  The total IE frequency is 1.8E-02/yr. The event probability is then determined by factoring in the length of time in POS 4.

- RCP Seal LOCAs: These are events that occur primarily due to a failure of seal cooling, such as, loss of CCW, loss of SW, and loss of offsite power. In addition, flooding events that are the result of breaks in the CCW or SW system can also result in RCP seal LOCAs due to loss of seal cooling. Loss of offsite power is addressed as a separate initiator and is discussed further below.

  The total CCW flooding frequency is 9.9E-04/yr and the total SW flooding frequency is 2.1E-04/yr. The event probabilities are then determined by factoring in the length of time for the applicable POS (POS 1, POS 2, POS 6, POS 7).

  The IE frequencies for the loss of CCW and loss of SW events are determined from fault tree evaluations in the model quantification. The component operating times in these fault trees are changed as required for the time in the POS.

- <u>Loss of Feedwater Control</u>: This event is applicable in POS 1 and POS 7 only. It considers reactor trips caused by loss of feedwater during the power reduction (POS 1) and power ascension (POS 7). Section 8.4 of Reference 5 provides a value for the probability of a reactor trip during a shutdown as 0.068 and the probability of a reactor trip during a startup as 0.088.

- <u>Loss of Decay Heat Removal</u>: This event is applicable in all the POSs. In POS 1 and POS 7 it is considered part of the Loss of Feedwater Control event or the loss of AFW after AFW is initiated. In POS 2, POS 3, POS 5, and POS 6 it is considered loss of auxiliary feedwater. In POS 4 it is considered loss of RHR. The IE frequencies for the loss of AFW and the loss of RHR are determined from fault tree evaluations in the model quantification.

- <u>Loss of Offsite Power</u>: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power IE frequency is used. The at-power IE frequency is 2.5E-02/yr. In POS 4 the events in Reference 6 were reviewed and an IE frequency determined. This frequency was similar to the 2.5E-02/yr value for the at-power LOSP event, therefore, the IE probability for POS 4 was also based on the 2.5E-02/yr value.

- <u>Cold Overpressurization</u>: This event only applies to POS 4. References 4 and 7 were reviewed to determine an appropriate frequency. Reference 7 was used since it provided a higher value of 1.8E-01/yr.

- <u>SG Tube Rupture</u>: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 2.7E-03/yr.

- <u>Secondary Side Breaks Inside and Outside Containment</u>: In POS 1, POS 2, POS 3, POS 5, POS 6, and POS 7 the at-power frequency is used. The at-power initiating event frequency is 7.2E-03/yr.

- <u>Rod Withdrawal</u>: This event is only considered in POS 1 and POS 7. For POS 1 and POS 7, it is combined with the Boron Dilution initiating event and modeled as a positive reactivity insertion event with the at-power IE frequency of 9.9E-02/yr.

- <u>Boron Dilution</u>: For POS 1 and POS 7, it is combined with the Rod Withdrawal initiating event, and modeled as a positive reactivity insertion event with the at-power IE frequency of 9.9E-02/yr. For POS 4, Boron Dilution is modeled by itself with a frequency of 6.0E-05/hr based on Reference 4.

6.3.1.4 System Unavailabilities

System models for a typical Westinghouse NSSS plant were included in the model. As noted in Section 6.3.1, the base PRA model includes unavailability models for the following system configurations:

- AFW system – two MD pumps and one TD pump
- ECCS – two train system with each train including a high head and low head subsystem
- RPS – SSPS

- SW – two train system
- CCW – two train system
- Electrical power system – a two train system with one EDG per train

All system unavailabilities due to test and maintenance activities that are typically included in the at-power PRA model were eliminated. When a plant is starting up or shutting down, the specific plant configuration is modeled. The availability of components is known. Therefore, all component unavailability related to testing and maintenance activities were removed from the model.

6.3.1.5 Operator Actions

Operator actions and human error probabilities are key parameters for mitigating events in the transition states, particularly after blocking the automatic signals. Safety injection and steamline isolation signals are blocked in POS 2 and 3. Safety injection, steamline isolation, and AFW start signals are blocked in POS 4. For mitigation of several events, consecutive dependent operator actions are required. Dependencies between operator actions are addressed and accounted for as necessary.

6.3.2 Model Quantification

The base core damage probability results from quantifying the POS models are provided on Table 6-9. These results assume all the equipment is available, that is, no equipment is out of service or inoperable and all the unavailabilities in the model for test and maintenance are set to zero. Based on this, the core damage probabilities for transitioning to and returning from Mode 4 (POS 3) and to and from Mode 5 (POS 4) are:

- CDP (transition to and from POS 3/Mode 4) = ~~5.49E-06~~ 6.02 E-06
- CDP (transition to and from POS 4/Mode 5) = ~~1.27E-05~~ 9.52 E-06

There is ~~a significant~~ an increase in CDP with the additional transition required to achieve Mode 5 as opposed to Mode 4. This is related to the risk associated with the transition from SG cooling to the shutdown (RHR) cooling and operator actions being required to initiate event mitigation equipment. The key initiating event is loss of RHR cooling with operator failure to establish alternate cooling.

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Table 6-9	Core Damage Probability Results by Plant Operating State		
POS	CDP	Time in POS (hours)	CDP/Hour in POS
1	2.18E-07	7	3.11E-08
2	1.66E-07	3	5.53E-08
3	7.09E-08	3	2.36E-08
4	7.21E-06	70	1.03E-07
5	4.79E-07	19	2.62E-08
6	3.30E-06	52	6.35E-08
7	1.26E-06	39	3.23E-08

Since the primary objective of this analysis is to identify the appropriate endstate, Mode 4 or Mode 5, an examination of the initiating event contributors to core damage for each of these endstates provides relevant insights. This information is provided on Table 6-10. The values in this table are given as core damage probabilities based on the plant remaining in the POS for the time indicated in Table 6-9 with all systems available. The following is concluded:

- ~~Adjusting for the differences in time in the POS, the~~ The core damage probability in Mode 5 (POS 4) is more than ~~4~~ 6 times greater than that for Mode 4 (POS 3).

- The largest initiating event contributor in either mode is a loss of decay heat removal. In Mode 4 the plant is using AFW for removal of decay heat and the event is initiated by a failure of the operating pump. The other motor-driven pump and the turbine-driven AFW pump are available. In Mode 5 the plant is using the RHR system for decay heat removal. Included in this initiating event is the switchover from SG (AFW) cooling to shutdown (RHR) cooling. All actuations of mitigating systems are by operator actions and the turbine-driven pump is not available.

- Small LOCAs or loss of inventory events are larger risk contributors in Mode 5 than in Mode 4. In Mode 5, loss of inventory events can be initiated by the alignment change from SG cooling to shutdown cooling. Event mitigation relies on operator actions in both endstates.

- ~~The loss of offsite power event is also a larger risk contributor in Mode 5 than in Mode 4. One significant difference is that in Mode 4 the turbine-driven pump is available for event mitigation, but not in Mode 5.~~

Based on these quantitative results, it is concluded that Mode 4 is preferred over Mode 5 as the endstate. The initiating event contributions for POS 3 and POS 4 in Table 6-9 are shown in Table 6-10. The top 100 cutsets for the POS 3 and POS 4 quantifications are presented in Appendix A. A sensitivity case that examines the time duration for POS 4 is presented in Section 6.5.

INSERT C

Table 6-9 Core Damage Probability Results by Plant Operating State			
POS	Core Damage Probability		Time in POS (hours)
	Cooldown to POS 3	Cooldown to POS 4	
1	2.18E-07	2.18E-07	7
2	1.66E-07	1.66E-07	3
3	5.95E-07	7.09E-08	49 (cooldown to POS 3) 3 (Cooldown to POS 4)
4	NA	4.03E-06	70
5	4.79E-07	4.79E-07	19
6	3.30E-06	3.30E-06	52
7	1.26E-06	1.26E-06	39
Total	6.02E-06	9.52E-06	

INSERT D

Table 6-10 Summary of Initiating Event Contribution for POS 3 and POS 4		
	Core Damage Probability	
Initiating Event	POS 3 (Mode 4) Endstate	POS 4 (Mode 5) Endstate
Loss of Decay Heat Removal/RHR	63.4%	71.8%
Loss of Offsite Power	12.7%	10.1%
Small LOCA/Loss of Inventory	5.7%	13.6%
SG Tube Rupture	9.4%	NA
Secondary Side Break Outside Containment	2.5%	NA
Secondary Side Break Inside Containment	5.0%	NA
Interfacing Systems LOCA	0.1%	NA
Reactor Vessel Rupture	<0.1%	NA
Cold Overpressure	NA	<0.1%
Boron Dilution	NA	4.4%

6.4 EVALUATION OF TECHNICAL SPECIFICATION REQUIRED ACTION ENDSTATES

This section provides an evaluation of each Technical Specification for which the endstate is proposed to be changed from Mode 5 to Mode 4. The Technical Specifications are listed in numerical order by Specification number as contained in NUREG-1431 (Reference 3). Qualitative and quantitative evaluations are presented to support the endstate change from Mode 5 to Mode 4.

Quantitative evaluations are performed if the components/systems are modeled in the POS risk models described in Section 6.3. In the quantitative evaluation, specific components/systems are modeled as inoperable in the POS risk models and the conditional CDP for each applicable POS is calculated. The risk models for POS 5, 6, and 7 model the restart of the unit after the inoperable equipment has been restored to operable status, therefore, the CDPs for these POSs are not requantified. The risk calculation results are presented ~~along with the base case results from Section 6.3.2~~? In describing the results, the total CDP representing a cool down to POS 4 (Mode 5) and startup is compared to the total CDP representing a cool down to POS 3 (Mode 4) and startup.

for a cooldown to POS 3 and for a cooldown to POS 4.

6.4.1 Technical Specification 3.3.2 – Engineered Safety Features Actuation System (ESFAS) Instrumentation

The ESFAS instrumentation initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. There are numerous ESFAS function LCOs. For this Technical Specification, each function is addressed separately.

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Table 6-10 Summary of Initiating Event Contribution for POS 3 and POS 4		
	Core Damage Probability	
Initiating Event	POS 3 (Mode 4) Endstate	POS 4 (Mode 5) Endstate
Loss of Decay Heat Removal/RHR	30.4%	57.9%
Loss of Offsite Power	23.1%	17.5%
Small LOCA/Loss of Inventory	11.1%	16.7%
SG Tube Rupture	18.2%	NA
Secondary Side Break Outside Containment	5.0%	NA
Secondary Side Break Inside Containment	9.8%	NA
Interfacing Systems LOCA	0.2%	NA
Reactor Vessel Rupture	0.1%	NA
Cold Overpressure	NA	<0.1%
Boron Dilution	NA	7.9%

Function 1. a. Safety Injection – Manual Initiation

Description

The safety injection system provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting the peak clad temperature to ≤ 2200°F), and

2. Boration to ensure recovery and maintenance of shutdown margin ($k_{eff} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other functions (e.g., reactor trip).

The operator can initiate both trains of safety injection at any time from the control room by pushing one of two push buttons. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

Limiting Condition for Operation

Two channels shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 include automatic actuation of safety injection and manual actuation of the equipment, however, credit is not taken for the manual initiation of safety injection. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel is inoperable, the other channel is available for the operator to initiate safety injection. If the operator is shutting down the unit because of an inoperable channel, there will be a heightened awareness that this protection feature is not fully operational. The operators can be expected to be prepared to address a unit transient requiring safety injection with one channel inoperable. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. The LERP in Mode 4 would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel is inoperable, the other channel is available for the operator to initiate safety injection. ~~In addition, the two trains of automatic actuation logic are required to be operable to support the actuation of safety injection equipment.~~ Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4. There will be a heightend operator awareness that this protection feature is not fully operational.

Function 1. b. Safety Injection – Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 1. a., Safety Injection – Manual Initiation.

There are two trains for automatic actuation. In Mode 4 adequate time is available to manually actuate required components in the event of a design basis accident, however, because of the large number of components actuated, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 model the block of the automatic SI signal for POSs 2, 3, 4, 5, and 6. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate safety injection. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring safety injection knowing that manual initiation may be required. In this case, the operator will have both manual channels available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions and mitigation strategies, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate safety injection. In addition, ~~the~~ there are two channels of manual actuation ~~are required to be operable to~~ that can perform the function. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 2. a. Containment Spray – Manual Initiation

Description

The containment spray system provides three primary functions:

1. Lowers containment pressure and temperature after a HELB in containment,

2. Reduces the amount of radioactive iodine in the containment atmosphere, and

3. Adjusts the pH of the water in the containment recirculation sump after a LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure,
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure, and
- Minimize corrosion of the components and systems inside containment following a LOCA.

The operator can initiate containment spray at any time from the control room by simultaneously actuating two containment spray actuation switches in the same train. Because an inadvertent actuation of containment spray could have undesirable consequences, two switches must be actuated simultaneously. There are two sets of two switches in the control room. Simultaneously actuating the two switches in either set will start both trains of containment spray.

Limiting Condition for Operation

Two channels per train and two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel or train inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel or train must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on evaluating the core damage probability. The containment spray system does not have a significant impact on the core damage probability for the plant operating states modeled as described in Section 6.3. This is confirmed by the results in Table 6-13. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel or train is inoperable, the other train is available for the operator to initiate containment spray. If the operator is shutting down the unit because of an inoperable channel or train, there will be a heightened awareness that this protection feature is not fully operational. A cool down to Mode 4 places the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. The LERP in Mode 4 would be small due to the lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression due to lower temperatures and pressures, and the corresponding increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel or train is inoperable, the other train is available for the operator to initiate containment spray. ~~In addition, the two trains of automatic actuation logic are required to be operable to support the actuation of containment spray equipment.~~ Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are required to be operable in Mode 4. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

There will be a heightened operator awareness that this protection feature is not fully operational.

Function 2. b. Containment Spray – Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 2. a., Containment Spray – Manual Initiation.

There are two trains for automatic actuation. In Mode 4 adequate time is available to manually actuate required components in the event of a design basis accident, however, because of the large number of components actuated, actuation is simplified by the use of the manual actuation push buttons. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on evaluating the core damage probability. The containment spray system does not have a significant impact on the core damage probability for the plant operating states modeled as described in Section 6.3. This is confirmed by the results in Table 6-13. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate containment spray. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring containment spray knowing that manual initiation may be required. In this case, the operator will have both manual trains available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate containment spray. In addition, ~~thee~~ there are two trains for manual initiation ~~are required to be operable to support the actuation of~~ that can actuate containment spray equipment. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are ~~required to be operable in Mode 4~~ available. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 3. a (1) Containment Isolation, Phase A Isolation, Manual Initiation

Description

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

There are two separate Containment Isolation signals, Phase A and Phase B. Phase A isolation isolates all automatically isolable process lines, except CCW, at a relatively low containment pressure indicative of primary or secondary system leaks. Phase A containment isolation is actuated automatically by SI, or manually via the automatic actuation logic. All process lines penetrating containment, with the exception of CCW, are isolated.

Manual Phase A Containment Isolation is accomplished by either of two switches in the control room. Either switch actuates both trains. Note that manual actuation of Phase A Containment Isolation also actuates Containment Purge and Exhaust Isolation.

Limiting Condition for Operation

Two channels shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel is inoperable, the other channel is available for the operator to initiate containment isolation. If the operator is shutting down the unit because of an inoperable channel, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring containment isolation with one channel inoperable. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. The LERP in Mode 4 would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel is inoperable, the other channel is available for the operator to initiate containment isolation. In addition, the two trains of automatic actuation logic are ~~required to be operable to support the actuation of~~ available to actuate containment isolation equipment. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are ~~required to be operable in Mode 4~~ available. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 3. a (2) Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 3. a (1), Containment Isolation, Phase A Isolation, Manual Initiation.

There are two trains for automatic actuation. In Mode 4 adequate time is available to manually actuate required components in the event of a design basis accident, however, because of the large number of components actuated, actuation is simplified by the use of the manual switches. Automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate containment isolation Phase A. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring containment isolation knowing that manual initiation may be required. In this case, the operator will have both manual channels available. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate containment isolation Phase A. In addition, the two channels of manual actuation are available ~~required to be operable~~ to perform the function. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. In addition, the containment, containment isolation valves, containment spray system, and containment cooling system are available ~~required to be operable in Mode 4~~. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 3. b (1) Containment Isolation, Phase B Isolation, Manual Initiation

Description

Containment Isolation provides isolation of the containment atmosphere, and all process systems that penetrate containment, from the environment. This function is necessary to prevent or limit the release of radioactivity to the environment in the event of a large break LOCA.

The Phase B signal isolates CCW. Manual Phase B containment isolation is accomplished by the same switches that actuate containment spray. When the two switches in either set are actuated simultaneously, Phase B containment isolation and containment spray will be actuated in both trains.

Limiting Condition for Operation

Two channels per train and two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel or train inoperable.

Current Required Action Endstate

The current endstate for Required Action B.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within 48 hours, or the unit must be in Mode 3 in 54 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2.2 to be in Mode 4 in 60 hours if the inoperable channel or train is not restored to operable status in 48 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change. The manual actuation of containment isolation Phase B uses the same switches and logic as containment spray. The bases for the proposed change provided for Function 2. a., Containment Spray – Manual Initiation, and for Function 3. a (1), Containment Isolation, Phase A Isolation, Manual Initiation, also apply to the manual initiation of containment isolation Phase B.

Defense-in-Depth Considerations

The defense-in depth considerations provided for Function 2. a., Containment Spray – Manual Initiation, and for Function 3. a (1), Containment Isolation, Phase A Isolation, Manual Initiation, also apply to the manual initiation of containment isolation Phase B.

Function 3. b (2) Containment Isolation, Phase B Isolation, Automatic Actuation Logic and Actuation Relays

Description

The general description is the same as that presented for Function 3. b (1), Containment Isolation, Phase B Isolation, Manual Initiation.

There are two trains for automatic actuation. The same channels and trains used for actuating containment spray are used for actuating containment isolation Phase B. Just as for containment spray, the automatic actuation logic and relays are required to be operable to support the manual initiation of containment isolation Phase B.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 are based on core damage probability and do not model containment isolation. Therefore, a qualitative evaluation is performed for this proposed endstate change. The automatic actuation of containment isolation Phase B uses the same channels and logic as containment spray. The bases for the proposed change provided for Function 2. b., Containment Spray – Automatic Actuation Logic and Actuation Relays, and Function 3. a (2), Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays also apply to the automatic actuation of containment isolation Phase B.

Defense-in-Depth Considerations

The defense-in depth considerations provided for Function 2. b., Containment Spray – Automatic Actuation Logic and Actuation Relays, and Function 3. a (2), Containment Isolation, Phase A Isolation, Automatic Actuation Logic and Actuation Relays, also apply to the automatic actuation of containment isolation Phase B.

Function 7. a. Automatic Switchover to Containment Sump, Automatic Actuation Logic and Actuation Relays

Description

At the end of the injection phase of a LOCA, the RWST will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

There are two trains for automatic actuation and the logic and actuation relays consist of the same features and operate in the same manner as described for Function 1. b.

Limiting Condition for Operation

Two trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One train inoperable.

Current Required Action Endstate

The current endstate for Required Action C.2.2 is Mode 5. Specifically, the inoperable train must be restored to operable status within 24 hours, or the unit must be in Mode 3 in 30 hours and Mode 5 in 60 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2.2 to be in Mode 4 in 36 hours if the inoperable train is not restored to operable status in 24 hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 do not include explicit modeling of two trains for this function. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one train is inoperable, the other train is available to initiate switchover to the containment sump. In addition, if the operator is shutting down the unit because of an inoperable train, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring safety injection and recirculation knowing that manual initiation of the switchover from the RWST to the containment sump may be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one train is inoperable, the other train is available to initiate switchover to the containment sump. In addition, the operator can perform the switchover manually. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Function 7. b and 7. c. Automatic Switchover to Containment Sump – Refueling Water Storage Tank (RWST) Level – Low Low Coincident With Safety Injection, and RWST Level – Low Low Coincident With Containment Sump Level – High

Description

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. Automatic switchover occurs only if the RWST low low level signal is coincident with SI. This prevents accidental switchover during normal operation.

In some units, additional protection from spurious switchover is provided by requiring a Containment Sump Level – High signal as well as RWST Level – Low Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level – High signal must be present, in addition to the SI signal and the RWST Level – Low Low signal, to transfer the suction of the RHR pumps to the containment sump.

The RWST has four level transmitters. Units with the containment sump level circuitry also have four channels for the sump level instrumentation. The logic requires two out of four channels to initiate the switchover from the RWST to the containment sump.

Limiting Condition for Operation

Four channels shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

One channel inoperable.

Current Required Action Endstate

The current endstate for Required Action K.2.2 is Mode 5. Specifically, the inoperable channel must be restored to operable status within [6] hours, or the unit must be in Mode 3 in [12] hours and Mode 5 in [42] hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action K.2.2 to be in Mode 4 in [18] hours if the inoperable channel is not restored to operable status in [6] hours.

Basis for Proposed Change

The risk models described in Section 6.3.1 do not include explicit modeling of four channels for this function. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel is inoperable, the other three channels are available to initiate switchover to the containment sump In addition, if the operator is shutting down the unit because of an inoperable channel, there will be a heightened awareness that this protection feature is not fully operational. The operators would be prepared to address a unit transient requiring safety injection and recirculation knowing that manual initiation of the switchover from the RWST to the containment sump may be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one channel is inoperable, the other three channels are available to initiate switchover to the containment sump. The system redundancy is such that a single channel failure in addition to one channel being inoperable will not defeat the initiation of switchover from the RWST to the containment sump. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.2 Technical Specification 3.3.7 – Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

Description

The CREFS provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. During normal operation, the Auxiliary Building Ventilation System provides control room ventilation. Upon receipt of an actuation signal, the CREFS initiates filtered ventilation and pressurization of the control room.

The actuation instrumentation consists of redundant radiation monitors in the air intakes and control room area. A high radiation signal from any of these detectors will initiate both trains of the CREFS. The operator can initiate the CREFS at any time by using either of two switches in the control room. The CREFS is also actuated by a SI signal.

