August 25, 2006

Mr. Gordon Bischoff, Manager
PWR Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR BAW-2441, REVISION 2, "RISK INFORMED JUSTIFICATION FOR LCO END-STATE CHANGES" (TAC NO. MC6241)

Dear Mr. Bischoff:

By letter dated January 19, 2004, the Babcock and Wilcox Owners Group (B&WOG), now known as Pressurized Water Reactor (PWR) Owners Group, submitted Topical Report (TR) BAW-2441, Revision 2, "Risk Informed Justification for LCO End-State Changes," to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval.

By letter dated May 11, 2006, an NRC draft safety evaluation (SE) regarding our approval of TR BAW-2441, Revision 2, was provided to the PWR Owners Group for review and comments. The PWR Owners Group commented on the draft SE in a letter dated June 28, 2006. The NRC staff agrees with the PWR Owners Group comments and the modifications as discussed in the letter have been made to the enclosed final SE. Therefore, details of the NRC staff's disposition of the PWR Owners Group's comments are not discussed in the final SE.

The NRC staff has found that TR BAW-2441, Revision 2, is acceptable for referencing in licensing applications 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 the PWR Owners Group publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

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

Sincerely,

/RA by JClifford for/

Ho K. Nieh, Acting Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 693

Enclosure: Final SE

cc w/encl:

Mr. Ronnie L. Gardner, Manager
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AREVA NP
3315 Old Forest Road
Lynchburg, VA 24501

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BAW-2441, REVISION 2, "RISK INFORMED JUSTIFICATION FOR LCO END-STATE CHANGES"

PWR OWNERS GROUP (FORMERLY B&W OWNERS GROUP)

PROJECT NO. 693

1.0 INTRODUCTION AND BACKGROUND

By letter dated January 19, 2004, B&W Owners Group (B&WOG), now known as the Pressurized Water Reactor (PWR) Owners Group, submitted topical report BAW-2441, Revision 2, “Risk Informed Justification For LCO End-State Changes,” (Ref. 1) for review by the NRC staff. The B&WOG supplemented the topical report by letter dated February 17, 2005 (Ref. 2), in response to the NRC staff request for additional information.

The purpose of BAW-2441 is to make changes to the end states of selected limiting conditions for operation (LCOs) from Mode 5, cold shutdown, to Mode 4, hot shutdown, to (1) reduce risk associated with unnecessary shutdown cooling (SDC) operations, and (2) reduce plant unavailability associated with reduced plant downtime caused by unnecessary cooldown to Mode 5 and subsequent reheat to Mode 3 or 4. With a condition not meeting an LCO, and the associated required action not met within the specified completion time (CT) from a Mode 1 condition, the B&WOG Standard Technical Specifications (STSs) generally require actions that result in Mode 3 entry in 6 hours and in a Mode 5 entry within 36 hours. BAW-2441 generally retains the B&WOG STS time philosophy in establishing the required mode entry times except that the end state of Mode 5 within 36 hours is replaced with Mode 4 within 12 hours.

BAW-2441 is similar to the report that the NRC staff approved for Combustion Engineering Owners Group (CEOG) PWRs (CE-NPSD-1186, Revision 00, “Technical Justification for the Risk-Informed Modification to Selected Required Action End States for CEOG Member PWRs,” July 17, 2001) and the report that the NRC staff approved for boiling water reactors (BWRs) (NEDC-32988, Revision 2, “Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants,” January 5, 2001).

To justify the proposed end-state change, the topical report provides a qualitative assessment and a quantitative analysis to confirm that Mode 4 is the preferred end state from a risk and operational perspective. The qualitative assessment describes the risk associated with operation in Mode 4 compared to operation in Mode 5, and is intended to justify that the end state of Mode 4, versus Mode 5, for the proposed LCO conditions invoked is acceptable. The qualitative assessment also concludes that the increment of risk associated with unnecessary SDC can be removed from the overall plant risk as a result of making the proposed LCO end-state changes.