Limiting Condition for Operation

Two trains and [2] channels shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel or train for one or more functions are inoperable, Required Action A.1 requires the operator to place one train of CREFS in emergency mode. If one or more functions with two channels or two trains are inoperable, Required Actions B.1.1 and B.1.2 require the operator to place one or both trains of CREFS in emergency mode. In the unlikely event that this does not occur, the inoperable equipment does not increase the likelihood of an initiating event. An independent initiating event with a radioactive release must occur ~~along with core damage~~ for radiation in the control room to be a concern. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The system design provides redundancy and defense in depth from the multiple channels, trains, and functions available to actuate CREFS. If one or two channels or trains in one or more functions are inoperable, the Required Actions require one or both CREFS trains to be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. In the unlikely event that this is not accomplished and Condition C is entered, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. The system design maintains sufficient defense-in-depth when the endstate is changed from Mode 5 to Mode 4.

6.4.3 Technical Specification 3.3.8 – Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation

Description

The FBACS ensures that radioactive materials in the fuel building atmosphere following a fuel handling accident [involving handling recently irradiated fuel] or a LOCA are filtered and adsorbed prior to exhausting to the environment. The system initiates filtered ventilation of the fuel building automatically following receipt of a high radiation signal (gaseous or particulate) or a SI signal. Initiation may also be performed manually as needed from the main control room.

High gaseous and particulate radiation, each monitored by either of [two] monitors, provides FBACS initiation. Each FBACS train is initiated by high radiation detected by a channel dedicated to that train. There are a total of [two] channels, one for each train. Each channel contains a gaseous and particulate monitor. High radiation detected by any monitor or an SI signal from the ESFAS initiates fuel building isolation and starts the FBACS.

Limiting Condition for Operation

Two trains and [two] channels shall be operable.

Applicability

Modes 1, 2, 3, and 4 during movement of [recently] irradiated fuel assemblies in the fuel building.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system does not affect conditional core damage probability and is not modeled in the risk models described in Section 6.3.1. FBACS is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one channel or train for one or more functions are inoperable, Required Action A.1 requires the operator to place one train of FBACS in operation. If one or more functions with two channels or two trains are inoperable, Required Actions B.1.1 and B.1.2 require the operator to place one train of FBACS in operation or both trains in emergency mode. In the unlikely event that this does not occur, the inoperable equipment does not increase the likelihood of an initiating event. An independent initiating event (e.g., LOCA or fuel handling accident) must occur to require the operation of FBACS. A cool down to Mode 4 reduces the likelihood of a LOCA, leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions and mitigation strategies, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2.

Defense-in-Depth Considerations

The system design provides redundancy and defense in depth from the multiple channels, trains, and functions available to actuate FBACS. If one or two channels or trains in one or more functions are inoperable, the Required Actions require one or both FBACS trains to be placed in the emergency radiation protection mode of operation. This accomplishes the actuation instrumentation function and places the unit in a conservative mode of operation. In the unlikely event that this is not accomplished and Condition C is entered, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for event mitigation. The system design maintains sufficient defense-in-depth when the endstate is changed from Mode 5 to Mode 4.

6.4.4 Technical Specification 3.4.13 – RCS Operational Leakage

Description

Verifying RCS leakage to be within the LCO limits ensures that the integrity of the reactor coolant pressure boundary is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage.

Limiting Condition for Operation

RCS operational leakage shall be limited to:

a. No pressure boundary leakage,
b. 1 gpm unidentified leakage,
c. 10 gpm identified leakage,
d. 1 gpm total primary to secondary leakage through all steam generators (SGs), and
e. [500] gallons per day primary to secondary leakage through any one SG.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met, or pressure boundary leakage exists.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met, or pressure boundary leakage exists.

Basis for Proposed Change

A RCS leakage that is not large enough to be considered a small LOCA would typically be classified as an event leading to a controlled shutdown. Controlled shutdowns are not included in the risk models described in Section 6.3.1, therefore a qualitative evaluation is performed for this proposed endstate change.

RCS leakage can be reduced to lower amounts in Mode 5 compared to Mode 4 because of the lower RCS pressure in Mode 5, however, the RCS pressure in Mode 4 is already significantly lower than at power which will reduce the effects of the RCS leakage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

In Mode 4, the RCS pressure is significantly reduced which reduces the leakage. All LOCA mitigating systems with the exception of the accumulators are available and RHR serves as the backup to auxiliary feedwater for decay heat removal. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.5 Technical Specification 3.4.14 – RCS Pressure Isolation Valve (PIV) Leakage

Description

RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary, that separate the high pressure RCS from an attached low pressure system. The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified leakage, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified leakage before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational leakage if the other is leak-tight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components.

Limiting Condition for Operation

Leakage from each RCS PIV shall be within limit.

Applicability

Modes 1, 2, and 3, and Mode 4, except valves in the RHR flow path when in, or during the transition to or from, the RHR mode of operation.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met.

Basis for Proposed Change

PIV leakage would not be considered a PRA initiating event and would be classified as an event leading to a controlled shutdown. Controlled shutdowns are not included in the risk models described in Section 6.3.1, therefore a qualitative evaluation is performed for this proposed endstate change.

This Technical Specification limits leakage primarily because of the concern of overpressurizing a lower pressure system that can lead to an interfacing system LOCA. PIV leakage can be reduced to a lower level in Mode 5 compared to Mode 4 because of the lower RCS pressure in Mode 5, however, the RCS pressure in Mode 4 is already significantly lower than at power which will reduce the effects of the PIV leakage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

In Mode 4, the RCS pressure is significantly reduced which reduces the PIV leakage. All LOCA mitigating systems with the exception of the accumulators are available and RHR serves as the backup to auxiliary feedwater for decay heat removal. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.6 Technical Specification 3.4.15 – RCS Leakage Detection Instrumentation

Description

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified leakage.

Limiting Condition for Operation

The following RCS leakage detection instrumentation shall be operable:

a. One containment sump (level or discharge flow) monitor,
b. One containment atmosphere radioactivity monitor (gaseous or particulate), and
c. [One containment air cooler condensate flow rate monitor.]

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action E.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action E.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The RCS leakage detection functions; containment sump monitor, containment atmosphere radioactivity monitor, and containment air cooler condensate flow, are not modeled in the risk models described in Section 6.3.1. These functions are not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one function is declared inoperable, the other functions are available to provide indication of RCS leakage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one function is inoperable, the other functions are available to provide indication of RCS leakage. In the unlikely event that Condition E occurs, the likelihood of an initiating event is not increased and placing the unit in Mode 5 does not increase the instrumentation available for detecting RCS leakage. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.7 Technical Specification 3.5.3 – ECCS – Shutdown

Description

This Technical Specification is only applicable in Mode 4. In Mode 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and RHR (low head). The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank can be injected into the RCS if required following an accident.

Limiting Condition for Operation

One ECCS train shall be operable.

Applicability

Mode 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of [Condition B] not met.

Current Required Action Endstate

The current endstate for Required Action C.1 is Mode 5. Specifically, the unit must be in Mode 5 in 24 hours.

Proposed Required Action and Endstate

Condition A is revised from "Required ECCS residual heat removal (RHR) subsystem inoperable." to "Required ECCS train inoperable." Required Action A.1 is revised from "Initiate action to restore required ECCS RHR subsystem to operable status." to "Initiate action to restore required ECCS RHR train to operable status." This change allows the unit to remain in Mode 4, rather than transitioning to Mode 5 with an inoperable ECCS high head subsystem.

Basis for Proposed Change

This Technical Specification is only applicable in Mode 4. There are two subsystems addressed by this Technical Specification; the ECCS RHR subsystem and the ECCS high head subsystem. Both subsystems are included in the risk models described in Section 6.3.1, therefore, a quantitative evaluation is performed.

Current Condition A addresses both RHR trains inoperable and Required Action A.1 requires that action be initiated to restore the required RHR subsystem to operable status with an immediate Completion Time. Required Action A.1 and the immediate Completion Time acknowledge that in this condition it is inappropriate to require the unit to be placed in a Mode where the only means of decay heat removal is not available, rather than to remain in a Mode where steam generator cooling is also available for decay heat removal. Therefore, the change in endstate to evaluate applies to an inoperable high head subsystem, for which a transition to Mode 5 is currently required by Required Action C.1 if it is not returned to operable status within the Completion Time.

To model the inoperability of both train of ECCS high head, the three charging pumps are modeled as inoperable. Only POS 3 and POS 4 are quantified because this Technical Specification is only applicable in Mode 4. The resulting CDPs for each POS are presented in Table 6-11. ~~Also provided are the CDPs from the base case.~~

INSERT E

Table 6-11 Technical Specification 3.5.3 ECCS – Shutdown		
	Core Damage Probability	
POS	One Train Inoperable	Base Case
1	NA	NA
2	NA	NA
3	2.39E-06	7.09E-08
4	9.38E-05	7.21E-06
5	4.79E-07	4.79E-07
6	NA	NA
7	NA	NA
TOTAL	9.66E-05	7.76E-06
TOTAL Excluding POS 4	2.86E-06	5.50E-07

The unavailability of a complete train of ECCS results in an increase in the CDP for both POS 3 and 4. When comparing the base case to the inoperable ECCS train case, the POS 3 CDP increased by a larger factor than the POS 4 CDP, however, the POS 4 CDP for the inoperable ECCS train is approximately 7 ~~40~~ times greater than the POS 3 CDP. Proceeding to Mode 5 does not increase the protection available and additional risk is introduced by switching from AFW cooling to RHR cooling. This case supports remaining in Mode 4 (POS 3) for this configuration rather than cooling down to Mode 5 (POS 4).

INSERT E

Table 6-11 Technical Specification 3.5.3 ECCS – Shutdown		
POS	**Core Damage Probability**	
	Cooldown to POS 3	**Cooldown to POS 4**
1	NA	NA
2	NA	NA
3	1.27E-05	2.39E-06
4	NA	9.23E-05
5	4.79E-07	4.79E-07
6	NA	NA
7	NA	NA
TOTAL	1.32E-05	9.52E-05

Defense-in-Depth Considerations

The proposed change to the Required Action C.1 endstate does not change the operability requirement for the ECCS. One train still must be operable in Mode 4. If one train of RHR is inoperable, then remaining in Mode 4 provides core cooling from the AFW pumps with the operable RHR pump as a backup. If both trains of RHR are inoperable, then the unit will remain on AFW cooling while one train is restored. The probability of transients occurring that require the ECCS are less likely in Mode 4 than at-power and the risk associated with transferring to RHR cooling from AFW cooling is eliminated by remaining in Mode 4. Sufficient defense-in-depth is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5.

6.4.8 Technical Specification 3.5.4 – Refueling Water Storage Tank (RWST)

Description

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

Limiting Condition for Operation

The RWST shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The RWST is included in the risk models described in Section 6.3.1, therefore, a quantitative assessment is made for changing the endstate. Because safety injection is dependent on the RWST for the source of borated water, its inoperability is expected to increase the core damage probabilities above the base case values. The RWST was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-12. ~~Also provided are the CDPs from the base case.~~ e

INSERT F

Table 6-12 Technical Specification 3.5.4, RWST		
	Core Damage Probability	
POS	RWST Inoperable	Base Case
1	1.00E-05	2.18E-07
2	3.36E-06	1.66E-07
3	2.36E-06	7.09E-08
4	9.31E-05	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.14E-04	1.27E-05
Total Excluding POS 4	2.08E-05	5.49E-06

With the RWST unavailable, safety injection and recirculation are not possible. Therefore, any loss of inventory events that cannot be isolated lead to core damage. For the inoperability of the RWST, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4, i.e., the upper portion of Mode 5) reduces the total core damage probability by more than a factor of ~~6~~ 3. The primary accidents that rely on the RWST are the LOCAs and steam line breaks. These accidents are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. In Mode 4, the control rods are inserted and the typical steamline break limiting assumption of the highest worth stuck rod is an unlikely scenario. In addition, the emergency boration system is likely to be available. In Mode 4, transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. Proceeding to Mode 5 does not increase the protection available and additional risk is introduced by switching from AFW cooling to RHR cooling. Variations in boron concentration are likely to be small, therefore, a shutdown to Mode 4 instead of Mode 5 is also appropriate. The RWST temperature variations are also expected to be small because the volume of the tank is large. The design basis accidents that conservatively use the RWST temperature are analyzed at power operation. Therefore, a shutdown to Mode 4 is also appropriate. Based on the risk results in Table 6-12 and the above discussion, if the RWST is inoperable for reasons other than boron concentration or temperature, a shutdown to Mode 4 is appropriate.

Defense-in-Depth Considerations

~~The proposed change to the Required Action C.2 endstate does not change the operability requirement for the RWST. It still must be operable in Mode 4.~~ e a In Mode 4, the transient conditions are less severe than at power so that variations in the RWST parameters or other reasons of inoperability are less significant. In addition, if the boron concentration is low, the emergency boration equipment is likely to be available to

INSERT F

Table 6-12	Technical Specification 3.5.4, RWST	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	1.00E-05	1.00E-05
2	3.36E-06	3.36E-06
3	1.24E-05	2.36E-06
4	NA	9.23E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	3.08E-05	1.13E-04

increase the RCS boron concentration. By changing the endstate for Required Action C.2 to Mode 4, the possibility of having a loss of inventory event due to switching to RHR cooling is eliminated, reducing the possibility that the RWST inventory would be required. Sufficient defense-in-depth is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5.

6.4.9 Technical Specification 3.6.1 – Containment (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

Description

The containment consists of the concrete reactor building, its steel liner, (or a free standing steel pressure vessel surrounded by a reinforced concrete shield building) and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions. The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier.

Limiting Condition for Operation

The containment shall be operable.

Applicability

Modes 1, 2, 3, and 4.

DELETE

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The containment Technical Specification has no direct impact on CDP. Containment integrity is not modeled in the risk models described in Section 6.3.1, therefore, a qualitative evaluation is performed for this proposed endstate change.

Significant leakage from containment that would result in a loss of sump inventory, fail recirculation cooling, and lead to core damage is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and the reduced likelihood of a LOCA or secondary side break due to the limited time in the shutdown modes and less severe thermal-hydraulic conditions. In Level 2 PRA models, containment leakage is not considered to contribute to LERF. A unit shutdown to Mode 5 requires switching to RHR cooling which introduces the potential for increased risks including LOCAs both inside and outside containment. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The proposed change to the Required Action B.2 endstate does not change the operability requirement for containment. The containment must still be operable in Mode 4. In Mode 4 the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are required to be operable. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.10 Technical Specification 3.6.2 – Containment Air Locks (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

Description

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all modes of operation. Each air lock is nominally a right circular cylinder, 10 ft in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a design basis accident in containment. As such, closure of a single door supports containment operability.

Limiting Condition for Operation

[Two] containment air lock[s] shall be operable.

DELETE

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

DELETE

Basis for Proposed Change

The air locks function to maintain containment integrity, therefore, the discussion for Technical Specification 3.6.1 also applies to this Technical Specification. The containment air lock Technical Specification has no direct impact on CDP. Containment integrity is not modeled in the risk models described in Section 6.3.1 so a qualitative evaluation is performed for this proposed endstate change.

Significant leakage from containment that would result in a loss of sump inventory, fail recirculation cooling, and lead to core damage, is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and the reduced likelihood of a LOCA or secondary side break due to the limited time in the shutdown modes and less severe thermal-hydraulic conditions. In addition, closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events at power. In Level 2 PRA models, containment leakage is not considered to contribute to LERF. A unit shutdown to Mode 5 requires switching to RHR cooling which introduces the potential for increased risks including LOCAs both inside and outside containment. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The proposed change to the Required Action D.2 endstate does not change the operability requirement for the containment air locks. The air locks must still be operable in Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are required to be operable. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.11 Technical Specification 3.6.3 – Containment Isolation Valves (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

Description

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

Limiting Condition for Operation

Each containment isolation valve shall be operable.

Applicability

Modes 1, 2, 3, and 4.

DELETE

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action F.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action F.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

A quantitative evaluation of containment isolation valves would be limited to changes in CDP because the risk models described in Section 6.3.1 do not include LERF branches. LERP impacts, not changes to CDP, are the primary concern for the containment isolation valves, therefore, a qualitative approach is taken to evaluate this Technical Specification.

Initiating events during the shutdown process are less severe for containment than those at-power because of the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and the reduced likelihood of a LOCA or secondary side break due to the limited time in the shutdown modes and less severe thermal-hydraulic conditions. Some of the containment penetration lines have a small enough diameter such that they would not contribute to LERF even if all isolation capability for the line is inoperable. A unit shutdown to Mode 5 requires switching to RHR cooling which introduces the potential for increased risks including LOCAs both inside and outside containment. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

DELETE

The proposed change to the Required Action F.2 endstate does not change the operability requirement for the containment isolation valves. The valves must still be operable in Mode 4. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Most containment penetration lines have two isolation valves and it is unlikely that both would be inoperable. In the unlikely event that the actions cannot be completed in time and Condition F is entered, placing the unit in Mode 5 does not increase the equipment available for event mitigation. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are required to be operable. In addition, some of the containment penetration lines have a small enough diameter such that their contribution to LERP would be insignificant. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.12 Technical Specification 3.6.4A – Containment Pressure (Atmospheric, Dual, and Ice Condenser)

Description

Containment pressure is a process variable that is monitored and controlled. The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a LOCA or steam line break. These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the containment spray system.

DELETE

Limiting Condition for Operation

Containment pressure shall be ≥ [-0.3] psig and ≤ [+1.5] psig.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

DELETE

Containment pressure is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment pressure. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The upper containment pressure limit is based on the Mode 1 design basis analyses. These analyses verify that the containment design pressure is not exceeded for a double-ended guillotine break of either the RCS or main steam piping. The containment design pressure is typically a factor of 2 or more below the containment failure pressure. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, containment loadings will be less than in Mode 1 and well below the design pressure and there will be significant margin to the failure pressure. Variations in containment pressure are expected to be small, therefore, any increase above the Technical Specification limit is expected to be small and it is concluded that there will still be sufficient margin to the design basis pressure and significant margin to the failure pressure.

The minimum Technical Specification containment pressure is established such that if there was an inadvertent actuation of the containment spray system, the minimum (negative) containment design pressure would not be exceeded. Inadvertent actuation of the containment spray system does not lead to core damage and LERF by itself, another event needs to occur to cause the core damage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

Defense-in-depth is maintained by the margin to containment failure in Mode 4. The containment pressure limit is based on Mode 1 design basis analyses that include higher energy releases than would occur in Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are available ~~required to be operable.~~ In addition, containment vacuum relief valves and the containment purge system could be used to mitigate containment pressure being outside of the Technical Specification limits. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.13 Technical Specification 3.6.4B – Containment Pressure (Subatmospheric)

Description

Containment air partial pressure is a process variable that is monitored and controlled. The containment air partial pressure is maintained as a function of refueling water storage tank temperature and service water temperature to ensure that, following a design basis accident, the containment would depressurize in less than 60 minutes to subatmospheric conditions. Controlling containment partial pressure within prescribed limits also prevents the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of an inadvertent actuation of the quench spray system.

Limiting Condition for Operation

DELETE

Containment air partial pressure shall be ≥ [9.0] psia and within the acceptable operation range shown on Figure 3.6.4B-1.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

Containment air partial pressure is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air partial pressure. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The upper containment pressure limit is based on the Mode 1 design basis analyses. These analyses verify that the containment design pressure is not exceeded for a double-ended guillotine break of either the RCS or main steam piping. The containment design pressure is typically a factor of 2 or more below the containment failure pressure. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, containment loadings will be less than in Mode 1 and well below the design pressure and there will be significant margin to the failure pressure. Variations in containment pressure are expected to be small, therefore, any increase above the Technical Specification limit is expected to be small and it is concluded that there will still be sufficient margin to the design basis pressure and significant margin to the failure pressure.

The minimum Technical Specification containment pressure is established such that if there was an inadvertent actuation of the quench spray system, the minimum (negative) containment design pressure would not be exceeded. Inadvertent actuation of the quench spray system does not lead to core damage and LERF by itself, another event needs to occur to cause the core damage. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

DELETE

Defense-in-depth is maintained by the margin to containment failure in Mode 4. The containment pressure limit is based on Mode 1 design basis analyses that include higher energy releases than would occur in Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., quench spray, containment cooling) are available ~~required to be operable.~~ Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.14 Technical Specification 3.6.5A – Containment Air Temperature (Atmospheric and Dual)

Description

The containment structure serves to contain radioactive material that may be released from the reactor core following a design basis accident. The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for LOCA or steam line break. The higher the initial temperature, the more energy that must be removed, resulting in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis.

Limiting Condition for Operation

Containment average air temperature shall be ≤ [120]°F.

Applicability

DELETE

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

Containment air temperature is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air temperature. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The containment air temperature limit is based on the Mode 1 design basis analyses and containment equipment qualification requirements. The containment air temperature may exceed the design limit for a short period of time, however, the equipment surface temperatures remain below the design limit. The containment air temperature is also an important initial assumption for calculating the peak containment pressure during a LOCA or main steam line break. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, the containment temperature will be well below the design temperature and there will be significant margin to the design temperature. In the shutdown modes, the containment air temperature is not expected to be high because of lower RCS and steam generators temperatures. Variations in containment air temperature are expected to be small, therefore, any change that exceeds the Technical Specification limit is expected to be small and it is concluded that there will still be sufficient margin to the design basis temperature. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

DELETE

Defense-in-Depth Considerations

Defense-in-depth is maintained by the margin to the containment design air temperature limit that is available in Mode 4. The containment design air temperature limit is based on the Mode 1 design basis analyses that include higher energy releases than would occur for Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray, containment cooling) are available ~~required to be operable~~? Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.15 Technical Specification 3.6.5B – Containment Air Temperature (Ice Condenser)

Description

The containment structure serves to contain radioactive material that may be released from the reactor core following a design basis accident. The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for LOCA or steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used.