BAW-2441 compared the core damage frequencies during the two modes of operation using the probabilistic safety assessment (PSA) for a typical B&W-designed plant, assuming the inoperable conditions specified in STSs. Important insights were also obtained from the assessment of the applicability of the representative B&W plant results to other B&WOG plants, through sensitivity studies accounting for design and operational differences and/or direct comparison of features using risk insights for the representative B&W plant. In addition to quantitative analysis, BAW-2441 evaluated the two modes of operation based on defense-in-depth considerations and then proposed a list of end-state changes.

BAW-2441 request would allow a Mode 4 end-state, rather than a Mode 5 end-state, for the selected LCOs listed in Table 1.

Table 1 LCOs Proposed for End-State Change

LCO	**CURRENT END-STATE**	**PROPOSED END-STATE**
3.3.5 Engineered Safety Features Actuation System (ESFAS) Instruments	5	4
3.3.6 ESFAS Manual Initiation	5	4
3.4.6 Reactor Coolant System (RCS) Loops Mode 4	5	4
3.4.15 RCS Leak Detection Instrumentation	5	4
3.5.4 Borated Water Storage Tank Boron Concentration	5	4
3.6.1 Containment	5	4
3.6.2 Containment Air Locks	5	4
3.6.3 Containment Isolation Valves	5	4
3.6.4 Containment Pressure	5	4
3.6.5 Containment Air Temperature	5	4
3.6.6 Containment Spray and Cooling	5	4
3.7.7 Component Cooling Water (CCW) System	5	4
3.7.8 Service Water System (SWS)	5	4
3.7.9 Ultimate Heat Sink	5	4
3.7.10 Control Room Emergency Ventilation System	5	4
3.7.11 Control Room Emergency Air Temperature System	5	4
3.8.1 Alternating Current (AC) Sources - Operating	5	4

3.8.4 Direct Current (DC) Sources - Operating	5	4
3.8.7 Inverters - Operating	5	4
3.8.9 Distribution System - Operating	5	4

2.0 REGULATORY EVALUATION

LCOs are the lowest functional capability or performance levels of equipment required for safe operation of a facility. The regulation at 50.36(c)(2), "Limiting conditions for operation," of Title 10 of the *Code of Federal Regulations* (10 CFR) specifies that "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." The regulation does not specify the shutdown conditions a plant must enter. For each LCO that is not met, the plant Technical Specification (TS) Required Action provides a CT for remedial actions to meet the LCO. If the LCO or remedial action cannot be met, then the reactor is required to be shutdown within a specified time. When the individual plant TSs were written, the shutdown conditions or end states specified were usually Mode 5, cold shutdown. This was based on the perception that putting a plant in Mode 5, cold shutdown, would result in the safest condition. However, preliminary risk and operational considerations have indicated that end-state modifications could be beneficial. For example, establishing Mode 4, hot shutdown, instead of Mode 5 as the end-state for several TS action statements could reduce operational costs without compromising safety and may enhance safety.

The proposed amendment to change the end state from Mode 5 to Mode 4 for failure to meet selected LCOs is based on risk-informed analysis with consideration of risk associated with the transition from Mode 4 to Mode 5. The risk analysis results demonstrated that Mode 4 end state is a preferred end state over Mode 5. Designation of Mode 4, hot shutdown, end state continues to comply with 10 CFR 50.36(c)(2)

3.0 EVALUATION OF ENGINEERING ASSESSMENTS

Section 4.4 of BAW-2441 provides the bases for each of the proposed LCO end-state changes. The bases are in general based on the argument discussed below.

Section 4.0 of BAW-2441 provides a qualitative engineering assessment to justify the proposed end state of Mode 4, hot shutdown, compared to current Mode 5, cold shutdown. The Mode 5 end state was generally based on the idea that the lower the RCS pressure, the lower the risk associated with operating with a given LCO in effect. However, various events occurring during shutdown operations have resulted in the NRC and industry assessing the risk associated with various modes of operation.