Limiting Condition for Operation

Containment average air temperature shall be ≥ [85]°F and ≤ [110]°F for the containment upper compartment and ≥ [100]°F and ≤ [120]°F for the containment lower compartment.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

DELETE

Containment air temperature is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air temperature. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The containment temperature limits are based on the Mode 1 design basis analyses and containment equipment qualification requirements. The containment air temperature may exceed the design limit for a short period of time, however, the equipment surface temperatures remain below the design limit. The containment temperature is also an important initial assumption for calculating the peak containment pressure during a LOCA or main steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, the containment temperature will be well below the design temperature and there will be significant margin to the design temperature. In the shutdown modes, the containment temperature is not expected to be high because of lower RCS and steam generators temperatures. Variations in containment temperature are expected to be small, therefore, any change that falls outside of the Technical Specification limits is expected to be small and it is concluded that there will still be sufficient margin to the design basis temperature limit. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

Defense-in-depth is maintained by the margin to the containment design air temperature limit that is available in Mode 4. The containment air temperature limit is based on the Mode 1 design basis analyses that include higher energy releases than would occur for Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., containment spray) are available ~~required to be operable.~~ Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.16 Technical Specification 3.6.5C – Containment Air Temperature (Subatmospheric)

Description

The containment structure serves to contain radioactive material that may be released from the reactor core following a design basis accident. The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for LOCA or steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used.

Limiting Condition for Operation

Containment average air temperature shall be ≥ [86]°F and ≤ [120]°F.

Applicability

DELETE

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

Containment air temperature is used as an input to PRA model success criteria analyses, but it is not modeled in Westinghouse NSSS plant PRA models. The risk models described in Section 6.3.1 do not include containment air temperature. Therefore, a qualitative evaluation is performed for this proposed endstate change.

The containment temperature limits are based on the Mode 1 design basis analyses and containment equipment qualification requirements. The containment air temperature may exceed the design limit for a short period of time, however, the equipment surface temperatures remain below the design limit. The containment temperature is also an important initial assumption for calculating the peak containment pressure during a LOCA or main steam line break. Depending on the design basis analysis, either the maximum or minimum temperature is used. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Consequently, the containment temperature will be well below the design temperature and there will be significant margin to the design temperature. In the shutdown modes, the containment temperature is not expected to be high because of lower RCS and steam generators temperatures. Variations in containment temperature are expected to be small, therefore, any change that falls outside of the Technical Specification limits is expected to be small and it is concluded that there will still be sufficient margin to the design basis temperature limit. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

DELETE

Defense-in-depth is maintained by the margin to the containment design air temperature limit that is available in Mode 4. The containment air temperature limit is based on the Mode 1 design basis analyses that include higher energy releases than would occur for Mode 4. In Mode 4, the systems designed to mitigate the effects of accidents on the containment (e.g., quench spray, recirculation spray) are ~~required~~ available ~~to be operable~~. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.17 Technical Specification 3.6.6A – Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

Description

The containment spray and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits.

The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of SW. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units are operating. The fans are normally operated at high speed with SW supplied to the cooling coils. In post accident operation following an actuation signal, the containment cooling system fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere.

Limiting Condition for Operation

Two containment spray trains and [two] containment cooling trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met, and Required Action and associated Completion Time of Condition C or D not met.

Current Required Action Endstate

The current endstate for Required Actions B.2 and E.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours for Condition B, and the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours for Condition E.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 60 hours, and revise the endstate for Required Action E.2 to be in Mode 4 in 12 hours.

Basis for Proposed Change

The containment spray system and containment cooling units are modeled for a few sequences in the risk models described in Section 6.3.1. In the fault tree models these systems provide backup cooling for recirculation. Their impact on CDF is minimal. The main impact of the inoperability of the containment spray and containment cooling units is in the containment response in the Level 2 analysis, which is not included in the risk models. For POS 4, neither system is credited for providing a backup cooling function. Note that POS 4 includes the upper portion of Mode 5 and these systems are not required to be operable.

Technical Specification 3.6.6A, Actions A, C, and D address combinations of inoperable trains of containment spray and containment cooling units. Technical Specification 3.6.6B, Actions A through E also address combinations of inoperable trains of containment spray and containment cooling units. The risk models described in Section 6.3.1 are used to model the combinations of inoperable equipment and determine the resulting CDP for POS 1, 2, and 3 for these two Specifications. The containment spray system and containment cooling system are not modeled for POS 4. Table 6-13 presents the combinations of inoperable equipment, the applicable Technical Specification and Action, and the resulting CDPs. ~~Also provided are the CDPs from the base case.~~

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Table 6-13	Technical Specifications 3.6.6A and 3.6.6B Containment Spray and Containment Cooling Systems					
	Core Damage Probability					
POS	One Train Containment Spray Inoperable[1]	Two Trains Containment Spray Inoperable[2]	One Train Containment Cooling Units Inoperable[3]	Two Trains Containment Cooling Units Inoperable[4]	One Train Containment Spray, One Train Containment Cooling Units Inoperable[5]	Base Case
1	2.18E-07	2.18E-07	2.22E-07	3.98E-07	2.22E-07	2.18E-07
2	1.66E-07	1.66E-07	1.67E-07	2.11E-07	1.67E-07	1.66E-07
3	7.09E-08	7.09E-08	7.13E-08	9.14E-08	7.13E-08	7.09E-08
4	7.21E-06	7.21E-06	7.21E-06	7.21E-06	7.21E-06	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	1.27E-05	1.27E-05	1.27E-05	1.29E-05	1.27E-05	1.27E-05
Total Excluding POS 4	5.49E-06	5.49E-06	5.50E-06	5.74E-06	5.50E-06	5.49E-06

Notes:
1. Technical Specifications 3.6.6A and 3.6.6B, Action A.
2. Technical Specification 3.6.6B, Action C.
3. Technical Specification 3.6.6A, Action C and Technical Specification 3.6.6B, Action B.
4. Technical Specification 3.6.6A, Action D and Technical Specification 3.6.6B, Action E.
5. Technical Specification 3.6.6B, Action D.

The results confirm that these two systems have little effect on the calculated CDPs from the base case. For containment spray, the results are the same as the base results. For the containment cooling units, the results are not significantly different. The conclusion for all five cases is that containment spray and containment cooling do not significantly affect the shutdown modes CDP and there is a risk increase by cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

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Table 6-13 Technical Specifications 3.6.6A and 3.6.6B Containment Spray and Containment Cooling Systems

POS	Core Damage Probability									
	One Train Containment Spray Unavailable[1]		Two Trains Containment Spray Unavailable[2]		One Train Containment Cooling Units Unavailable[3]		Two Trains Containment Cooling Units Unavailable[4]		One Train Containment Spray, One Train Containment Cooling Units Unavailable[5]	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	2.18E-07	2.18E-07	2.18E-07	2.18E-07	2.22E-07	2.22E-07	3.98E-07	3.98E-07	2.22E-07	2.22E-07
2	1.66E-07	1.66E-07	1.66E-07	1.66E-07	1.67E-07	1.67E-07	2.11E-07	2.11E-07	1.67E-07	1.67E-07
3	5.95E-07	7.09E-08	5.95E-07	7.09E-08	5.98E-07	7.13E-08	7.11E-07	9.14E-08	5.98E-07	7.13E-08
4	NA	4.03E-06	NA	4.03E-06	NA	4.03E-06	NA	4.03E-06	NA	4.03E-06
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	6.02E-06	9.52E-06	6.02E-06	9.52E-06	6.03E-06	9.53E-06	6.36E-06	9.77E-06	6.03E-06	9.53E-06

Notes:

1. Technical Specifications 3.6.6A and 3.6.6B, Action A.
2. Technical Specification 3.6.6B, Action C.
3. Technical Specification 3.6.6A, Action C and Technical Specification 3.6.6B, Action B.
4. Technical Specification 3.6.6A, Action D and Technical Specification 3.6.6B, Action E.
5. Technical Specification 3.6.6B, Action D.

Defense-in-Depth Considerations

The containment spray and containment cooling systems are designed for accident conditions initiated at power. One train of each system satisfies the assumptions in the safety analyses and one train of containment spray is required to satisfy assumptions regarding iodine removal. If one train of either containment spray or containment cooling is inoperable the other train is available to mitigate the accident along with both trains of the other system. If both trains of containment cooling are inoperable, containment spray can serve as the cooling system and it also serves to remove iodine. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.18 Technical Specification 3.6.6B – Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit not taken for iodine removal by the Containment Spray System)

Description

The containment spray and containment cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure reduces the release of fission product radioactivity from containment to the environment, in the event of a design basis accident, to within limits.

The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump(s).

Two trains of containment cooling, each of sufficient capacity to supply 100% of the design cooling requirement, are provided. Each train of two fan units is supplied with cooling water from a separate train of SW. Air is drawn into the coolers through the fan and discharged to the steam generator compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment. During normal operation, all four fan units are operating. The fans are normally operated at high speed with SW supplied to the cooling coils. In post accident operation following an actuation signal, the containment cooling system fans are designed to start automatically in slow speed if not already running. If running in high (normal) speed, the fans automatically shift to slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher mass atmosphere.

Limiting Condition for Operation

Two containment spray trains and [two] containment cooling trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A, B, C, D, or E not met.

Current Required Action Endstate

The current endstate for Required Action F.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action F.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A, B, C, D, or E not met.

Basis for Proposed Change

This Technical Specification is very similar to Technical Specification 3.6.6A. However, because no credit is taken for iodine removal, a Required Action is provided to restore two inoperable trains of containment spray. If two trains of containment spray are inoperable, the containment cooling units are still available to provide containment cooling. The cases presented in Table 6-13 demonstrate that the containment spray case results are the same as the base results. For the containment cooling units, the results are not significantly different. The conclusion for all five cases is that containment spray and containment cooling do not significantly affect the shutdown modes CDP and there is a risk increase by cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The containment spray and containment cooling systems are designed for accident conditions initiated at power. One train of each system satisfies the assumptions in the safety analyses. If one train of either containment spray or containment cooling is inoperable the other train is available to mitigate the accident conditions along with both trains of the other system. If both trains of one system are unavailable, the two trains of the other system are available to provide containment cooling. Note that this Technical Specification does not take credit for iodine removal by the containment spray system. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.19 Technical Specification 3.6.6C – Containment Spray System (Ice Condenser)

Description

The containment spray system provides containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduce the release of fission product radioactivity from containment to the environment, in the event of a design basis accident.

Each train includes a containment spray pump, one containment spray heat exchanger, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment recirculation sump(s).

The diversion of a portion of the recirculation flow from each train of RHR to additional redundant spray headers completes the containment spray system heat removal capability. Each RHR train is capable of supplying spray coverage, if required, to supplement the containment spray system. The RHR spray operation is initiated manually, when required by the emergency operating procedures, after the ECCS is operating in the recirculation mode.

Limiting Condition for Operation

Two containment spray trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 60 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

If one train of containment spray is inoperable, the other train is still available to provide accident mitigation. The inoperability of one train of containment spray would not significantly affect the shutdown modes CDP and there is risk associated with cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The containment spray system is designed for accident conditions initiated at power. One train satisfies the assumptions in the safety analyses. In addition, the containment ice condenser is ~~required to be operable~~ available and it is designed to handle a heat load in excess of the initial blowdown of a design basis LOCA, or any feedwater or steamline break event inside containment. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. In addition, RHR spray could be used if necessary for continued containment cooling. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.20 Technical Specification 3.6.6D – Quench Spray (QS) System (Subatmospheric)

Description

The quench spray system is designed to provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. The QS system, operating in conjunction with the recirculation spray (RS) system, is designed to cool and depressurize the containment structure to subatmospheric pressure in less than 60 minutes following a design basis accident. Reduction of containment pressure and the iodine removal capability of the spray limit the release of fission product radioactivity from containment to the environment in the event of a design basis accident.

The QS System consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The RWST supplies borated water to the QS system. The QS system is actuated either automatically by a containment High-High pressure signal or manually. Each train of the QS system provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal.

Limiting Condition for Operation

Two QS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

If one train of quench spray is inoperable, the other train is still available to provide accident mitigation. The inoperability of one train of quench spray would not significantly affect the shutdown modes CDP and there is risk associated with cooling down to Mode 5 (POS 4). A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, and there is increased time for operator actions and mitigation strategies. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The quench spray system is designed for accident conditions initiated at power. One train satisfies the assumptions in the safety analyses. In addition, the containment temperature and pressure limits are set to account for the effects of an energy release during an event in full power operation. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.21 Technical Specification 3.6.6E – Recirculation Spray (RS) System (Subatmospheric)

Description

The recirculation spray system, operating in conjunction with the quench spray system, is designed to limit the post accident pressure and temperature in the containment to less than the design values and to depressurize the containment structure to a subatmospheric pressure in less than 60 minutes following a design basis accident. The reduction of containment pressure and the removal of iodine from the containment atmosphere by the spray limit the release of fission product radioactivity from containment to the environment in the event of a design basis accident.

The RS system consists of two separate trains of equal capacity, each capable of meeting the design and accident analysis bases. Each train includes one RS subsystem outside containment and one RS subsystem inside containment. Each subsystem consists of one 50% capacity spray pump, one spray cooler, one 180° coverage spray header, nozzles, valves, piping, instrumentation, and controls. Each outside RS subsystem also includes a casing cooling pump with its own valves, piping, instrumentation, and controls. The two outside RS subsystems' spray pumps are located outside containment and the two inside RS subsystems' spray pumps are located inside containment. Each RS train (one inside and one outside RS subsystem) is powered from a separate ESF bus. Each train of the RS system provides adequate spray coverage to meet the system design requirements for containment heat and iodine fission product removal.

Limiting Condition for Operation

Four RS subsystems [and a casing cooling tank] shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action F.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action F.2 to be in Mode 4 in 60 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include recirculation spray because recirculation spray is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If any of Technical Specification 3.6.6E Conditions A through E are entered, the recirculation spray system can still perform its safety function. Note that if the casing cooling tank is inoperable, the net positive suction head available to the outside RS subsystem pumps may not be sufficient. This situation is the same as Condition D. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur. This evaluation does not take credit for location differences of the subsystems and is, therefore, applicable to similar system configurations that have all of the pumps and heat exchangers located either inside or outside of the containment.

Defense-in-Depth Considerations

The recirculation spray system is designed for accident conditions initiated at power. One train (two subsystems) satisfies the assumptions in the safety analyses. In addition, the containment temperature and pressure limits are set to account for the effects of an energy release during an event in full power operation. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.22 Technical Specification 3.6.7 – Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

Description

The spray additive system is a subsystem of the containment spray system that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a design basis accident.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a design basis accident. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms.

For an eductor feed system, the spray additive system consists of one spray additive tank that is shared by the two trains of spray additive equipment. Each train of equipment provides a flow path from the spray additive tank to a containment spray pump and consists of an eductor for each containment spray pump, valves, instrumentation, and connecting piping. Each eductor draws the NaOH spray solution from the common tank using a portion of the borated water discharged by the containment spray pump as the motive flow. The eductor mixes the NaOH solution and the borated water and discharges the mixture into the spray pump suction line.

For a gravity feed system, the spray additive system consists of one spray additive tank, two parallel redundant motor operated valves in the line between the additive tank and the RWST, instrumentation, and recirculation pumps. The NaOH solution is added to the spray water by a balanced gravity feed from the additive tank through the connecting piping into a weir within the RWST. There, it mixes with the borated water flowing to the spray pump suction.

Limiting Condition for Operation

The spray additive system shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 84 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 60 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the spray additive system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. The spray additive system assists in reducing the iodine fission product inventory. Containment spray by itself removes some iodine from the containment atmosphere, so iodine removal will still occur with an inoperable spray additive system. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The spray additive system is designed for accident conditions initiated at power. The containment spray system will remove some iodine from the containment atmosphere without the additive system and two trains of containment spray are required to be operable. The spray additive system also serves to provide the proper pH in the containment sump. For most containments, a backup system for containment sump pH is not available, but proceeding to Mode 5 does not increase the protection available. Note that the ice condenser containments have ice that is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere and minimizes the occurrence of the chloride and caustic stress corrosion on mechanical systems and components. Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.22a Recirculation Fluid pH Control System

Some Westinghouse NSSS plants have replaced the spray additive system with a passive ECCS recirculation fluid pH control system. Although the Technical Specification for this system is not contained in NUREG-1431, the endstate is Mode 5 if the system is inoperable, and the Required Action and associated Completion Time are not met. The system consists of baskets in the containment sump with a specified amount of trisodium phosphate in each basket. The trisodium phosphate dissolves when the containment sump level increases to the level of the baskets.

It is highly unlikely that all of the baskets would be empty, therefore, an inoperable recirculation fluid pH control system would still provide some pH control. The justification for changing the endstate to Mode 4 for Technical Specification 3.6.7, "Spray Additive System," is also applicable to the recirculation fluid pH control system, since they perform the same function.

The recirculation fluid pH control system Technical Specification currently requires the unit to be in Mode 3 in 6 hours and Mode 5 in 84 hours if the system is inoperable, and the Required Action and associated Completion Time are not met. The current Mode 5 endstate is proposed to be changed to require the unit to be in Mode 4 in 60 hours if the Required Action and associated Completion Time are not met.

6.4.23 Technical Specification 3.6.8 – Shield Building (Dual and Ice Condenser)

Description

DELETE

The shield building is a concrete structure that surrounds the steel containment vessel. Between the containment vessel and the shield building inner wall is an annulus that collects containment leakage than may occur following a loss of coolant accident, steam line break, or control rod ejection.

Limiting Condition for Operation

The shield building shall be operable.

DELETE

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The shield building Technical Specification has no impact on CDP. Shield building integrity is not modeled in the risk models described in Section 6.3.1, therefore, a qualitative evaluation is performed for this proposed endstate change.

Significant leakage from the containment vessel and the shield building in Mode 4 is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and the reduced likelihood of a LOCA or secondary side break due to reduced thermal-hydraulic conditions. In Level 2 PRA models, shield building and containment vessel leakage is not considered to contribute to LERF. A unit shutdown to Mode 5 requires transferring to RHR cooling which introduces the potential for increased risks including LOCAs both inside and outside containment. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

~~The proposed change to the Required Action B.2 endstate does not change the operability requirement for the shield building. The shield building must still be operable in Mode 4.~~ In Mode 4, two trains of containment spray are ~~required to be operable~~ available to mitigate radioactive releases after an event. Significant leakage from the containment vessel and the shield building is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, the reduced likelihood of a LOCA in the shutdown modes, and reduced thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.24 Technical Specification 3.6.11 – Iodine Cleanup System (ICS) (Atmospheric and Subatmospheric)

Description

The iodine cleanup system functions together with the containment spray and cooling systems following a design basis accident to reduce the potential release of radioactive material, principally iodine, from the containment to the environment.

The iodine cleanup system consists of two 100% capacity, separate, independent, and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, a demister, a HEPA filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. Ductwork, valves and/or dampers, and instrumentation also form part of the system. Each ICS train is powered from a separate ESF bus and is provided with a separate power panel and control panel. During normal operation, the containment cooling system is aligned to bypass the ICS HEPA filters and charcoal adsorbers. For ICS operation following a design basis accident, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

Limiting Condition for Operation

Two ICS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the iodine cleanup system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. The iodine cleanup system assists in reducing the iodine fission product inventory. If one iodine cleanup system train is inoperable, the other is available to perform its function. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

The iodine cleanup system is designed for accident conditions initiated at power. One train of the iodine cleanup system is available and capable of performing its design basis function. In addition, the containment spray system will also remove iodine from the containment atmosphere and two trains of containment spray are available ~~required to be operable.~~ Events, such as a LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.25 Technical Specification 3.6.12 – Vacuum Relief Valves (Atmospheric and Ice Condenser)

Description

The purpose of the vacuum relief lines is to protect the containment vessel against negative pressure (i.e., a lower pressure inside than outside). Excessive negative pressure inside containment can occur if there is an inadvertent actuation of containment cooling features, such as the containment spray system. Multiple equipment failures or human errors are necessary to cause inadvertent actuation of these systems.

The containment pressure vessel contains two 100% vacuum relief lines that protect the containment from excessive external loading.

Limiting Condition for Operation

[Two] vacuum relief lines shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the vacuum relief valves because they are a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. The vacuum relief valves protect the containment from negative pressure due to an inadvertent actuation of the containment spray system. Inadvertent actuation of the containment spray system does not lead directly to core damage and large early releases. Another event needs to occur to cause the core damage. In addition, if one vacuum relief line is inoperable, the other line is available to provide the containment protection. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

Inadvertent actuation of containment spray is unlikely due to the instrumentation design for automatic and manual initiation. Inadvertent actuation of the containment spray system does not lead directly to core damage and large early releases by itself. Another event needs to occur to cause core damage. In addition, if one vacuum relief line is inoperable, the other line is available to provide the containment protection. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.26 Technical Specification 3.6.13 – Shield Building Air Cleanup System (SBACS) (Dual and Ice Condenser)

Description

The containment has a secondary containment called the shield building, that is a concrete structure that surrounds the steel primary containment vessel. Between the containment vessel and the shield building inner wall is an annular space that collects any containment leakage that may occur following a LOCA. This space also allows for periodic inspection of the outer surface of the steel containment vessel.

The SBACS establishes a negative pressure in the annulus between the shield building and the steel containment vessel. Filters in the system then control the release of radioactive contaminants to the environment.

The SBACS consists of two separate and redundant trains. Each train includes a heater, [cooling coils,] a prefilter, moisture separators, a HEPA filter, an activated charcoal adsorber section for removal of radioiodines, and a fan. During normal operation, the shield building cooling system is aligned to bypass the SBACS's HEPA filters and charcoal adsorbers. For SBACS operation following a design basis accident, however, the bypass dampers automatically reposition to draw the air through the filters and adsorbers.

Limiting Condition for Operation

Two SBACS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the SBACS because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one SBACS train is inoperable, the other train is available to provide the annulus air cleanup. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one SBACS train is inoperable, the other train is available to provide the annulus air cleanup. In addition, two trains of containment spray are ~~required to be operable~~ available to mitigate radioactive releases after an event. Significant leakage from containment is highly unlikely due to the significantly reduced RCS temperature and pressure conditions as the unit is being shutdown, and reduced likelihood of a LOCA in the shutdown modes, and less severe thermal-hydraulic conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.27 Technical Specification 3.6.14 – Air Return System (ARS) (Ice Condenser)

Description

The air return system is designed to assure the rapid return of air from the upper to the lower containment compartment after the initial blowdown following a design basis accident. The return of this air to the lower compartment and subsequent recirculation back up through the ice condenser assists in cooling the containment atmosphere and limiting the post accident pressure and temperature in containment to less than design values. The air return system provides post accident hydrogen mixing in selected areas of containment. The air return system also functions, after all the ice has melted, to circulate any steam still entering the lower compartment to the upper compartment where the containment spray system can cool it.

The air return system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a 100% capacity air return fan, associated damper, and hydrogen collection headers with isolation valves. The ARS fans are automatically started and the hydrogen collection header isolation valves are opened by the containment pressure High-High signal 10 minutes after the containment pressure reaches the pressure setpoint. Each train is powered from a separate ESF bus.

Limiting Condition for Operation

Two ARS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the air return system because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one air return train is inoperable, the other train is available to assist in cooling the containment atmosphere. In addition, two trains of containment spray are available ~~required to be operable~~. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one air return train is inoperable, the other train is available to assist in cooling the containment atmosphere. Containment cooling is still available from the containment ice condenser and from two trains of containment spray ~~that are required to be operable~~. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.28 Technical Specification 3.6.15 – Ice Bed (Ice Condenser)

Description

DELETE

The ice absorbs energy and limits containment peak pressure and temperature during a design basis accident. The ice condenser is an annular compartment enclosing approximately 300° of the perimeter of the upper containment compartment, but penetrating the operating deck so that a portion extends into the lower containment compartment. The ice bed consists of over [2,721,600] lb of ice stored in [1944] baskets within the ice condenser. Its primary purpose is to provide a large heat sink in the event of a release of energy from a design basis accident in containment.

The ice baskets contain the ice within the ice condenser. The ice baskets position the ice within the ice bed in an arrangement to promote heat transfer from the steam to the ice. This arrangement enhances the ice condenser's primary function of condensing steam and absorbing heat energy released to the containment during a design basis accident.

In the event of a design basis accident, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condenser limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

The ice, together with the containment spray, is adequate to absorb the initial blowdown of steam and water from a design basis accident and the additional heat loads that would exist in containment during several hours following the initial blowdown. As ice melts, the water passes through the ice condenser floor drains into the lower compartment. Thus, a second function of the ice bed is to serve as a large source of borated water (via the containment sump) for long term ECCS and containment spray system heat removal functions in the recirculation mode.

A third function of the ice bed and melted ice is to remove fission product iodine that may be released from the core during a design basis accident. Iodine removal occurs during the ice melt phase of the accident and continues as the melted ice is sprayed into the containment atmosphere by the containment spray system. The ice is adjusted to an alkaline pH that facilitates removal of radioactive iodine from the containment atmosphere. The alkaline pH also minimizes the occurrence of chloride and caustic stress corrosion on mechanical systems and components exposed to ECCS and containment spray system fluids in the recirculation mode of operation.

Limiting Condition for Operation

The ice bed shall be operable.

DELETE

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the ice bed because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. Two phenomena that can degrade the ice bed during power operation are the loss of ice by melting or sublimation and the obstruction of flow passages through the ice bed due to buildup of frost or ice. Both of these degrading phenomena are reduced by minimizing air leakage into and out of the ice condenser. Due to the very large mass of stored ice, if the ice bed is inoperable, it would still retain cooling capability and the containment spray system and the air return system would also provide cooling for accident conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations DELETE

Due to the very large mass of stored ice, if the ice bed is inoperable, it is highly unlikely that it would have no contribution to containment cooling if an event occurred. In addition, two trains of the containment spray system and two trains of the air return system provide cooling capability for accident conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.29 Technical Specification 3.6.16 – Ice Condenser Doors (Ice Condenser)

Description

The ice condenser doors consist of the inlet doors, the intermediate deck doors, and the top deck doors. The functions of the doors are to seal the ice condenser from air leakage and open in the event of a design basis accident to direct the hot steam air mixture into the ice bed, where the ice would absorb energy and limit containment peak pressure and temperature during the accident transient. The ice baskets contained in the ice bed within the ice condenser are arranged to promote heat transfer from steam to ice.

In the event of a design basis accident, the ice condenser inlet doors (located below the operating deck) open due to the pressure rise in the lower compartment. This allows air and steam to flow from the lower compartment into the ice condenser. The resulting pressure increase within the ice condenser causes the intermediate deck doors and the top deck doors to open, which allows the air to flow out of the ice condenser into the upper compartment. Steam condensation within the ice condensers limits the pressure and temperature buildup in containment. A divider barrier separates the upper and lower compartments and ensures that the steam is directed into the ice condenser.

Limiting Condition for Operation

The ice condenser inlet doors, intermediate deck doors, and top deck [doors] shall be operable and closed.

Applicability

DELETE

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A or C not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A or C not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the ice condenser because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. There are a series of ice condenser doors at each level (lower, intermediate, and upper). It is highly unlikely that a sufficient number of doors would be inoperable to the extent that they would render the ice condenser ineffective for cooling containment during accident conditions. In addition, the containment spray system and the air return system provide cooling capability. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. A cooldown to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one or more ice condenser doors are inoperable such that the ice condenser is degraded for cooling containment during accident conditions, two trains of the containment spray system and two trains of the air return system are ~~required~~ available to ~~be operable and would~~ provide cooling capability. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event that would challenge containment integrity occurring in Mode 4 is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.30 Technical Specification 3.6.17 – Divider Barrier Integrity (Ice Condenser)

DELETE

Description

The divider barrier consists of the operating deck and associated seals, personnel access doors, and equipment hatches that separate the upper and lower containment compartments. Divider barrier integrity is necessary to minimize bypassing of the ice condenser by the hot steam and air mixture released into the lower compartment during a design basis accident. This ensures that most of the gases pass through the ice bed, which condenses the steam and limits pressure and temperature during the accident transient.

Limiting Condition for Operation

Divider barrier integrity shall be maintained.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the divider barrier because it is a containment system and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. It is unlikely that the divider barrier would create a large enough bypass of the ice condenser to render the ice condenser completely ineffective during accident conditions. During accident conditions, the containment spray system and the air return system will also provide cooling capability. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

DELETE

Defense-in-Depth Considerations

It is unlikely that the divider barrier would create a large enough bypass of the ice condenser to render the ice condenser completely ineffective during accident conditions. Two trains of the containment spray system and two trains of the air return system are ~~required to be operable and would~~ available to provide cooling capability during accident conditions. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.31 Technical Specification 3.6.18 – Containment Recirculation Drains (Ice Condenser)

Description

The containment recirculation drains consist of the ice condenser drains and the refueling canal drains. [Twenty of the 24] ice condenser bays have a floor drain at the bottom to drain the melted ice into the lower compartment (in the [4] bays that do not have drains, the water drains through the floor drains in the adjacent bays). A check (flapper) valve at the end of each pipe keeps warm air from entering during normal operation, but when the water exerts pressure, the check valve opens to allow the water to spill into the lower compartment. This prevents water from backing up and interfering with the ice condenser inlet doors. The water delivered to the lower containment serves to cool the atmosphere as it drains to the floor and provides a source of borated water at the containment sump for long term use by the ECCS and the containment spray system during the recirculation mode of operation.

The two refueling canal drains are at low points in the refueling canal. In the event of a design basis accident, the refueling canal drains are the main return path to the lower compartment for containment spray system water sprayed into the upper compartment.

Limiting Condition for Operation

The ice condenser floor drains and the refueling canal drains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The CDP risk models described in Section 6.3.1 do not include the ice condenser and refueling canal drains because they are containment systems and the risk models are based on core damage probability. Therefore, a qualitative evaluation is performed for this proposed endstate change. If one drain is inoperable, there are other drains available to perform the function of transferring water to its intended destination. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one ice condenser floor drain is inoperable, there are [19] others available to drain the water to the lower compartment. If one refueling canal drain is inoperable, there is another refueling canal drain to transfer the containment spray water to the lower compartment. An event in Mode 4 that releases energy into containment will release far less energy than an event in Mode 1. The likelihood of an event occurring in Mode 4 that would challenge containment integrity is reduced along with the consequences because of the significantly reduced RCS temperature and pressure conditions. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.32 Technical Specification 3.7.7 – Component Cooling Water (CCW) System

Description

The component cooling water system provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident. The CCW system serves as a barrier to prevent the release of radioactive byproducts between potentially radioactive systems and the service water system, and then to the environment.

A typical CCW System is arranged as two independent, full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pump in each train is automatically started on receipt of a safety injection signal, and all nonessential components are isolated.

Limiting Condition for Operation

Two CCW trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met.

Basis for Proposed Change

The CCW system is included in the risk models described in Section 6.3.1. The model includes two independent trains with one pump in each train, and a third pump that can be aligned to either train. A quantitative evaluation is performed using the risk models.

Two scenarios are modeled. In the first scenario, CCW Train A, that has two pumps aligned to the train, is assumed to be inoperable. The Train A configuration is conservative for units that have a backup pump in each train. In the second scenario, CCW Train B, that has one pump aligned to the train, is assumed to be inoperable. Both cases are analyzed and the results are presented in Table 6-14. ~~along with the results from the base case.~~

INSERT H

Table 6-14 Technical Specification 3.7.7, CCW

POS	Core Damage Probability: Train A – Two Pumps Inoperable	Train B – One pump Inoperable	Base Case
1	3.29E-06	1.47E-06	2.18E-07
2	1.12E-06	1.22E-06	1.66E-07
3	7.65E-08	7.25E-08	7.09E-08
4	2.98E-05	2.78E-05	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06
Total	3.93E-05	3.56E-05	1.27E-05
Total Excluding POS 4	9.53E-06	7.80E-06	5.49E-06

INSERT H

Table 6-14	Technical Specification 3.7.7, CCW			
POS	Core Damage Probability			
	Train A – Two Pumps Unavailable		Train B – One Pump Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	3.29E-06	3.29E-06	1.47E-06	1.47E-06
2	1.12E-06	1.12E-06	1.22E-06	1.22E-06
3	6.85E-07	7.65E-08	6.20E-07	7.25E-08
4	NA	1.51E-05	NA	1.44E-05
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	1.01E-05	2.46E-05	8.35E-06	2.22E-05

For the inoperability of CCW Train A that includes the swing pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by a factor of ~~4~~. The 2 POS 1 and 2 contributions are relatively high because of the loss of CCW initiating event and resulting RCP seal LOCAs that are modeled in these POSs, and this case assumes two CCW pumps are inoperable.

For the inoperability of CCW Train B that includes one pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of ~~4~~. 2 The CCW Train B case total CDP and the POS 4 CDP are less than the Train A case results because the Train A case models two inoperable CCW pumps. The conclusion for both cases is there is less risk associated with a cool down to Mode 4 than there is with a cool down to Mode 5 when a train of CCW is inoperable.

Defense-in-Depth Considerations

One CCW train will be operating when the unit enters Mode 4, therefore, failures of the pump to start or valves to open are not applicable. Each train is designed to handle 100% of the heat loads during power operation and accident conditions. If the unit design includes a swing pump, it is highly probable that the swing pump would be available to backup the operating pump. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.33 Technical Specification 3.7.8 – Service Water System (SWS)

Description

The service water system provides a heat sink for the removal of process and operating heat from safety related components during a design basis accident. During normal operation, and a normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related function is covered by this LCO.

A typical SWS consists of two separate, 100% capacity, safety related, cooling water trains. Each train consists of two 100% capacity pumps, one component cooling water heat exchanger, piping, valving, instrumentation, and two cyclone separators. The pumps and valves are remote and manually aligned, except in the unlikely event of a LOCA. The pumps aligned to the critical loops are automatically started upon receipt of a safety injection signal, and all essential valves are aligned to their post accident positions. The SWS also provides emergency makeup to the spent fuel pool and CCW system and typically is the backup water supply to the auxiliary feedwater system.

Limiting Condition for Operation

Two SWS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met.

Basis for Proposed Change

The service water system is included in the risk models described in Section 6.3.1. The model includes two independent trains with one pump in each train, and a third pump that can be aligned to either train. A quantitative evaluation is performed using the risk model.

Two scenarios are modeled. In the first scenario, SWS Train A, that has two pumps aligned to the train, is assumed to be inoperable. The Train A configuration is conservative for units that have a backup pump in each train. In the second scenario, SWS Train B, that has one pump aligned to the train, is assumed to be inoperable. Both cases are analyzed and the results are presented in Table 6-15. ~~along with the results from the base case.~~

INSERT 1

Table 6-15 Technical Specification 3.7.8, SWS

POS	Core Damage Probability		
	Train A – Two Pumps Inoperable	Train B – One Pump Inoperable	Base Case
1	1.56E-05	8.68E-07	2.18E-07
2	8.99E-07	3.67E-07	1.66E-07
3	2.06E-07	1.58E-07	7.09E-08
4	4.07E-05	3.43E-05	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06
Total	6.24E-05	4.07E-05	1.27E-05
Total Excluding POS 4	2.17E-05	6.43E-06	5.49E-06

INSERT I

Table 6-15 Technical Specification 3.7.8, SWS

POS	Core Damage Probability			
	Train A – Two Pumps Unavailable		Train B – One Pump Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	1.56E-05	1.56E-05	8.68E-07	8.68E-07
2	8.99E-07	8.99E-07	3.67E-07	3.67E-07
3	2.58E-06	2.06E-07	1.82E-06	1.58E-07
4	NA	2.21E-05	NA	2.07E-05
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	2.41E-05	4.38E-05	8.09E-06	2.71E-05

For the inoperability of SWS Train A that includes the swing pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by almost a factor of ~~3.~~ 2

For the inoperability of SWS Train B that includes one pump, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of ~~6.~~ 3 This results in a greater reduction in CDP than the Train A case, although the POS 4 CDP for the Train A case is greater than the POS 4 CDP for the Train B case. The greater reduction in CDP for the Train B case is because the POS 4 CDP dominates the Train B results, whereas the Train A results have larger CDPs, particularly for POS 1. ~~This decreases the impact of subtracting the POS 4 results for the Train A case.~~ The main conclusion of both cases is the same; a cool down to Mode 4 instead of Mode 5 reduces the risk of the shutdown process when a train of the SWS is inoperable.

<u>Defense-in-Depth Considerations</u>

One SWS train will be operating when the unit enters Mode 4, therefore, failures of the pump to start or valves to open are not applicable. Each train is designed to handle 100% of the heat loads during power operation and accident conditions. If the plant design includes a swing pump or redundant pumps in each train, it is highly probable that another pump would be available to backup the operating pump. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.34 Technical Specification 3.7.9 – Ultimate Heat Sink (UHS)

<u>Description</u>

The ultimate heat sink provides a heat sink for the removal of process and operating heat from safety related components during an accident, as well as during normal operation. This is done by utilizing the service water system and the component cooling water system.

The UHS has been defined as the water sources, including necessary retaining structures (e.g., a pond with its dam, or a river with its dam), and the canals or conduits connecting the sources with, but not including, the cooling water system intake structures as discussed in the FSAR. If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the heat sink. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown, and dissipation of residual heat after an accident.

A variety of water sources are used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the water sources include a water source contained by a structure, it is likely that a second source will be required.

<u>Limiting Condition for Operation</u>

The UHS shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

[Required Action and associated Completion Time of Condition A or B not met, or] UHS inoperable [for reasons other than Condition A or B].

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the [Required Action and associated Completion Time of Condition A or B not met, or] UHS inoperable [for reasons other than Condition A or B].

Basis for Proposed Change

The risk models described in Section 6.3.1 do not include cooling towers because the models are based on a plant that has a cooling pond that supplies cooling water to the service water system. Therefore, a qualitative approach is used for this endstate change.

The Actions of Technical Specification 3.7.9 address degradations to the cooling capability of the ultimate heat sink. Because of the limitations on water temperature and the variety of designs of the ultimate heat sink, the most likely scenario for entering Condition C is that the cooling capability of the ultimate heat sink is only partially degraded. A cool down to Mode 4 places the unit in a state in which the heat loads are significantly less than at full power. There are additional risks associated with a cool down to Mode 5, e.g., switching to RHR cooling, and transferring the heat load to the component cooling water system.

Defense-in-Depth Considerations

The ultimate heat sink is designed to remove 100% of the heat loads generated during power operation and accident conditions. The heat loads will be significantly less in the shutdown modes and some accidents are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.35 Technical Specification 3.7.10 – Control Room Emergency Filtration System (CREFS)

Description

The CREFS provides a protected environment from which the operators can control the unit following an uncontrolled release of radioactivity, chemicals, or toxic gas.

The CREFS consists of two independent, redundant trains that recirculate and filter the control room air. Each train consists of a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters to remove water droplets from the air stream. A second bank of HEPA filters is downstream of the adsorber section to collect carbon fines and provide backup in case of failure of the main HEPA filter bank.

Limiting Condition for Operation

Two CREFS trains shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time for Condition A or B not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

INSERT J1

If one CREFS train is inoperable, the other train provides the necessary filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event must occur along with core damage and containment isolation failure for filtration to be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

INSERT J2

If one CREFS train is inoperable, the other train remains available to provide control room filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event and subsequent failures must occur for filtration to be required. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.36 Technical Specification 3.7.11 – Control Room Emergency Air Temperature Control System (CREATCS)

Description

The CREATCS provides temperature control for the control room following isolation of the control room.

The CREATCS consists of two independent and redundant trains that provide cooling and heating of recirculated control room air. Each train consists of heating coils, cooling coils, instrumentation, and controls to provide for control room temperature control.

Limiting Condition for Operation

Two CREATCS trains shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A not met in Mode 1, 2, 3, or 4.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

INSERT J1

If one CREFS train is inoperable, the other train provides the necessary filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event with a radioactive release must occur for radioactive filtration to be required. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2 of the WCAP. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event was to occur.

CREFS also provides protection from chemical releases, toxic gas releases, or radiation releases from other sources on-site and offsite. The likelihood of these events occurring are independent of the unit operating Mode.

INSERT J2

If one CREFS train is inoperable, the other train remains available to provide control room filtration. If two CREFS trains are inoperable due to an inoperable control room boundary, an independent initiating event and radioactive release must occur for filtration to be required. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A not met in Mode 1, 2, 3, or 4.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one CREATCS train is inoperable, the other train provides the necessary temperature control. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one CREATCS train is inoperable, the other train remains available to provide control room temperature control. The slower nature of accident event progression in the shutdown modes, and increased time for operator actions and mitigation strategies, limit the severity of accidents in the shutdown modes. The inoperability of equipment does not affect the likelihood of an event occurring and some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.37 Technical Specification 3.7.12 – Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)

Description

The ECCS PREACS filters air from the area of the active ECCS components during the recirculation phase of a LOCA. The ECCS PREACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the ECCS pump room area and the lower areas of the Auxiliary Building.

The ECCS PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters functioning to reduce the relative humidity of the air stream. A second bank of HEPA filters is downstream of the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the accident analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the pump room following receipt of a SI signal.

Limiting Condition for Operation

Two ECCS PREACS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. This system has no impact on plant CDP. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one ECCS PREACS train is inoperable, the other train provides the necessary filtration. If two trains are inoperable due to an inoperable ECCS pump room boundary, a LOCA must also occur to require the operation of the ECCS PREACS. A LOCA in Mode 4 is less likely due to the reduced RCS temperature and pressure. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one ECCS PREACS train is inoperable, the other train remains available to provide pump room air filtration. If two trains are inoperable due to an inoperable ECCS pump room boundary, a LOCA must also occur to require operation of the ECCS PREACS. The slower nature of accident event progression in the shutdown modes, and increased time for operator actions and mitigation strategies, limit the severity of accidents in the shutdown modes. In addition, a LOCA is less likely to occur in the shutdown modes.

Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.38 Technical Specification 3.7.13 – Fuel Building Air Cleanup System (FBACS)

Description

The FBACS filters airborne radioactive particulates from the area of the fuel pool following a fuel handling accident or a LOCA. A LOCA is analyzed to address radioactive leakage from the ECCS. The FBACS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the fuel pool area.

The FBACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation also form part of the system, as well as demisters, functioning to reduce the relative humidity of the airstream. A second bank of HEPA filters is downstream of the adsorber section to collect carbon fines and provide backup in case the main HEPA filter bank fails. The downstream HEPA filter is not credited in the analysis, but serves to collect charcoal fines, and to back up the upstream HEPA filter should it develop a leak. The system initiates filtered ventilation of the fuel handling building following receipt of a high radiation signal.

Limiting Condition for Operation

Two FBACS trains shall be operable.

Applicability

Modes 1, 2, 3, 4, [5, and 6], during movement of [recently] irradiated fuel assemblies.

Condition Requiring Entry into Actions or a Unit Shutdown

[Required Action and associated Completion Time of Condition A or B not met in Mode 1, 2, 3, or 4.] or Two FBACS trains inoperable in Mode 1, 2, 3, or 4 for reasons other than Condition B.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the [Required Action and associated Completion Time of Condition A or B not met in Mode 1, 2, 3, or 4.] or Two FBACS trains inoperable in Mode 1, 2, 3, or 4 for reasons other than Condition B.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. This system has no impact on plant CDP. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one FBACS train is inoperable, the other train provides the necessary filtration. If two FBACS trains are inoperable, a LOCA or fuel handling accident must also occur to require the operation of the FBACS. A LOCA in Mode 4 is less likely due to the reduced RCS temperature and pressure. If irradiated fuel is being moved, Condition E of the Specification requires that the movement be suspended. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one FBACS train is inoperable, the other train remains available to provide fuel building air filtration. If two FBACS trains are inoperable, a LOCA or fuel handling accident must also occur to require operation of the FBACs. LOCAs are less likely in Mode 4 because of the reduced RCS temperature and pressure in Mode 4 and Condition E reduces the probability of a fuel handling accident. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.39 Technical Specification 3.7.14 – Penetration Room Exhaust Air Cleanup System (PREACS)

Description

The PREACS filters air from the penetration area between containment and the Auxiliary Building.

The PREACS consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the air stream, also form part of the system. A second bank of HEPA filters, downstream of the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter, although not credited in the accident analysis, collects charcoal fines and serves as a backup should the upstream HEPA filter develop a leak. The system initiates filtered ventilation following receipt of a safety injection signal.

Limiting Condition for Operation

Two PREACS trains shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action C.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action C.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

This system is not modeled in the risk models described in Section 6.3.1, and it is not typically modeled in Westinghouse NSSS plant PRAs. This system has no impact on plant CDP. Therefore, a qualitative evaluation is performed for this proposed endstate change.

If one PREACS train is inoperable, the other train provides the necessary filtration. If two PREACS trains are inoperable due to an inoperable penetration room boundary, a LOCA and a passive failure in the penetration room must occur to require air filtration. LOCAs are less likely in Mode 4 because of the reduced RCS temperature and pressure. A cool down to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions, and there is a lower overall risk than proceeding to Mode 5 as demonstrated in Section 6.3.2. LERP in the shutdown modes would be reduced due to lower levels of radionuclide inventory in the RCS, the slower nature of severe accident event progression, and increased time for operator actions and mitigation strategies if an event were to occur.

Defense-in-Depth Considerations

If one PREACS train is declared inoperable, the other train remains available to provide penetration room air filtration. If two PREACS trains are inoperable due to an inoperable penetration room boundary, a LOCA and passive failure in the penetration room must occur to require air filtration. A LOCA is less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.40 Technical Specification 3.8.1 – AC Sources – Operating

Description

The unit Class 1E AC Electrical Power Distribution System AC sources consist of the offsite power sources (preferred power sources, normal and alternate(s)), and the onsite standby power sources (Train A and Train B DGs). The design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the ESF systems.

The onsite Class 1E AC Distribution System is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two preferred offsite power sources and a single DG.

Offsite power is typically supplied to the unit switchyard(s) from the transmission network by two transmission lines. From the switchyard(s), two electrically and physically separated circuits provide AC power, through step down station auxiliary transformers, to the 4.16 kV ESF buses. An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus(es).

Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E Distribution System. Within [1] minute after the initiating signal is received, all automatic and permanently connected loads needed to recover the unit or maintain it in a safe condition are returned to service via the load sequencer.

After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone.

Limiting Condition for Operation

The following AC electrical sources shall be operable:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System,

b. Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystem(s), and

c. [Automatic load sequencers for Train A and Train B.]

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time of Condition A, B, C, D, E, or [F] not met.

Current Required Action Endstate

The current endstate for Required Action G.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action G.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time of Condition A, B, C, D, E, or [F] not met.

Basis for Proposed Change

Electric power is included in the risk models described in Section 6.3.1. Each condition of this Technical Specification is evaluated quantitatively by modeling the inoperable equipment. The Conditions are grouped according to equipment and are not necessarily in the order in which the Condition appears in the Technical Specification.

Conditions A and C – Offsite Power Circuits

Condition A – One Offsite Circuit Inoperable

A transformer in the risk models was selected as the component to use for one offsite circuit inoperable. The transformer was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The results are presented in Table 6-16. ~~Also provided are the CDPs from the base case.~~

Condition C – Two Offsite Circuits Inoperable

A transformer common cause basic event in the risk models was selected to model two offsite circuits inoperable. The common cause basic event was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The results are presented in Table 6-16. ~~Also provided are the CDPs from the base case.~~

There is a significant increase in the CDP when two offsite power circuits are inoperable compared to one circuit being inoperable. This is expected because the unit is dependent on the diesel generators for power when both circuits are inoperable. For one circuit inoperable, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by ~~approximately~~ almost a factor of ~~3~~ 2. When both offsite circuits are inoperable, the decrease in CDP for remaining in Mode 4 is approximately a factor of ~~4~~ 5.

INSERT K

Table 6-16 Technical Specification 3.8.1, Conditions A and C, Offsite AC Circuits Inoperable			
	Core Damage Probability		
POS	One Offsite Circuit Inoperable	Two Offsite Circuits Inoperable	Base Case
1	5.82E-07	8.62E-04	2.18E-07
2	2.35E-07	1.47E-04	1.66E-07
3	1.37E-07	1.47E-04	7.09E-08
4	1.17E-05	8.06E-03	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06
Total	1.77E-05	9.22E-03	1.27E-05
Total Excluding POS 4	5.99E-06	1.16E-03	5.49E-06

<u>Conditions B and E – Diesel Generators</u>

Condition B – One Diesel Generator (DG) Inoperable

One diesel generator was modeled as inoperable and the core damage probabilities were recalculated for POS 1, 2, 3, and 4. The resulting CDPs are presented in Table 6-17 along with the CDPs from the base case.

Condition E – Two Diesel Generators Inoperable

A common cause basic event for both diesel generators was selected to model the inoperability of two diesel generators. The POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-17 ~~along with the CDPs from the base case.~~ e

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Table 6-17 Technical Specification 3.8.1, Conditions B and E, Diesel Generator(s) Inoperable			
	Core Damage Probability		
POS	One DG Inoperable	Two DGs Inoperable	Base Case
1	3.54E-07	2.38E-06	2.18E-07
2	2.15E-07	9.51E-07	1.66E-07
3	1.49E-07	1.31E-06	7.09E-08
4	1.37E-05	1.10E-04	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06
Total	1.95E-05	1.20E-04	1.27E-05
Total Excluding POS 4	5.76E-06	9.68E-06	5.49E-06

INSERT K

Table 6-16	Technical Specification 3.8.1, Conditions A and C, Offsite AC Circuits Unavailable			
POS	Core Damage Probability			
	One Offsite Circuit Unavailable		Two Offsite Circuits Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	5.82E-07	5.82E-07	8.62E-04	8.62E-04
2	2.35E-07	2.35E-07	1.47E-04	1.47E-04
3	9.34E-07	1.37E-07	1.09E-04	1.47E-04
4	NA	6.53E-06	NA	4.86E-03
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	6.79E-06	1.25E-05	1.12E-03	6.02E-03

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Table 6-17	Technical Specification 3.8.1, Conditions B and E, Diesel Generator(s) Unavailable			
POS	Core Damage Probability			
	One DG Unavailable		Two DGs Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	3.54E-07	3.54E-07	2.38E-06	2.38E-06
2	2.15E-07	2.15E-07	9.51E-07	9.51E-07
3	1.68E-06	1.49E-07	1.79E-05	1.31E-06
4	NA	1.04E-05	NA	1.05E-04
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	7.29E-06	1.62E-05	2.63E-05	1.15E-04

For the inoperability of one DG, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of ~~3.~~ 2 When both DGs are inoperable, the decrease in CDP for remaining in Mode 4 is approximately a factor ~~12.~~ 4

Condition D – One Offsite Circuit and One DG Inoperable

A transformer and diesel generator were selected to model the inoperable equipment for this condition. The POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-18. ~~Also provided are the CDPs from the base case.~~

INSERT M

Table 6-18 Technical Specification 3.8.1, Condition D One Offsite Circuit and One DG Inoperable

POS	Core Damage Probability: One Offsite Circuit and One DG Inoperable	Core Damage Probability: Base Case
1	4.64E-06	2.18E-07
2	1.03E-06	1.66E-07
3	9.37E-07	7.09E-08
4	6.69E-05	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	7.85E-05	1.27E-05
Total Excluding POS 4	1.16E-05	5.49E-06

The CDPs for one offsite AC circuit and one DG inoperable, as expected, are greater than the CDPs for one offsite AC circuit inoperable and are also greater than the CDPs for one DG inoperable. The total CDP decreases by more than a factor of ~~6~~ 3 when the unit is cooled down to Mode 4 (POS 3) instead of Mode 5 (POS 4). This is a larger decrease than for either the one offsite AC circuit inoperable case or the one DG inoperable case.

Condition F – One Load Sequencer Inoperable

One load sequencer was selected to model this condition. The load sequencer was modeled as inoperable and the POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-19. ~~Also provided are the CDPs from the base case.~~

INSERT M

Table 6-18	Technical Specification 3.8.1, Condition D One Offsite Circuit and One DG Unavailable	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	4.64E-06	4.64E-06
2	1.03E-06	1.03E-06
3	5.95E-06	9.37E-07
4	NA	3.95E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.67E-05	5.11E-05

INSERT N

Table 6-19 Technical Specification 3.8.1, Condition F One Load Sequencer Inoperable

POS	Core Damage Probability	
	One Load Sequencer Inoperable	Base Case
1	3.94E-07	2.18E-07
2	2.32E-07	1.66E-07
3	1.57E-07	7.09E-08
4	1.42E-05	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	2.00E-05	1.27E-05
Total Excluding POS 4	5.82E-06	5.49E-06

The results for this case are similar to the results for the one DG inoperable case, which is expected, because the consequences of the inoperable equipment are similar. For the inoperability of one load sequencer, remaining in Mode 4 (POS 3) instead of cooling down to Mode 5 (POS 4) reduces the total core damage probability by more than a factor of ~~2.~~ 2

For each of the Conditions of this Technical Specification that were evaluated, the results show that a cool down to Mode 4 instead of Mode 5 reduces the risk of the shutdown process when the unit has inoperable electrical power components. The reduction in risk ranges from a factor of 3 to a factor of 12.

Defense-in-Depth Considerations

The electric power design maintains defense-in-depth when the Conditions for this Technical Specification are considered. Two trains of diesel generators are available if two offsite power circuits are inoperable and two offsite power circuits are available if two diesel generators are inoperable. If an offsite power circuit and/or a diesel generator are inoperable, at least one of each remains available. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.41 Technical Specification 3.8.4 – DC Sources – Operating

Description

The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety related equipment and preferred AC vital bus power (via inverters). The DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure.

INSERT N

Table 6-19 Technical Specification 3.8.1, Condition F One Load Sequencer Unavailable		
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	3.94E-07	3.94E-07
2	2.32E-07	2.32E-07
3	1.81E-06	1.57E-07
4	NA	1.09E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	7.48E-06	1.67E-05

The typical 125/250 VDC electrical power system consists of two independent and redundant safety related Class 1E DC electrical power subsystems (Train A and Train B). Each subsystem consists of two 125 VDC batteries (each battery with 50% capacity), the associated battery charger(s) for each battery, and all the associated control equipment and interconnecting cabling.

The typical 250 VDC source is obtained by use of the two 125 VDC batteries connected in series. Additionally there is one spare battery charger per subsystem, which provides backup service in the event that the preferred battery charger is out of service. If the spare battery charger is substituted for one of the preferred battery chargers, then the requirements of independence and redundancy between subsystems are maintained.

During normal operation, the 125/250 VDC load is powered from the battery chargers with the batteries floating on the system. In case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

Limiting Condition for Operation

The Train A and Train B DC electrical power subsystems shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

DC power is included in the risk models described in Section 6.3.1. Each Condition of this Technical Specification is evaluated quantitatively by modeling the inoperable equipment.

Condition A – One [or Two] Battery Charger[s on one train] Inoperable

There is one battery charger per battery (and per train) in the risk models described in Section 6.3.1. One battery charger was modeled as inoperable and the POS 1, 2, 3, and 4 risk models were requantified.

The resulting CDPs for each POS are presented in Table 6-20. ~~Also provided are the CDPs from the base case.~~

INSERT O

Table 6-20 Technical Specification 3.8.4, Condition A Battery Charger Inoperable		
	Core Damage Probability	
POS	Battery Charger Inoperable	Base Case
1	2.18E-07	2.18E-07
2	1.66E-07	1.66E-07
3	7.14E-08	7.09E-08
4	7.21E-06	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.27E-05	1.27E-05
Total Excluding POS 4	5.49E-06	5.49E-06

The results are basically the same as the base case. The CDP is reduced by slightly more than a factor of 1.5 ~~two~~ when the unit is cooled down to Mode 4 instead of Mode 5.

Condition B – One Battery/Two Batteries on One Train Inoperable

There is one battery modeled per train in the risk models described in Section 6.3.1. One battery was modeled as inoperable and the POS 1, 2, 3, and 4 risk models were requantified. The resulting CDPs for each POS are presented in Table 6-21. ~~Also provided are the CDPs from the base case.~~

INSERT P

Table 6-21 Technical Specification 3.8.4, Condition B One Battery Inoperable		
	Core Damage Probability	
POS	One Battery Inoperable	Base Case
1	3.85E-07	2.18E-07
2	2.36E-07	1.66E-07
3	1.21E-07	7.09E-08
4	1.43E-05	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	2.01E-05	1.27E-05
Total Excluding POS 4	5.83E-06	5.49E-06

INSERT O

Table 6-20	Technical Specification 3.8.4, Condition A Battery Charger Unavailable	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	2.18E-07	2.18E-07
2	1.66E-07	1.66E-07
3	5.95E-07	7.14E-08
4	NA	4.03E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	6.02E-06	9.52E-06

INSERT P

Table 6-21	Technical Specification 3.8.4, Condition B One Battery Unavailable	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	3.85E-07	3.85E-07
2	2.36E-07	2.36E-07
3	1.78E-06	1.71E-07
4	NA	1.10E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	7.44E-06	1.68E-05

The results are expected to be similar to those for the battery charger. The inoperable battery CDPs are larger than the CDPs for an inoperable battery charger. The battery chargers are the primary source for DC power and the batteries serve as the backup, however, the failure probability for the battery chargers is twice that of the batteries. Therefore, modeling the battery as inoperable results in a larger change in the CDPs. For this case, the CDP is reduced by more than a factor of ~~three~~ 2 when the unit is cooled down to Mode 4 instead of Mode 5.

Condition C – One DC Subsystem (train) Inoperable for Reasons Other Than Condition A or B

A DC panel was selected to model the inoperability of one DC train. The risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-22. ~~Also provided are the base CDPs.~~

INSERT Q

Table 6-22 Technical Specification 3.8.4, Condition C One DC Subsystem Inoperable for Reasons Other Than Condition A or B

	Core Damage Probability	
POS	One DC Subsystem Inoperable	Base Case
1	5.73E-05	2.18E-07
2	6.11E-05	1.66E-07
3	6.10E-05	7.09E-08
4	8.79E-05	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	2.72E-04	1.27E-05
Total Excluding POS 4	1.84E-04	5.49E-06

The CDP values for POS 1, 2, 3, and 4 are larger than those of the previous two cases. This is expected because one entire DC subsystem is inoperable. For this case, the CDP is reduced by ~~more~~ a factor of approximately 1.5 when the unit is cooled down to Mode 4 instead of Mode 5. This is a smaller decrease than the other two cases. The POS 4 CDP is greater for this case than the other two cases, however, the CDPs for the POS 1, 2, and 3 are much greater for this case than they are for the previous two cases. The POS 1, 2, and 3 and CDPs lessen the effect of the POS 4 contribution. ~~Note that the DC panel chosen for this case fails the Train A pump of emergency feedwater (EFW, also known as AFW) that is modeled as the running pump for POS 3. The model treats this as a failure during POS 3 and then addresses starting the Train B pump and associated check valve opening failures. These actions would have taken place prior to entering POS 3. As a result, the loss of EFW initiating event dominates the results with cutsets that are not applicable for this situation. Therefore, there is a greater difference between the POS 3 risk and the POS 4 risk than indicated by the results.~~

INSERT Q

Table 6-22	Technical Specification 3.8.4, Condition C One DC Subsystem Unavailable for Reasons Other Than Condition A or B	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	5.73E-05	5.73E-05
2	6.11E-05	6.11E-05
3	8.05E-06	6.10E-05
4	NA	3.49E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.31E-04	2.19E-04

All three cases presented demonstrate a reduction in risk when the unit is cooled down to Mode 4 instead of Mode 5.

Defense-in-Depth Considerations

The DC power system is designed for the battery chargers to provide the normal DC power. The batteries back up the chargers in the event that AC power to the chargers is lost. There are two redundant trains of DC power so if one is inoperable, the other is available to provide the necessary DC power. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.4.42 Technical Specification 3.8.7 – Inverters – Operating

Description

The inverters are the preferred source of power for the AC vital buses because of the stability and reliability they achieve. The function of the inverters is to provide AC electrical power to the vital buses. The inverters can be powered from an internal AC source/rectifier or from the station battery. The station battery provides an uninterruptible power source for the instrumentation and controls for the RPS and ESFAS. The four inverters [(two per train)] ensure an uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized.

Limiting Condition for Operation

The required Train A and Train B inverters shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action B.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action B.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The inverters are included in the risk models described in Section 6.3.1. Based on the specific design of the unit chosen for the generic PRA model, the two inverter trains are not symmetrically modeled. Train A includes one inverter and Train B includes two inverters. Two cases were evaluated for this Technical Specification. One case models Train A inoperable and the other case models Train B inoperable. The risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4 for each case. The resulting CDPs for each POS are presented in Table 6-23. ~~Also provided are the base CDPs.~~

INSERT R

Table 6-23 Technical Specification 3.8.7, Inverters – Operating

POS	Core Damage Probability		
	Train A – One Inverter Inoperable	Train B – Two Inverters Inoperable	Base Case
1	4.05E-07	4.00E-07	2.18E-07
2	2.36E-07	2.35E-07	1.66E-07
3	1.58E-07	1.52E-07	7.09E-08
4	1.42E-05	1.37E-05	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06
Total	2.00E-05	1.95E-05	1.27E-05
Total Excluding POS 4	5.84E-06	5.83E-06	5.49E-06

Although the fault tree modeling is slightly different for the two trains, the results are basically the same. For these two cases, the CDP is reduced by more than a factor of ~~[illegible]~~ 2 when the unit is cooled down to Mode 4 instead of Mode 5.

Defense-in-Depth Considerations

The inverters can be powered from AC sources or from the batteries. There are two redundant trains of inverters so if one is inoperable, the other is available to provide the necessary AC power. The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

INSERT R

Table 6-23 Technical Specification 3.8.7, Inverters – Operating

POS	Core Damage Probability			
	Train A – One Inverter Unavailable		Train B – Two Inverters Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	4.05E-07	4.05E-07	4.00E-07	4.00E-07
2	2.36E-07	2.36E-07	2.35E-07	2.35E-07
3	1.82E-06	1.58E-07	1.73E-06	1.52E-07
4	NA	1.09E-05	NA	1.04E-05
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	7.50E-06	1.67E-05	7.40E-06	1.62E-05

6.4.43 Technical Specification 3.8.9 – Distribution Systems – Operating

Description

The typical onsite Class 1E AC, DC, and AC vital bus electrical power distribution systems are divided by trains into two redundant and independent AC, DC, and AC vital bus electrical power distribution subsystems.

The AC electrical power subsystem for each train consists of a primary ESF 4.16 kV bus and secondary 480 and 120 V buses, distribution panels, motor control centers and load centers. Each 4.16 kV ESF bus has at least one separate and independent offsite source of power as well as a dedicated onsite DG source. Each 4.16 kV ESF bus is normally connected to a preferred offsite source. After a loss of the preferred offsite power source to a 4.16 kV ESF bus, a transfer to the alternate offsite source is accomplished by utilizing a time delayed bus undervoltage relay. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

The secondary AC electrical power distribution subsystem for each train includes safety related buses, load centers, motor control centers, and distribution panels.

The 120 VAC vital buses are arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the vital buses are Class 1E constant voltage source transformers powered from the same train as the associated inverter. Each constant voltage source transformer is powered from a Class 1E AC bus.

The DC electrical power distribution subsystem typically consists of 125 V buses and distribution panels.

Limiting Condition for Operation

Train A and Train B AC, DC, and AC vital bus electrical power distribution subsystems shall be operable.

Applicability

Modes 1, 2, 3, and 4.

Condition Requiring Entry into Actions or a Unit Shutdown

Required Action and associated Completion Time not met.

Current Required Action Endstate

The current endstate for Required Action D.2 is Mode 5. Specifically, the unit must be in Mode 3 in 6 hours and Mode 5 in 36 hours.

Proposed Required Action and Endstate

Revise the endstate for Required Action D.2 to be in Mode 4 in 12 hours if the Required Action and associated Completion Time not met.

Basis for Proposed Change

The electrical distribution systems are included in the risk models described in Section 6.3.1. Each Condition of this Technical Specification is evaluated quantitatively by modeling the inoperable equipment.

Condition A – One or More AC Electrical Power Distribution Subsystems Inoperable

A representative electrical bus was chosen to provide representative results for one AC electrical power distribution subsystem inoperable. The bus was modeled as inoperable and the risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-24. ~~Also provided are the base CDPs.~~ INSERT S

Table 6-24 Technical Specification 3.8.9, Condition A AC Power Distribution Subsystem Inoperable

	Core Damage Probability	
POS	**One AC Power Distribution Subsystem Inoperable**	**Base Case**
1	2.11E-05	2.18E-07
2	3.52E-06	1.66E-07
3	2.43E-06	7.09E-08
4	7.38E-05	7.21E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	1.05E-04	1.27E-05
Total Excluding POS 4	3.11E-05	5.49E-06

The CDPs for this case are greater than those for the loss of an offsite source (Table ~~6-26~~ 6-16). They are also greater than those for having one DG inoperable (Table 6-17). This is expected because an inoperable onsite source disables the offsite power feed as well as the associated diesel generator. For this case, the CDP is reduced by ~~more than~~ almost a factor of ~~3~~ 2 when the unit is cooled down to Mode 4 instead of Mode 5.

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Table 6-24	Technical Specification 3.8.9, Condition A AC Power Distribution Subsystem Unavailable	
POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	2.11E-05	2.11E-05
2	2.52E-06	2.52E-06
3	8.17E-06	2.43E-06
4	NA	3.53E-05
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
Total	3.68E-05	6.64E-05

Condition B – One or More AC Vital Buses Inoperable

Two cases are examined for this Technical Specification Condition; one inoperable panel and three inoperable panels to model one or more vital buses. The equipment was modeled as inoperable and the risk models described in Section 6.3.1 were requantified for POS 1, 2, 3, and 4. The resulting CDPs for each POS are presented in Table 6-25. ~~Also provided are the base CDPs.~~

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Table 6-25 Technical Specification 3.8.9, Condition B AC Vital Buses Inoperable

POS	Core Damage Probability		
	One Panel Inoperable	Three Panels Inoperable	Base Case
1	4.05E-07	5.13E-06	2.18E-07
2	2.36E-07	1.99E-06	1.66E-07
3	1.58E-07	1.45E-06	7.09E-08
4	1.42E-05	1.18E-04	7.21E-06
5	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06
Total	2.00E-05	1.24E-04	1.27E-05
Total Excluding POS 4	5.84E-06	1.36E-05	5.49E-06

The CDPs for the three panel case are ~~almost a factor of 10~~ greater than the one panel case for POSs 1, 2, 3, and 4. For the one panel case, the CDP is reduced by more than a factor of ~~2~~ 2 when the unit is cooled down to Mode 4 instead of Mode 5. For the three panel case, the CDP is reduced by more than a factor of ~~9~~ 3 when the unit is cooled down to Mode 4 instead of Mode 5.

Condition C – One or More DC Electrical Power Distribution Subsystems Inoperable

This Condition evaluates one DC subsystem being inoperable. If more than one DC subsystem is inoperable, the results will differ, however, the results of Technical Specification 3.8.4 Condition C (Table 6-22) and other DC cases run have demonstrated the same conclusion; the risk of the shutdown process, in terms of CDP, is reduced when the unit is cooled down to Mode 4 instead of Mode 5. Therefore, a separate quantitative risk evaluation was not performed for this Condition.

The risk evaluation cases run for Technical Specification 3.8.9 demonstrate that the risk of the shutdown process, in terms of CDP, is reduced when the unit is cooled down to Mode 4 instead of Mode 5.

Defense-in-Depth Considerations

If Technical Specification 3.8.9 Condition D applies, there is no loss of safety function and the operable electrical power distribution subsystems are capable of supporting the minimum safety functions necessary to mitigate accidents and shut down the reactor and maintain it in a safe shutdown condition.

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Table 6-25	Technical Specification 3.8.9, Condition B AC Vital Buses Unavailable			
POS	Core Damage Probability			
	One Panel Unavailable		Three Panels Unavailable	
	Cooldown to POS 3	Cooldown to POS 4	Cooldown to POS 3	Cooldown to POS 4
1	4.05E-07	4.05E-07	5.13E-06	5.13E-06
2	2.36E-07	2.36E-07	1.99E-06	1.99E-06
3	1.82E-06	1.58E-07	2.00E-05	1.45E-06
4	NA	1.09E-05	NA	1.05E-04
5	4.79E-07	4.79E-07	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06	1.26E-06	1.26E-06
Total	7.50E-06	1.67E-05	3.22E-05	1.19E-04

The slower nature of event progression in the shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur in the shutdown modes. Therefore, sufficient defense-in-depth is maintained when the endstate is changed from Mode 5 to Mode 4.

6.5 SENSITIVITY ANALYSIS

6.5.1 POS 4 Time Sensitivity

The time spent in POS 4 was estimated to be 70 hours as described in Section 6.3.1.2. 24 hours were assumed to begin and complete the transition from AFW to RHR cooling as required by the Technical Specifications to be in Mode 5 from Mode 4. The remaining 46 hours were estimated from plant startup data. The number of hours in the POS has an important effect on the predicted CDP for that POS. Because the quantitative evaluations in this report are used to demonstrate an increase in risk by cooling down to POS 4, a sensitivity case is performed to examine the effects of reducing the amount of time assumed for POS 4. 24 hours were chosen for the sensitivity study because the base model assumes that it will take 24 hours to complete the transition from Mode 4 to Mode 5.

Basic events and initiating event probabilities that are dependent on the time assumed for POS 4 were recalculated and the POS 4 fault tree was requantified. Table 6-26 presents the revised POS 4 CDP with the base case results.

Table 6-26 Pos 4 Time Sensitivity Case

	Core Damage Probability	
POS	**24 hour POS 4**	**Base Case**
1	2.18E-07	2.18E-07
2	1.66E-07	1.66E-07
3	7.09E-08	7.09E-08
4	~~2.30E-06~~ 2.08E-06	~~7.21E-06~~ 4.03E-06
5	4.79E-07	4.79E-07
6	3.30E-06	3.30E-06
7	1.26E-06	1.26E-06
TOTAL	~~7.08E-06~~ 7.57E-06	~~1.27E-05~~ 9.52E-06
~~TOTAL Excluding POS 4~~	~~5.49E-06~~	~~5.49E-06~~

For the time sensitivity case, the CDP is reduced by a factor of 1.4 when the unit is cooled down to POS 3 (Mode 4) instead of POS 4 (Mode 5). For the base case, the CDP is reduced by more than a factor of 2 when the unit is cooled down to POS 3 instead of POS 4. The results demonstrate that there is a dependency on the time assumed for POS 4. The Technical Specification Condition specific runs in Section 6.3 would also show lower reduction factors for removing the POS 4 CDP if 24 hours are

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assumed for POS 4. The numerical results would change, however, not the conclusion that there is greater risk for the unit to cool down to Mode 5 (POS 4) instead of Mode 4 (POS 3).

Another insight from Table 6-26 comes from comparing the base case POS 3 and POS 5 CDPs to the 24 hour POS 4 CDP. POS 3 and POS 5 are very similar states and their combined time is 22 hours (refer to Table 6-9). The POS 3 and POS 5 total CDP is 5.5E-07. The 24 hour POS 4 CDP is 2.2E-06. The POS 4 time is less than 10 percent greater than the combined POS 3 and POS 5 times (24 versus 22 hours), however, the POS 4 CDP is 4 times greater than the combined POS 3 and POS 5 total. This is another demonstration of the increase in risk when the plant cools down to POS 4 versus remaining in POS 3.

6.5.2 Steam Generator Tube Rupture Initiating Event Frequency Sensitivity

To address the various types of steam generators installed in the Westinghouse NSSS designed plants, the effects of assuming a larger SGTR initiating event frequency was investigated. As stated in the text following Table 6-8, the SGTR initiating event frequency used was 2.7E-03/yr. This value was increased by a factor of 4 for the base case and the results are provided in Table 6-27.

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Table 6-27 Steam Generator Tube Rupture Frequency Sensitivity

POS	Core Damage Probability: Base Case	Core Damage Probability: Sensitivity Case	Percent Increase
1	2.18E-07	2.19E-07	0.3
2	1.66E-07	1.86E-07	12
3	7.09E-08	9.09E-08	28
4	7.21E-06	7.21E-06	0.0
5	4.79E-07	6.05E-07	26
6	3.30E-06	3.65E-06	10
7	1.26E-06	1.26E-06	0.3
Total	1.27E-05	1.32E-05	4
Total Excluding POS 4	5.49E-06	6.01E-06	9

POS 3 and POS 5 are the most affected by the increase in the SGTR frequency. The increase is small for the total ~~and the total excluding POS 4~~. Note that POS 4 does not include the SGTR initiating event and, therefore, is not affected by the sensitivity. Although there is an increase in the core damage probability, the increase is small compared to the change in the initiating event frequency and the changes do not alter the conclusions drawn from the model quantifications presented in Section 6.4.

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The CDP for the 24 hour POS 4 case is 2.08E-06. This is reduced from the base case POS 4 CDP of 4.03E-06 as expected. The base case CDP for POS 3 is 5.95E-07 and it assumes a 49 hour mission time. This provides further evidence that a shutdown to Mode 4 (POS 3) has less risk associated with it than a shutdown to Mode 5 (POS 4).

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Table 6-27 Steam Generator Tube Rupture Frequency Sensitivity

POS	Core Damage Probability	
	Cooldown to POS 3	Cooldown to POS 4
1	2.19E-07	2.19E-07
2	1.86E-07	1.86E-07
3	9.22E-07	9.09E-08
4	NA	4.03E-06
5	6.05E-07	6.05E-07
6	3.65E-06	3.65E-06
7	1.26E-06	1.26E-06
Total	6.84E-06	1.00E-05

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6.5.3 Probability of Power Recovery Sensitivity

For the loss of offsite power in POS 4, the probability of not recovering power within 2 hours is 0.5. This is based on power recovery information in Reference 6. This probability also includes the probability of equipment failing to start after the power recovery and operator actions. A sensitivity case was run assuming that the probability of not recovering power within 2 hours is 0.1. The revised base case POS 4 results are reduced from a core damage probability of 4.03E-06 to 3.46E-06. This is not a large change in CDP for a factor of 5 reduction in the probability of not recovering power. Although there is a decrease in the POS 4 core damage probability, the decrease is small compared to the change in the probability of not recovering power and the change does not alter the conclusions drawn from the model quantifications presented in Section 6.4.

6.6 TIER 2 AND 3 REQUIREMENTS

Regulatory Guide 1.177 defines a Tier 2 assessment as providing reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is out-of-service consistent with the proposed Technical Specification change. The risk-informed evaluations provided in Section 6.4 address one Technical Specification Condition at a time, evaluating a change to the endstate from Mode 5 to Mode 4. The endstate change does not change the Technical Specification requirements for other equipment required to be operable in the shutdown modes. In addition, no completion times are being extended and no surveillance test intervals are being changed. The evaluations in Section 6.4 demonstrate that there is a reduction in risk and advantages in defense-in-depth when the unit cools down to Mode 4 instead of continuing the cooldown to Mode 5. ~~Therefore, Tier 2 requirements are not applicable to the proposed change to the Technical Specification endstate.~~ INSERT X

Regulatory Guide 1.177 defines a Tier 3 assessment as a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity. 10 CFR 50.65 (a)(4) requires that prior to performing maintenance activities, risk assessments shall be performed to assess and manage the increased risk that may result from proposed maintenance activities. These requirements are applicable to all plant modes. The plant-specific implementation of 10 CFR 50.65 (a)(4) provides assurance that risk-significant plant equipment outage configurations will not occur when equipment in addition to the equipment associated with the Technical Specification Condition that resulted in the shutdown is taken out of service.

6.7 EXTERNAL EVENTS EVALUATION

As noted in Section 6.3.1.3, CCW and SW internal flooding events are included in the PRA model because of their impact on RCP seal LOCAs. Due to reduced RCS pressure, RCP seal LOCAs are not modeled for POS 3, 4, and 5. Therefore, conclusions regarding the effects of flooding on the POS 3 and 4 risk cannot be drawn directly from the quantitative results. By the time the unit has entered POS 3, feedwater system operation has been terminated and secondary cooling is provided by the auxiliary feedwater system. Therefore, feedwater system flooding events are not applicable. The flooding events of interest for POS 3 and 4 involve the CCW and SW systems. The quantitative results presented in Tables 6-14 and 6-15 for CCW and SW, respectively, show that the risk for POS 4 is much greater than that for POS 3 when a train of these systems is inoperable. A partial or total loss of these systems and the subsequent consequences due to flooding events would exhibit a similar difference in risk between POS 3 and POS 4. It is therefore concluded that consideration of flooding events would not change the conclusions presented in Section 6.4.

For fire events, the Appendix R evaluations address the safe shutdown requirements. Operating procedures are in place to respond to fires. The operators will respond to fires occurring in Mode 4 similar to fires occurring in Mode 5. Additionally, more safety system equipment is required to be operable in Mode 4 than in Mode 5 by the Technical Specifications. This provides additional assurance of successful mitigation of fires and other external events.

A seismic PRA is not available for Modes 4 and 5, however, several observations can be made. It is reasonable to assume that the trains of a seismically qualified system will respond to the event in a similar manner for either mode. If one train of a system can still perform its functions after a seismic event, then

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The Mode 4 (POS 3) PRA model includes the auxiliary feedwater turbine-driven pump. The availability of this pump is important to both the qualitative and quantitative bases for determining that the risk associated with a plant shutdown to Mode 4 is less than the risk associated with a plant shutdown to Mode 5. Therefore, a Tier 2 requirement is added to the Technical Specification Bases for the Technical Specifications listed in Table 2-1.

it is likely that all trains will be able to perform their functions after a seismic event and the defense in depth rational developed for changing the endstate to Mode 4 remains applicable. In addition, a seismic event is likely to lead to a loss of power and there are more independent sources of power required to be available in Mode 4 than in Mode 5. It is concluded that considerations of seismic events would not change the conclusions developed in Section 6.4.

Although not included in the PRA models used to assess the risk for the various plant operating states, external events would not change the conclusions presented in Section 6.4. The risk associated with POS 3 is less than that for POS 4. With more equipment options ~~required~~ available for POS 3, there is less likelihood that all redundancy and defense-in-depth will be defeated by an external event in POS 3 than in POS 4.

7 SUMMARY OF RESULTS

An evaluation is performed to assess the risk associated with a cooldown to Mode 4 instead of Mode 5 for specific Technical Specification Required Actions. To perform this evaluation, plant operating states are defined for the changing conditions as a plant shuts down from full power operation and cools down to Mode 5. These POSs are evaluated qualitatively and quantitatively. The qualitative evaluation of the POSs demonstrates that there are advantages in risk and in defense-in-depth when the plant remains in POS 3 (Mode 4) rather than continuing the cooldown to POS 4 (Mode 5). Each Technical Specification listed in Table 2-1 is evaluated for a change in endstate from Mode 5 to Mode 4. These evaluations are presented in Sections 6.4.1 through 6.4.43. Technical Specifications addressing equipment included in the POS PRA models are evaluated quantitatively. In all cases, it is demonstrated that there is an increase in plant risk if the plant cools down to POS 4 (Mode 5) instead of POS 3 (Mode 4). Technical Specifications addressing equipment not included in the POS PRA models are evaluated qualitatively. Defense-in-depth is addressed for each Technical Specification and Sections 6.4.1 through 6.4.43 demonstrate that the endstates for the Technical Specifications listed in Table 2-1 can be changed from Mode 5 to Mode 4. Sensitivity cases are presented in Section 6.5 that examine the influence of the time modeled for POS 4, ~~and~~ the effects of assuming a different SGTR initiating event frequency, In support of these results, Section 6.6 addresses Tier 2 requirements and Section 6.7 qualitatively evaluates external events.

and a change in the probability of not recovering power for POS 4.

8 CONCLUSIONS

The Technical Specifications listed in Table 2-1 have been evaluated for a change in endstate from Mode 5 to Mode 4. The qualitative evaluation of the POSs in Section 6.2 concludes that there are advantages in risk and in defense-in-depth when the plant remains in POS 3 (Mode 4) rather than continuing to cooldown to POS 4 (Mode 5). Safety margins are not reduced as discussed in Section 5.2. Technical Specification-specific evaluations are presented in Sections 6.4.1 through 6.4.43. These qualitative and quantitative evaluations demonstrate that there is less risk if the endstates for these Technical Specifications are changed from Mode 5 to Mode 4. The model used for the quantitative evaluations is described and shown to be representative of Westinghouse NSSS plants. Defense-in-depth is addressed for each Technical Specification and the evaluations support the conclusion that the endstate can be changed from Mode 5 to Mode 4.

From a sensitivity case presented in Section 6.5, the results demonstrate that changes to the time duration modeled for POS 4 (Mode 5) does not affect the conclusion that there is less risk associated with a cooldown to Mode 4 than there is for a cooldown to Mode 5. ~~The other~~ Another sensitivity case of assuming a different SGTR initiating event frequency was applied to all POSs and the conclusion that there is less risk associated with a cooldown to Mode 4 than there is for a cooldown to Mode 5 remains applicable.

In support of the evaluations of risk for the Technical Specifications, Section 6.6 addresses Tier 2 and Tier 3 requirements. For Tier 3 requirements, it ~~It~~ is concluded that the program used to assess and manage the risk associated with maintenance as required by 10CFR50.65 (a)(4) provides reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment in addition to the affected Technical Specification is taken out of service.

Section 6.7 qualitatively evaluates external events and concludes that the inclusion of external events into the PRA models for the POSs would not change the conclusions presented in Section 6.4.

The above conclusions support changing the endstate from Mode 5 to Mode 4 for the Technical Specifications listed in Table 2-1, as discussed in Section 6.4. ~~and shown in the markups in Appendix B.~~

The final sensitivity case examines the effects of changing the probability of not recovery power for POS 4. The results do not change the previously stated conclusions.

A Tier 2 requirement is added to the Technical Specification Bases to ensure the availability of the turbine-driven auxiliary feedwater pump.

9 REFERENCES

1. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Rev. 1, November 2002.

2. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998.

3. NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.

4. NUREG/CR-6144, Vol. 2, Parts 1A and 1B, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," June 1994.

5. WCAP-14333-P-A, Rev. 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times," October 1998.

6. EPRI Technical Report 10029987, "Losses of Offsite Power at U.S. Nuclear Power Plants through 2001," April 2002.

7. WCAP-11737, "Low Temperature Overpressurization," March 1989.

APPENDIX A
TOP 100 CUTSETS FOR POS 3 AND POS 4

5.43E-07 5.95E-07

The top 100 cutsets for POS 3 are presented in Table A-1. These cutsets are from the base case that assumes that no equipment is out of service for testing or maintenance. The time duration for POS 3 is 49 ~~7~~ hours. The CDP for the top 100 cutsets is ~~6.48 E-08~~ which is slightly more than 90% of ~~7.09E-08~~, the total CDP calculated for POS 3. The largest initiating event contributor to the POS 3 CDP is Loss of Decay Heat Removal (EFW). This event is the loss of emergency feedwater (also known as AFW) and ~~in the top 100 cutsets,~~ it accounts for ~~4.18E-08~~ (approximately ~~60~~%) of the total CDP.

1.81E-07 30

The top 100 cutsets for POS 4 are presented in Table A-2. These cutsets are from the base case that assumes that no equipment is out of service for testing or maintenance. The time duration for POS 4 is 70 hours. The CDP for the top 100 cutsets is ~~7.06E-06~~ which is ~~98%~~ of ~~7.21E-06~~, the total CDP calculated for POS 4. The largest initiating event contributor to the POS 4 CDP is Loss of RHR. ~~In the top 100 cutsets,~~ This event accounts for ~~5.09E-06~~ (approximately ~~70%~~) of the total CDP.

3.89E-06 4.03E-06 96%

2.33E-06 58%

BLIND NOTE - TABLES A-1 AND A-2 WILL BE FORMATTED FOR THE WCAP WHEN IT IS REVISED

Cutsets with Descriptions Report
@CDFALLF = 5.95E-07
O:\RRA\WOG Projects\WOG Tech Spec Endstates\POS3 49 Hr Quantification\3I62POS3Rev1.CUT 11/13/07 4:56 PM

#	Cutset Prob	Event Prob	Event	Description
1	9.73E-08	1.52E-05	%SGR	STEAM GENERATOR TUBE RUPTURE INITIATING EVENT
		1.00E+00	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)
		6.40E-03	OAI_1	OPERATOR FAILS TO DIAGNOSE,IDENTIFY, & ISOLATE RUPTURED SG
2	8.16E-08	1.00E+00	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
		5.44E-06	EFWICC-XPP-FR04	XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF
		1.00E+00	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER
		1.50E-02	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
3	6.29E-08	8.39E-07	%SLO	SMALL LOCA INITIATING EVENT
		1.00E+00	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)
		7.50E-02	OAESF3	
4	2.05E-08	4.01E-05	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
		1.00E+00	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)
		7.50E-02	OAESF3	
		6.80E-03	UC-FTIFLTSGCHE	HUMAN ERROR FAILURE TO CLOSE EFW FLOW VALVES TO STEAM GEN C
5	1.61E-08	1.00E+00	%EFW1	LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
		7.35E-03	DBPTI---XPP8FR	TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
		1.46E-04	EFWICC-XPP-FR01	XPP21A, XPP21B FAIL TO RUN BY CCF
		1.00E+00	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER
		1.50E-02	OAR4	OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
6	1.56E-08	4.01E-05	%SSBI	SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
		1.00E+00	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)
		7.50E-02	OAESF3	
		5.20E-03	OAT2	OPERATOR FAILS TO TERMINATE SI GIVEN SSB
7	1.56E-08	4.01E-05	%SSBO	SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
		1.00E+00	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)
		7.50E-02	OAESF3	
		5.20E-03	OAT2	OPERATOR FAILS TO TERMINATE SI GIVEN SSB
8	1.34E-08	1.38E-04	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT
		3.68E-01	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
		1.83E-01	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
		5.83E-02	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
		5.83E-02	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
		1.00E+00	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER
		4.24E-01	XHR_2	FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
9	1.15E-08	1.38E-04	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT
		3.68E-01	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
		1.83E-01	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
		5.83E-02	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
		5.83E-02	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
		1.00E+00	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER
		3.65E-01	XHR_1	FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
10	1.12E-08	1.12E-08	%MLO	MEDIUM LOCA INITIATING EVENT
		1.00E+00	ESF-CC-DRIC-FOR	1 combination of 2 of 2 cards (driver circuit)
11	1.09E-08	1.38E-04	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT
		3.68E-01	1HR_1	FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
		1.83E-01	4HR_1	FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
		5.83E-02	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
		5.83E-02	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
		3.46E-01	CNU_2	CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
		1.00E+00	SBO-FLAG	STATION BLACKOUT SEQUENCE MARKER

12 1.00E-08 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
13 9.22E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
14 8.98E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.44E-06 EFWICC-XPP-FR04 XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
1.50E-01 OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)
15 8.01E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
3.99E-01 CNU_1 CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
6.35E-01 XHR_1-SUCCESS XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS
16 7.94E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
17 7.52E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
18 6.53E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.96E-03 DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
5.00E-01 EFW8P
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
19 6.02E-09 4.01E-05 %SSBI SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
7.50E-02 OAESF3
2.00E-03 UCAVIFV3551-FC FAILURE TO CLOSE IFV-3551 WHICH SUPPLIES MD HEADER FLOW
20 6.02E-09 4.01E-05 %SSBI SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
7.50E-02 OAESF3
2.00E-03 UCAVIFV3556-FC FAILURE TO CLOSE IFV-3556 WHICH SUPPLIES TD HEADER FLOW
21 5.81E-09 1.52E-05 %SGR STEAM GENERATOR TUBE RUPTURE INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
5.10E-03 OAD_1 OPERATOR FAILS TO DEPRESSURIZE SECONDARY SIDE (NORMAL COOL DOWN)
7.50E-02 OAESF3
22 5.51E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR

2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
3.99E-01 CNU_1 CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
6.35E-01 XHR_1-SUCCESS XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS
23 4.89E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
2.83E-02 CNU_4 CORE IS UNCOVERED AT 4 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
24 4.49E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.44E-06 EFWICC-XPP-FR04 XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
7.50E-02 OAESF3
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
25 4.45E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
6.32E-01 1HR_1-SUCCESS 1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
26 4.05E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.96E-03 DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
5.00E-01 EFW8P
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
27 3.36E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
2.83E-02 CNU_4 CORE IS UNCOVERED AT 4 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
28 3.13E-09 4.01E-05 %SSBI SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
5.20E-03 OAT2 OPERATOR FAILS TO TERMINATE SI GIVEN SSB
29 3.13E-09 4.01E-05 %SSBO SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
5.20E-03 OAT2 OPERATOR FAILS TO TERMINATE SI GIVEN SSB
30 3.06E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
6.32E-01 1HR_1-SUCCESS 1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
31 2.85E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
1.30E-03 D-AV-IFV2030FO AIR-OP FLOW CONTROL IFV-2030FAILS TO OPEN DUE TO LOCAL FLT
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
32 2.50E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.44E-06 EFWICC-XPP-FR04 XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
4.60E-04 Q-SIRWSTLOLOFA NO SAFGUARDS ACTUATION SIGNAL (RWST LO-LO LEVEL)
33 2.28E-09 1.52E-05 %SGR STEAM GENERATOR TUBE RUPTURE INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
7.50E-02 OAESF3
2.00E-03 UAAVIFV3531-FC FAILURE TO CLOSEIFV-3531 WHICH SUPPLIES MD HEADER FLOW

34 2.28E-09 1.52E-05 %SGR STEAM GENERATOR TUBE RUPTURE INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
7.50E-02 OAESF3
2.00E-03 UAAVIFV3536-FC FAILURE TO CLOSEIFV-3536 WHICH SUPPLIES TD HEADER FLOW
35 2.10E-09 4.01E-05 %SSBI SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
6.99E-04 MSIV-CC-FC2 XVM-2801A AND -2801B FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
36 2.10E-09 4.01E-05 %SSBI SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
6.99E-04 MSIV-CC-FC3 XVM-2801B AND -2801C FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
37 2.10E-09 4.01E-05 %SSBO SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
6.99E-04 MSIV-CC-FC1 XVM-2801A AND -2801B FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
38 2.10E-09 4.01E-05 %SSBO SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
6.99E-04 MSIV-CC-FC2 XVM-2801A AND -2801B FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
39 2.10E-09 4.01E-05 %SSBO SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
6.99E-04 MSIV-CC-FC3 XVM-2801B AND -2801C FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
40 1.82E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS
8.13E-03 DBPMI-XPP21BFR MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
2.50E-01 XPP21AB8
41 1.80E-09 4.01E-05 %SSBI SECONDARY SIDE BREAK INSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
5.98E-04 MSIV-CC-FC4 XVM-2801A, -2801B, AND -2801C FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
42 1.80E-09 4.01E-05 %SSBO SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
5.98E-04 MSIV-CC-FC4 XVM-2801A, -2801B, AND -2801C FAIL DUE TO COMMON CAUSE
7.50E-02 OAESF3
43 1.77E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
1.50E-01 OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)
44 1.77E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
1.30E-03 D-AV-IFV2030FO AIR-OP FLOW CONTROL IFV-2030FAILS TO OPEN DUE TO LOCAL FLT
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
45 1.62E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.40E-04 D-VLVMISPOS-HE HUMAN ERROR FAIL TO RESTORE VALVE SETTINGS AFTER TEST
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
46 1.48E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS

8.13E-03 DBPMI-XPP21BFR MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS
5.96E-03 DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
5.00E-01 EFW8P
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
5.00E-01 XPP21AB

47 1.48E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS
1.21E-05 EFWICC-XPP-FR03 XPP21B, XPP8 FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))

48 1.48E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DBPMI-XPP21BFR MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS
1.21E-05 EFWICC-XPP-FR02 XPP21A, XPP8 FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))

49 1.40E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS
1.15E-05 EFT-CC-SACV-FO4 SPRING ASSISTED CHECK VALVES XVC-1009A, -1009B, -1009C CCF TO OPEN
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))

50 1.40E-09 1.40E-09 %LLO LARGE LOCA INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)

51 1.38E-09 8.39E-07 %SLO SMALL LOCA INITIATING EVENT
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
1.50E-01 OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)

52 1.31E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
1.00E+00 ALLTRIPS-ATWS-PF INITIATORS THAT RESULT IN A PARTIAL FLOW ATWS
6.00E-04 D-MVSPURSIGNFA DURG TDP OPER SPUR SIGN TO ISLXTIE SUPPL STM CLOSE 2802A&B
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))

53 1.27E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
7.35E-03 DBPT----XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER

54 1.22E-09 4.01E-05 %SSBO SECONDARY SIDE BREAK OUTSIDE CONTAINMENT INITIATING EVENT
4.49E-03 EAAVXVM2801AFC FAILURE TO ISOL MS FLOW FROM SG A, XVM-2801A FAILS TO CLOSE
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
6.80E-03 UA-FTIFLTSGAHE HUMAN ERROR FAILURE TO CLOSEEFW FLOW VALVES TO STEAM GEN A

55 1.18E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-04 ACP-CC-BUSBK-FO 1 combination of 2 of 2 Bus feeder breakers
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS

56 1.10E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
1.50E-01 OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)

57 1.10E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS
9.01E-06 EFW-CC-XVC-F16 CV XVC 1014, 1016, 1013A, 1013B, 1048A, 1048B FAIL BY CCF

1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
58 1.02E-09 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-04 ACP-CC-BUSBK-FO 1 combination of 2 of 2 Bus feeder breakers
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
59 1.01E-09 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.40E-04 D-VLVMISPOS-HE HUMAN ERROR FAIL TO RESTORE VALVE SETTINGS AFTER TEST
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
60 9.65E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-04 ACP-CC-BUSBK-FO 1 combination of 2 of 2 Bus feeder breakers
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
61 8.85E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
7.50E-02 OAESF3
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
62 8.82E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
1.20E-07 ESF-CC-720BU-FOR 1 combination of 2 of 2 7200 VAC buses
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
63 8.73E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
7.35E-03 DBPT----XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
64 8.60E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
6.32E-01 1HR_1-SUCCESS 1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
2.90E-03 OAQ_1 OPERATOR FAILS TO RESTORE EQUIPMENT AFTER SBO & RECOVERY OF OFFSITE POWER
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
65 8.46E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.68E-03 BBPM--XPP39BFS LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
66 8.16E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
1.00E+00 ALLTRIPS-ATWS-PF INITIATORS THAT RESULT IN A PARTIAL FLOW ATWS
6.00E-04 D-MVSPURSIGNFA DURG TDP OPER SPUR SIGN TO ISLXTIE SUPPL STM CLOSE 2802A&B
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
67 7.28E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR

5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.68E-03 BBPM--XPP39BFS LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
68 7.18E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.96E-03 DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
5.00E-01 EFW8P
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
1.50E-01 OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)
69 7.18E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.96E-03 DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
1.46E-04 EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
7.50E-02 OAESF3
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
70 7.06E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-04 ACP-CC-BUSBK-FO 1 combination of 2 of 2 Bus feeder breakers
3.99E-01 CNU_1 CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
6.35E-01 XHR_1-SUCCESS XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS
71 6.90E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.68E-03 BBPM--XPP39BFS LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
72 6.89E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-03 AACB-----DGAFC DIESEL GENERATORBREAKER FAILS TOCLOSE
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
73 6.89E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-03 AACB--XSW1DAFO BUS XSW1DA FEEDER BREAKER FAILS TO OPEN
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
74 6.89E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB-----DGBFC DIESEL GENERATOR BREAKER FAILS TOCLOSE
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
75 6.89E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS

3.00E-03 ABCB--XSW1DBFO BUS XSW1DB FEEDER BREAKER FAILS TO OPEN
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
4.24E-01 XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
76 6.44E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS
1.30E-03 D-AV-IFV2030FO AIR-OP FLOW CONTROL IFV-2030FAILS TO OPEN DUE TO LOCAL FLT
8.13E-03 DBPMI-XPP21BFR MDP XPP-21B FAILS TO RUN DUE TO LOCAL FAULTS
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
5.00E-01 XPP21AB
77 5.93E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-03 AACB-----DGAFC DIESEL GENERATORBREAKER FAILS TOCLOSE
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
78 5.93E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-03 AACB--XSW1DAFO BUS XSW1DA FEEDER BREAKER FAILS TO OPEN
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
79 5.93E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB-----DGBFC DIESEL GENERATOR BREAKER FAILS TOCLOSE
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
80 5.93E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB--XSW1DBFO BUS XSW1DB FEEDER BREAKER FAILS TO OPEN
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
3.65E-01 XHR_1 FAILURE TO RECOVER OFFSITE POWER AT 14 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
81 5.92E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
6.32E-01 1HR_1-SUCCESS 1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
2.90E-03 OAQ_1 OPERATOR FAILS TO RESTORE EQUIPMENT AFTER SBO & RECOVERY OF OFFSITE POWER
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
82 5.71E-10 7.61E-09 %ISL INTERFACING SYSTEMS LOCA INITIATING EVENT
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
7.50E-02 OAESF3
83 5.63E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-03 AACB-----DGAFC DIESEL GENERATORBREAKER FAILS TOCLOSE
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
84 5.63E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
3.00E-03 AACB--XSW1DAFO BUS XSW1DA FEEDER BREAKER FAILS TO OPEN

5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
85 5.63E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB-----DGBFC DIESEL GENERATOR BREAKER FAILS TOCLOSE
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
86 5.63E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB--XSW1DBFO BUS XSW1DB FEEDER BREAKER FAILS TO OPEN
3.46E-01 CNU_2 CORE IS UNCOVERED AT 12 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
87 5.59E-10 5.59E-10 %VRP REACTOR VESSEL RUPTURE INITIATING EVENT
88 5.50E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E+00 ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
7.50E-02 OAESF3
1.10E-02 OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
89 5.44E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
5.44E-06 EFWICC-XPP-FR04 XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF
1.00E-04 F-CVXVC08926FO XVC-8926 FAILS TO OPEN
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
90 5.15E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
5.96E-03 DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
5.00E-01 EFW8P
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
91 5.05E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
1.83E-01 4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.68E-03 BBPM--XPP39BFS LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START
3.99E-01 CNU_1 CORE IS UNCOVERED AT 14 HOURS (WITH RCS COOLDOWN)
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
6.35E-01 XHR_1-SUCCESS XHR SUCCESS, POWER IS RECOVERED AT 14 HOURS
92 5.02E-10 1.00E+00 %EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
8.13E-03 DAPMI-XPP21AFR MDP XPP-21A FAILS TO RUN DUE TO LOCAL FAULTS
1.12E-03 DBPM--XPP21BFS MDP XPP-21B FAILS TO START DUE TO LOCAL FAULTS
7.35E-03 DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
5.00E-01 EFW21P
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.50E-02 OAR4 OPERATOR FAILS TO ALIGN HP CL RECIRC (Non- ISLOCA))
93 5.01E-10 1.38E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.68E-01 1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
2.90E-03 OAQ_1 OPERATOR FAILS TO RESTORE EQUIPMENT AFTER SBO & RECOVERY OF OFFSITE POWER
1.00E+00 SBO-FLAG STATION BLACKOUT SEQUENCE MARKER

94	4.94E-10	1.00E+00	%EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
		7.35E-03	DBPTI---XPP8FR TD PUMP FAILS TO RUN DUE TO MECHANICAL FAILURE
		1.46E-04	EFWICC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
		1.00E+00	NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
		4.60E-04	Q-SIRWSTLOLOFA NO SAFGUARDS ACTUATION SIGNAL (RWST LO-LO LEVEL)
95	4.89E-10	1.38E-04	%LSP LOSS OF OFFSITE POWER INITIATING EVENT
		6.32E-01	1HR_1-SUCCESS 1HR_1 SUCCESS, POWER IS RESTORED AT 1 HOUR
		5.83E-02	AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
		5.83E-02	ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
		1.10E-02	OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
		1.50E-01	OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)
		1.00E+00	SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
96	4.73E-10	1.38E-04	%LSP LOSS OF OFFSITE POWER INITIATING EVENT
		3.68E-01	1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
		1.83E-01	4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
		1.20E-04	ACP-CC-INV-FOR 1 combination of 2 of 2 Inverters
		1.00E+00	SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
		4.24E-01	XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
97	4.69E-10	1.00E+00	%EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
		5.44E-06	EFWICC-XPP-FR04 XPP21A, XPP21B, XPP8 FAIL TO RUN BY CCF
		8.62E-05	HPI-CC-PM43-FS 1 combination of 2 of 2 pumps
		1.00E+00	NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
98	4.57E-10	1.38E-04	%LSP LOSS OF OFFSITE POWER INITIATING EVENT
		3.68E-01	1HR_1 FAILURE TO RECOVER OFFSITE POWER WITHIN 1HR
		1.83E-01	4HR_1 FAILURE TO RECOVER OFFSITE POWER AT 4 HOURS GIVEN NO RECOVERY AT 1 HR
		1.16E-04	ACP-CC-DGOKR-FC 1 combination of 2 of 2 DG output breakers
		1.00E+00	SBO-FLAG STATION BLACKOUT SEQUENCE MARKER
		4.24E-01	XHR_2 FAILURE TO RECOVER OFFSITE POWER AT 12 HRS GIVEN NO RECOVERY AT 1 & 4 HRS
99	4.46E-10	1.00E+00	%EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
		5.96E-03	DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
		5.00E-01	EFW8P
		9.07E-05	EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
		1.00E+00	NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
		1.10E-02	OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs
		1.50E-01	OARC OPERATOR FAILS TO ALIGN & ESTABLISH CL RECIRC (CONDITIONAL)
100	4.46E-10	1.00E+00	%EFW1 LOSS OF DECAY HEAT REMOVAL (EFW) SPECIAL INITIATING EVENT
		5.96E-03	DBPT----XPP8FS TD PUMP FAILS TO START DUE TO MECHANICAL FAILURE
		9.07E-05	EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
		1.00E+00	ESF-CC-DRIC-FOR 1 combination of 2 of 2 cards (driver circuit)
		1.00E+00	NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
		7.50E-02	OAESF3
		1.10E-02	OAH_1 OPERATOR FAILS TO ALIGN CCW TO RHR HXs

Report Summary:

Filename: O:\RRA\WOG Projects\WOG Tech Spec Endstates\POS3 49 Hr Quantification\3I62POS3Rev1.CUT
Print date: 11/13/07 4:56 PM
Not sorted
Printed the first 100

Cutsets with Descriptions Report
@CDFALL = 4.03E-06
O:\RRA\WOG Projects\WOG Tech Spec Endstates\POS4 Files\POS 4 Rev 1\3I62POS4Rev1.CUT 11/13/07 5:01 PM

#	Cutset Prob	Event Prob	Event	Description
1	6.64E-07	1.00E+00	%LRHR	LOSS OF RHR INITIATING EVENT
		4.20E-03	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
		4.10E-03	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS
		1.30E-02	OAB2-HS	INITIATE FEED AND BLEED
		5.00E-01	OARHRREC	FAILURE TO RECOVER RHR
		4.24E-03	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
		1.40E+03	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
2	3.50E-07	1.40E-04	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4
		6.10E-01	POS4ISOLEAK	POS 4 LEAK NOT ISOLABLE
		4.10E-03	SIPOS4HE	OPERATOR ACTION TO ACTUATE HPSI
3	3.35E-07	1.97E-04	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT
		5.83E-02	AADG-----DGAFR	DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
		5.83E-02	ABDG-----DGBFR	DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
		1.00E+00	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER
		5.00E-01	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS
4	3.12E-07	4.20E-03	%DIL	BORON DILUTION EVENT IN POS 4
		3.10E-04	OAE_1	OPERATOR FAILS TO IMPLEMENT EMERGENCY BORATION
		5.00E-04	POS4OADILTM	OPERATOR FAILS TO TERMINATE DILUTION EVENT
		4.80E+02	OAE_1D	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO IMPLEMENT EMERGENCY BORATION
5	2.36E-07	1.00E+00	%LRHR	LOSS OF RHR INITIATING EVENT
		4.10E-03	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS
		1.30E-02	OAB2-HS	INITIATE FEED AND BLEED
		5.00E-01	OARHRREC	FAILURE TO RECOVER RHR
		2.27E-04	POS4CC-PM31-FR	RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT
		3.90E+01	OAB2-HS-D4	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
6	2.30E-07	1.97E-04	%LSP	LOSS OF OFFSITE POWER INITIATING EVENT
		2.34E-03	ACP-CC-DG-FR	1 combination of 2 of 2 DGs
		1.00E+00	NOSBO-FLAG	NO STATION BLACKOUT SEQUENCE MARKER
		5.00E-01	POS4PWRREC	POWER NOT RECOVERED IN 2 HOURS
7	2.12E-07	1.40E-04	%LOI	LOSS OF INVENTORY INITIATING EVENT FOR POS 4
		1.00E-02	POS4OAISOLEAK	OPERATOR ACTION TO ISOLATE LEAK
		4.10E-03	SIPOS4HE	OPERATOR ACTION TO ACTUATE HPSI
		3.70E+01	SIPOS4HE-D	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE HPSI
8	2.10E-07	1.00E+00	%LRHR	LOSS OF RHR INITIATING EVENT
		4.20E-03	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
		4.10E-03	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS
		2.30E-02	OAIA	OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
		5.00E-01	OARHRREC	FAILURE TO RECOVER RHR
		4.24E-03	POS4XPP0031AFR	RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
		2.50E+02	OAIA-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
9	9.65E-08	1.00E+00	%LRHR	LOSS OF RHR INITIATING EVENT
		6.16E-04	CAMVXVB9503AFO	MOTOR-OPERATED VALVE XVB-9503A FAILS TO OPEN
		4.20E-03	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
		4.10E-03	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS
		1.30E-02	OAB2-HS	INITIATE FEED AND BLEED
		5.00E-01	OARHRREC	FAILURE TO RECOVER RHR
		1.40E+03	OAB2-HS-D2	DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
10	9.65E-08	1.00E+00	%LRHR	LOSS OF RHR INITIATING EVENT
		4.20E-03	CBPM---XPP1BHE	OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
		4.10E-03	D-TRANOPSTRTHE	FAILURE OF OPERATOR TO START THE EFW PUMPS
		6.16E-04	IAMVXVG8811AFO	VALVE FAILS TO OPEN

1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
11 9.65E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IAMVXVG8812AFO VALVE FAILS TO OPEN
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
12 9.31E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.00E-01 OARHRREC-L RHR RECOVERY DURING BLEED AND FEED
2.27E-04 POS4CC-PM31-FR RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT
13 6.00E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
1.20E-07 ESF-CC-720BU-FOR 1 combination of 2 of 2 7200 VAC buses
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
14 5.89E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
3.76E-04 GAPMXPP0031AFS XPP-31A FAILS TO START
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
15 4.61E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO01 XVG8811A, XVG8811B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
16 4.61E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO03 XVG8811A, XVG8812B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
17 4.61E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO08 XVG8811B, XVG8812A FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
18 4.61E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO11 XVG8812A, XVG8812B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
19 3.32E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.27E-04 POS4CC-PM31-FR RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
20 3.16E-08 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
3.70E-04 OAR2 OPERATOR FAILS TO ALIGN FOR LP CL RECIRCULATION
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
21 3.05E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT

6.16E-04 CAMVXVB9503AFO MOTOR-OPERATED VALVE XVB-9503A FAILS TO OPEN
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
22 3.05E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IAMVXVG8811AFO VALVE FAILS TO OPEN
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
23 3.05E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IAMVXVG8812AFO VALVE FAILS TO OPEN
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
24 2.82E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO04 XVG8811A, XVG8811B, XVG8812A FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
25 2.82E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO05 XVG8811A, XVG8811B, XVG8812B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
26 2.82E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO06 XVG8811A, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
27 2.82E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO10 XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
28 2.47E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.38E-05 HPR-CC-PM31-FS 1 combination of 2 of 2 pumps
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
29 2.11E-08 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.68E-03 BBPM--XPP39BFS LOCAL FAULTS OF MDP XPP-39B CAUSE FAILURE TO START
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
30 1.86E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS

3.76E-04 GAPMXPP0031AFS XPP-31A FAILS TO START
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
31 1.48E-08 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.00E-04 ACP-CC-BUSBK-FO 1 combination of 2 of 2 Bus feeder breakers
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
5.00E-01 PWRRECD FAILURE TO RESET BREAKER
32 1.20E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
1.20E-07 ESF-CC-720BU-FOR 1 combination of 2 of 2 7200 VAC buses
1.00E-01 OARHRREC-L RHR RECOVERY DURING BLEED AND FEED
33 1.18E-08 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
1.20E-04 ACP-CC-INV-FOR 1 combination of 2 of 2 Inverters
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
34 1.06E-08 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.17E-03 OAH_1IE OPERATOR FAILS TO ALIGN CCW TO RHR HXs (NORMAL TRANSITION IN MODE 4)
1.00E-01 OARHRREC-L RHR RECOVERY DURING BLEED AND FEED
35 9.34E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
2.12E-03 POS4XPP0031BFR RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
36 8.62E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.00E-03 AACB-----DGAFC DIESEL GENERATORBREAKER FAILS TOCLOSE
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
5.00E-01 PWRRECD FAILURE TO RESET BREAKER
37 8.62E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
3.00E-03 AACB--XSW1DAFO BUS XSW1DA FEEDER BREAKER FAILS TO OPEN
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
5.00E-01 PWRRECD FAILURE TO RESET BREAKER
38 8.62E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB-----DGBFC DIESEL GENERATOR BREAKER FAILS TOCLOSE
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
5.00E-01 PWRRECD FAILURE TO RESET BREAKER
39 8.62E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
3.00E-03 ABCB--XSW1DBFO BUS XSW1DB FEEDER BREAKER FAILS TO OPEN
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
5.00E-01 PWRRECD FAILURE TO RESET BREAKER
40 8.54E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
1.00E-04 F-CVXVC08926FO XVC-8926 FAILS TO OPEN
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
41 8.27E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
1.44E-03 ABDG-----DGBFS DIESEL GENERATORFAILS TO START DUE TO RANDOM FAULTS
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER

5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
42 8.27E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
1.44E-03 AADG-----DGAFS DIESEL GENERATORFAILS TO START DUE TO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
43 7.52E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.80E-05 AACT-XSW1DA1OP 7200/480 VAC TRANSFORMER FAILS DURING OPERATION
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
44 7.52E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.80E-05 AACT-XSW1DA2OP 7200/480 VAC TRANSFORMER FAILS DURING OPERATION
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
45 7.36E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
8.62E-05 HPI-CC-PM43-FS 1 combination of 2 of 2 pumps
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
46 6.96E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO02 XVG8811A, XVG8812A FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
47 6.64E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
7.77E-05 HPR-CC-PM31-FR 1 combination of 2 of 2 pumps
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
48 6.49E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO01 XVG8811A, XVG8811B FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
49 6.49E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO03 XVG8811A, XVG8812B FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
50 6.49E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO08 XVG8811B, XVG8812A FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
51 6.49E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO11 XVG8812A, XVG8812B FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
52 5.71E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT

1.16E-04 ACP-CC-DGOKR-FC 1 combination of 2 of 2 DG output breakers
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
5.00E-01 PWRRECD FAILURE TO RESET BREAKER
53 5.49E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
5.28E-06 LPR-CCMVFO07 XVG8811A, XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
54 4.92E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.40E-06 ACP-CC-7248T-SO 1 combination of 2 of 2 7200/480 transformers
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
55 4.08E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
2.40E-06 ACP-CC-TRFM-FOR 1 combination of 2 of 2 transformers
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
56 3.96E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO04 XVG8811A, XVG8811B, XVG8812A FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
57 3.96E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO05 XVG8811A, XVG8811B, XVG8812B FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
58 3.96E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO06 XVG8811A, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
59 3.96E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.71E-05 LPR-CCMVFO10 XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
60 3.79E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
4.44E-05 LPR-CCMVFO01 XVG8811A, XVG8811B FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
61 3.79E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
4.44E-05 LPR-CCMVFO03 XVG8811A, XVG8812B FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
62 3.79E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
4.44E-05 LPR-CCMVFO08 XVG8811B, XVG8812A FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
63 3.79E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
4.44E-05 LPR-CCMVFO11 XVG8812A, XVG8812B FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
64 3.76E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.40E-05 AACBXMC1DA2YCO 480 VAC BREAKER TRANSFERS OPEN DURING OPERATION

4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
65 3.76E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.40E-05 AACB-XSW1DA1CO 480 VAC BREAKER TRANSFERS OPEN DURING OPERATION
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
66 3.76E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.40E-05 AACB-XSW1DA2CO 480 VAC BREAKER TRANSFERS OPEN DURING OPERATION
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
67 3.76E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.40E-05 AACB--XSW1EACO 7200 VAC BKR XSW1EA XFERS OPEN DURING OPS
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.40E+03 OAB2-HS-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
68 3.69E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.00E-01 OARHRREC-L RHR RECOVERY DURING BLEED AND FEED
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
2.12E-03 POS4XPP0031BFR RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT
69 3.54E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
6.16E-04 BBMVXVB3116BFO LOCAL FAULTS OF MOTOR-OPERATED VALVE XVB-3116B
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
70 3.48E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.38E-05 HPR-CC-PM31-FS 1 combination of 2 of 2 pumps
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.10E+00 OAIA-D3 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
71 2.81E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
2.40E-06 ACP-CC-TRFM-FOR 1 combination of 2 of 2 transformers
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
72 2.71E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
6.16E-04 CBMVXVB9503BFO MOTOR-OPERATED VALVE XVB-9503B FAILS TO OPEN
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
73 2.71E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IBMVXVG8811BFO VALVE FAILS TO OPEN
1.30E-02 OAB2-HS INITIATE FEED AND BLEED

5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
74 2.71E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IBMVXVG8812BFO VALVE FAILS TO OPEN
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
75 2.56E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
3.00E-05 HPR-CCMVFO03 XVG8706A, XVG8706B, XVG8885 FAIL BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
76 2.45E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
5.55E-04 CBPM---XPP1BFS PUMP XPP-1B FAILS TO START
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
77 2.38E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.80E-05 AACT-XSW1DA1OP 7200/480 VAC TRANSFORMER FAILS DURING OPERATION
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
78 2.38E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.80E-05 AACT-XSW1DA2OP 7200/480 VAC TRANSFORMER FAILS DURING OPERATION
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
79 2.31E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.71E-05 LPR-CCMVFO04 XVG8811A, XVG8811B, XVG8812A FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
80 2.31E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.71E-05 LPR-CCMVFO05 XVG8811A, XVG8811B, XVG8812B FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
81 2.31E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.71E-05 LPR-CCMVFO06 XVG8811A, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
82 2.31E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.71E-05 LPR-CCMVFO10 XVG8811B, XVG8812A, XVG8812B FAIL TO OPEN BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
83 2.20E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.20E-03 CBPM---XPP1BHE OPERATOR FAILS TO MANUALLY ACTUATE MDP XPP-1B
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
4.44E-05 LPR-CCMVFO02 XVG8811A, XVG8812A FAIL TO OPEN BY CCF
2.30E-02 OAIA OPERATOR FAILS TO CLEAR CONT ISOL AND RESTORE IAS TO PORV
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.50E+02 OAIA-D2 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO RESTORE IAS TO PORV
84 2.06E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
9.07E-05 EFW-CC-PM21-FS 1 combination of 2 of 2 MD pumps to start
1.00E-01 OARHRREC-L RHR RECOVERY DURING BLEED AND FEED
2.27E-04 POS4CC-PM31-FR RHR PUMPS XPP-31A & B CCF FAIL TO RUN FOR LOSS OF RHR INIT EVENT
85 2.04E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT

5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.20E-06 ESF-CC-720BK-SO 1 combination of 2 of 2 7200 VAC feed breakers
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
86 2.03E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.38E-05 HPR-CC-PM31-FS 1 combination of 2 of 2 pumps
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
87 1.94E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.27E-05 HPHLRCCMVFO07 XVG8884, XVG8886, XVG8706A, XVG8706B FAIL BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
88 1.94E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.27E-05 HPI-CCMVFO07 LCV115B, LCV115D, XVG8801A, XVG8801B FAIL BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
89 1.94E-09 1.40E-04 %LOI LOSS OF INVENTORY INITIATING EVENT FOR POS 4
2.27E-05 HPR-CCMVFC07 LCV115B, LCV115D, XVG8809A, XVG8809B FAIL BY CCF
6.10E-01 POS4ISOLEAK POS 4 LEAK NOT ISOLABLE
90 1.66E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
1.60E-06 SWS-CC-STR-BL04 TRAVELING SCREENS XRS-2A, 2B, 2C FAIL BY CCF
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
91 1.66E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
3.76E-04 GBPMXPP0031BFS XPP-31B FAILS TO START
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
92 1.64E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
1.58E-06 CCW-CCPMFR04 XPP1A, XPP1B, XPP1C FAIL TO RUN BY CCF
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
93 1.57E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
3.55E-04 CBPM---XPP1BFR PUMP XPP-1B FAILS TO RUN - RANDOM
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
4.24E-03 POS4XPP0031AFR RHR PUMP XPP-31A FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
94 1.40E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
2.34E-03 ACP-CC-DG-FR 1 combination of 2 of 2 DGs
1.20E-06 ESF-CC-720BK-SO 1 combination of 2 of 2 7200 VAC feed breakers
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
95 1.39E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
1.19E-05 EFW-CC-XPP-FR01 XPP21A, XPP21B FAIL TO RUN BY CCF
1.17E-03 OAH_1IE OPERATOR FAILS TO ALIGN CCW TO RHR HXs (NORMAL TRANSITION IN MODE 4)
1.00E-01 OARHRREC-L RHR RECOVERY DURING BLEED AND FEED
96 1.38E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
5.83E-02 AADG-----DGAFR DIESEL GENERATORFAILS TO RUN DUETO RANDOM FAULTS
2.40E-03 ABIV-XIT5903OP INVERTER XIT-5903 FAILS DURING OPERATION
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.00E-01 NVERT1
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS

97 1.38E-09 1.97E-04 %LSP LOSS OF OFFSITE POWER INITIATING EVENT
2.40E-03 AAIV-XIT5901OP INVERTER XIT-5901 FAILS DURING OPERATION
5.83E-02 ABDG-----DGBFR DIESEL GENERATOR FAILS TO RUN DUE TO RANDOM FAULTS
1.00E+00 NOSBO-FLAG NO STATION BLACKOUT SEQUENCE MARKER
1.00E-01 NVERT1
5.00E-01 POS4PWRREC POWER NOT RECOVERED IN 2 HOURS
98 1.36E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
6.16E-04 CAMVXVB9503AFO MOTOR-OPERATED VALVE XVB-9503A FAILS TO OPEN
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.12E-03 POS4XPP0031BFR RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
99 1.36E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IAMVXVG8811AFO VALVE FAILS TO OPEN
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.12E-03 POS4XPP0031BFR RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED
100 1.36E-09 1.00E+00 %LRHR LOSS OF RHR INITIATING EVENT
4.10E-03 D-TRANOPSTRTHE FAILURE OF OPERATOR TO START THE EFW PUMPS
6.16E-04 IAMVXVG8812AFO VALVE FAILS TO OPEN
1.30E-02 OAB2-HS INITIATE FEED AND BLEED
5.00E-01 OARHRREC FAILURE TO RECOVER RHR
2.12E-03 POS4XPP0031BFR RHR PUMP XPP-31B FAILS TO RUN FOR LOSS OF RHR INIT EVENT
3.90E+01 OAB2-HS-D4 DEPENDENT MULTIPLIER FOR OPERATOR FAILS TO INITIATE FEED AND BLEED

Report Summary:
Filename: O:\RRA\WOG Projects\WOG Tech Spec Endstates\POS4 Files\POS 4 Rev 1\3I62POS4Rev1.CUT
Print date: 11/13/07 5:01 PM
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~~APPENDIX B~~
~~MARKED-UP TECHNICAL SPECIFICATIONS AND BASES~~

NOTE

The Technical Specification and Bases markups were deleted from the WCAP. The markups will be included in the Technical Specification Task Force (TSTF) Traveler associated with this WCAP.

APPENDIX F RESPONSE TO A REQUEST FOR ADDITIONAL INFORMATION, DATED SEPTEMBER 20, 2009

Note that the date on the letter contained a typographical error and should have been November 20, 2009, the date of the actual transmittal.

NUCLEAR ENERGY INSTITUTE

Biff Bradley
DIRECTOR
RISK ASSESSMENT
NUCLEAR GENERATION DIVISION

September 20, 2009

Ms. Tanya M. Mensah
Senior Project Manager
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to a Request for Additional Information (RAI) on Technical Specifications and Bases Associated with PWROG Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs" (MD5134).

Project Number: 689

Dear Ms. Mensah:

Attached for NRC review is a response to an RAI regarding the Technical Specifications and Bases markups associated with WCAP-16294-NP. Attachment 1 includes the detailed response to the RAI received via e-mail on October 29, 2009. Attachment 2 contains the revision to the Technical Specification and Bases markups as discussed in the RAI response, while Attachment 3 contains a markup of the relevant portions WCAP-16294-NP, which reflect the change to the Completion Time for the Required Action to be in Mode 4.

We appreciate NRC staff's continued efforts to review this report. If you have any questions concerning these comments, please contact me at 202-739-8083; reb@nei.org or Victoria Anderson at 202-739-8101; vka@nei.org.

Sincerely,

Biff Bradley

Attachments

c: Mr. Carl S. Schulten, NRR/ADRO/DIRS/IT, NRC
Document Control Desk

Attachment 1

Response to RAI Received on October 29, 2009

REVIEW COMMENTS
APPENDIX B, "MARKED-UP TECHNICAL SPECIFICATIONS AND BASES

NUREG-1431, "Standard Technical Specifications, Westinghouse Plants" TS 3.6.6A, Containment Spray and Cooling Systems (Atmospheric and Dual); TS 3.6.6C, Containment Spray System (Ice Condenser); TS 3.6.6E, Recirculation Spray (RS) System (Subatmospheric); and TS 3.6.7, Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual) have Completion Times for Condition B.1, B.1, F.1 and B.1 (respectfully) requiring the unit to be placed in MODE 3 in 6 hours and have MODE 5 endstates for Condition B.2, B.2, F.2 and B.2 (respectfully) requiring the unit to be placed in MODE 5 in 84 hours. The MODE 5 completion time is described as an "extended completion" time because in accordance with LCO 3.0.3 when an LCO is not met and the associated ACTIONS direct placing the unit in MODE in which the LCO is not applicable TS [LCO 3.0.3 Action requirements] requirements specify 6 hours to reach MODE 3, 12 hours to reach MODE 4 and 36 hours to reach MODE 5.

Bases for the extended Completion Time (i.e., be in MODE 5) for each of the Containment Spray specifications is stated as: "The extended interval to reach MODE 5 allows additional time for attempting restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3."

Bases for the extended Completion Time (i.e., be in MODE 5) for the Spray Additive System (TS 3.6.7) is stated as: "The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System.

WCAP-16294 proposes to modify the MODE 5 endstate for the above TSs to be MODE 4 with a 60 hour Completion Time. WCAP-16294 did not provide analysis which justifies applying an extended Completion Time to the proposed MODE 4 modified endstate. Therefore, WCAP-16294, Appendix B should be revised be consistent with LCO 3.0.3 requirements by revising TS 3.6.6A, TS 3.6.6C, TS 3.6.6E, and TS 3.6.7 MODE 4 endstate Completion Times from 60 hours to 12 hours and make conforming changes to the Bases for these TS.

Response:
As discussed above, the Bases for TS 3.6.7, state: "The extended interval to reach MODE 5 allows 48 hours for restoration of the Spray Additive System in MODE 3 and 36 hours to reach MODE 5. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System." Therefore 48 hours is allowed to restore the Spray Additive System to Operable status in Mode 3, and is based on the

reduced pressure and temperature in Mode 3. Since the endstate of Required Action B.2 is being revised from Mode 5 to Mode 4, adding the time allowed to reach Mode 4 (6 hours) results in a total time allowed to reach Mode 4 of 54 hours. The Completion Time for Required Action B.2 was revised from 60 hours to 54 hours, and the Bases were also revised to reflect this change.

The same change was also made to the Completion Time for Required Action B.2 contained in TS 3.6.6A and 3.6.6C, and to Required Action F.2 contained in TS 3.6.6E. The Bases for these TS were also revised to be consistent with the changes made to the Bases for TS 3.6.7.

Attachment 1 contains revisions to these TS and Bases to reflect the changes discussed above.

Attachment 2 contains markups of WCAP-16294 pages 6-70, 6-74, 6-77, 6-79, and 6-80 that reflect the revision to the Completion Time for the Required Action to reach Mode 4 from 60 hours to 54 hours.

Attachment 2
Revisions to TS and Bases Markups

3.6 CONTAINMENT SYSTEMS

3.6.6A Containment Spray and Cooling Systems (Atmospheric and Dual) (Credit taken for iodine removal by the Containment Spray System)

LCO 3.6.6A Two containment spray trains and [two] containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable. -------------------NOTE------------------- LCO 3.0.4.a is not applicable when entering MODE 4 ~~from MODE 5~~. --	A.1 Restore containment spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE ~~5~~ 4.	6 hours 54 ~~84~~ hours
C. One [required] containment cooling train inoperable.	C.1 Restore [required] containment cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
D. Two [required] containment cooling trains inoperable.	D.1 Restore one [required] containment cooling train to OPERABLE status.	72 hours

3.6 CONTAINMENT SYSTEMS

3.6.6C Containment Spray System (Ice Condenser)

LCO 3.6.6C Two containment spray trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

-----NOTE-----
LCO 3.0.4.a is not applicable when entering MODE 4. ~~from MODE 5~~.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 LCO 3.0.4.a is not applicable when entering MODE 4. Be in MODE ~~5~~ 4.	6 hours ~~60~~ 54 ~~84~~ hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6C.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6C.2	Verify each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.6E Recirculation Spray (RS) System (Subatmospheric)

LCO 3.6.6E Four RS subsystems [and a casing cooling tank] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

---------------------NOTE---------------------
LCO 3.0.4.a is not applicable when entering MODE 4 ~~from MODE 5.~~

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RS subsystem inoperable.	A.1 Restore RS subsystem to OPERABLE status.	7 days
B. Two RS subsystems inoperable in one train.	B.1 Restore one RS subsystem to OPERABLE status.	72 hours
C. [Two inside RS subsystems inoperable.	C.1 Restore one RS subsystem to OPERABLE status.	72 hours]
D. [Two outside RS subsystems inoperable.	D.1 Restore one RS subsystem to OPERABLE status.	72 hours]
E. [Casing cooling tank inoperable.	E.1 Restore casing cooling tank to OPERABLE status.	72 hours]
F. Required Action and associated Completion Time not met.	F.1 Be in MODE 3. AND F.2 Be in MODE ~~5~~ 4.	6 hours ~~84~~ 54 hours
G. Three or more RS subsystems inoperable.	G.1 Enter LCO 3.0.3.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.7 Spray Additive System (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

LCO 3.6.7 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

------NOTE------
LCO 3.0.4.a is not applicable when entering MODE 4 ~~from MODE 5~~.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE ~~5~~ 4.	6 hours ~~60~~ 54 ~~84~~ hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Verify each spray additive manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.7.2	Verify spray additive tank solution volume is ≥ [2568] gal and ≤ [4000] gal.	184 days
SR 3.6.7.3	Verify spray additive tank [NaOH] solution concentration is ≥ [30]% and ≤ [32]% by weight.	184 days

BASES

ACTIONS (continued)

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, "Completion Times," for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

B.1 and B.2

to OPERABLE status in MODE 3.

Insert 1

If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must ~~be brought to a MODE in which the LCO does not apply.~~ To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ 54 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows ~~additional time for attempting restoration of~~ 48 hours to restore the containment spray train ~~and~~ is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3. Insert 2 This

Insert 3A

C.1

With one of the required containment cooling trains inoperable, the inoperable required containment cooling train must be restored to OPERABLE status within 7 days. The components in this degraded condition provide iodine removal capabilities and are capable of providing at least 100% of the heat removal needs. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action C.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this Specification coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

BASES

APPLICABILITY

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

In MODES 5 and 6, the probability and consequences of these events are reduced because of the pressure and temperature limitations of these MODES. Thus, the Containment Spray System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

With one containment spray train inoperable, the affected train must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal and iodine removal needs after an accident. The 72 hour Completion Time was developed taking into account the redundant heat removal and iodine removal capabilities afforded by the OPERABLE train and the low probability of a DBA occurring during this period.

B.1 and B.2

Insert 13

If the affected containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the ~~LCO does not apply~~. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ ~~60~~ 54 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows ~~additional time and~~ is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3. Insert 2

This Insert 3A

48 hours to restore the containment spray train to OPERABLE status in MODE 3.

SURVEILLANCE REQUIREMENTS

SR 3.6.6C.1

Verifying the correct alignment of manual, power operated, and automatic valves, excluding check valves, in the Containment Spray System provides assurance that the proper flow path exists for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since they were verified in the correct position prior to being secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned, are in the correct position.

BASES

ACTIONS (continued)

[C.1

With two inside RS subsystems inoperable, at least one of the inoperable subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.]

[D.1

With two outside RS subsystems inoperable, at least one of the inoperable subsystems must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.]

[E.1

With the casing cooling tank inoperable, the NPSH available to the outside RS subsystem pumps may not be sufficient. The inoperable casing cooling tank must be restored to OPERABLE status within 72 hours. The components in this degraded condition are capable of providing 100% of the heat removal needs after an accident. The 72 hour Completion Time was chosen based on the same reasons as given in Required Action B.1.]

F.1 and F.2

Insert 9

If the inoperable RS subsystem(s) [or the casing cooling tank] cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the ~~LCO does not apply~~. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ 54 ~~60~~ hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows ~~additional time and~~ is reasonable considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3. Insert 2

This

Insert 3

48 hours to restore the RS subsystem(s) [or casing cooling tank] to OPERABLE ~~status~~ in MODE 3.

BASES

LCO (continued)

spray flow until the Containment Spray System suction path is switched from the RWST to the containment sump, and to raise the average spray solution pH to a level conducive to iodine removal, namely, to between [7.2 and 11.0]. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODE 5 or 6.

ACTIONS

A.1

If the Spray Additive System is inoperable, it must be restored to OPERABLE within 72 hours. The pH adjustment of the Containment Spray System flow for corrosion protection and iodine removal enhancement is reduced in this condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst case DBA occurring during this period.

B.1 and B.2

Insert 11

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the ~~LCO does not apply~~. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE ~~5~~ 4 within ~~84~~ ~~60~~ 54 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE ~~5~~ 4 allows 48 hours ~~for restoration of~~ to restore the Spray

BASES

ACTIONS (continued)

TO OPERABLE STATUS

Additive System in MODE 3 ~~and 36 hours to reach MODE 5~~. This is reasonable when considering the reduced pressure and temperature conditions in MODE 3 for the release of radioactive material from the Reactor Coolant System. Insert 2

Insert 3A

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Verifying the correct alignment of Spray Additive System manual, power operated, and automatic valves in the spray additive flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves outside containment and capable of potentially being mispositioned are in the correct position.

SR 3.6.7.2

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the RWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency was developed based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, so that there is high confidence that a substantial change in level would be detected.

SR 3.6.7.3

This SR provides verification of the NaOH concentration in the spray additive tank and is sufficient to ensure that the spray solution being injected into containment is at the correct pH level. The 184 day Frequency is sufficient to ensure that the concentration level of NaOH in the spray additive tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.