Section 4.2 of BAW-2441 provides a qualitative assessment of a general risk comparison between Modes 3, 4, and 5, and concludes there are some risk advantages associated with operations in Modes 3 and 4 compared to Mode 5 (and a portion of Mode 4 with SDC) operations. The characteristics that differentiate the plant operations between Mode 4 and Mode 5 is the core cooling mechanism. In Mode 4 heat removal continues through the steam generator (SG) until transition into the SDC systems at lower RCS temperature and pressure, whereas in Mode 5, SDC systems are used. The transition process required to go from Mode 4 to Mode 5 presents increased risk due to SDC system vulnerabilities and fewer mitigating systems being immediately available. In Mode 5, there is less equipment available because plant realignments lead to general unavailability of the emergency feedwater (EFW) and high pressure injection (HPI) systems. The transition from Mode 4 to Mode 5 also exposes the plant to several potential failure modes, including inadvertent closure of SDC system valves, inadvertent diversion of reactor coolant through the SDC system, and loss of reactor coolant through the SDC relief valves or other leak paths. With the SDC system being aligned to the RCS to cool the core, core cooling can be terminated with the inadvertent closure of a single valve in the SDC system suction line. There is also a possibility of inadvertent RCS draining caused by inappropriate valve alignments and/or maintenance activities while on SDC. Therefore, remaining in Mode 4 on SG cooling has the advantage of increased redundancy and diversity of mitigating systems, as well as the avoidance of human error during SDC initiation and operation, loss of SDC system, and inadvertent RCS draining.

3.1 ESFAS Instrumentation

The ESFAS provides an automatic actuation of the engineered features that are required for mitigation of design-basis accidents (DBAs), especially the loss-of-coolant accident (LOCA) and steam line break events. The ESFAS actuates the following systems: HPI, low-pressure injection (LPI), reactor building (RB) cooling, RB spray, RB isolation, and onsite standby power source start. The ESFAS also provides a signal to EFW initiation and control (EFIC) system and initiates EFW, when HPI is initiated.

The ESFAS consists of two trains. Each train consists of three protection channels. Each channel provides input to logic circuits that initiate equipment with a two-out-of-three logic on each component. Four parameters are used by the ESFAS for actuation: Low RCS pressure, Low Low RCS pressure, High RB pressure, and High High RB pressure.

TS LCO 3.3.5 specifies that three channels of ESFAS instrumentation for each parameter in TS Table 3.3.5-1 (i.e., Low RCS pressure, Low Low RCS pressure, High RB pressure, and High High RB pressure setpoints) shall be OPERABLE in each ESFAS train. For Condition B with two or more channels of one or more actuation parameters inoperable, or one channel inoperable and required action and associated CT time not met, Required Action B requires an initial cooldown to Mode 3 within 6 hours. If the inoperable channels are that of the High RB Pressure setpoint or High High RB Pressure setpoints, Required Action B.2.3 further requires the plant to be in the end state of Mode 5 within 36 hours. The proposed change would change the Required Action B.2.3 end state to be in Mode 4 within 12 hours.

BAW-2441 provides a qualitative assessment for changing the end state from Mode 5 to Mode 4. When operating in Mode 4, the reactor is in a shutdown and subcritical condition, and there is no power generation except for decay heat. The RCS thermal-hydraulic conditions are very different from those associated with a DBA occurring at power. In Mode 4 the RCS temperature is less than 350 ^{o}F, the RCS pressure is reduced, and the likelihood of an initiating event occurring is greatly reduced. Also, loss of core cooling and loss of inventory events are characterized by lower initial fuel temperatures and a lower decay heat generation rate because of the time elapsed since power operation. Therefore, the transient will proceed more slowly and with reduced challenges to the reactor and containment systems than those associated with at-power conditions. These characteristics enhance both a reduced likelihood of events and the ability to respond to events should they occur.

When operating in Mode 4 there are more mitigation systems (e.g., HPI and EFW) available to respond to initiating events that could challenge RCS inventory or decay heat removal than when operating in Mode 5. In addition, all redundant functions initiated by the ESFAS can be manually initiated to mitigate transients that will proceed more slowly and with reduced challenges to the reactor and containment systems than those associated with at-power conditions.

Figure 2 of BAW-2441 also shows that the risk is lower operating in Mode 4 with the SG in operation than Mode 5 with SDC operation. Therefore, the NRC staff concludes that the proposed end state of Mode 4 for Required Action B.2.3 is acceptable.

3.2 ESFAS Manual Initiation

TS LCO 3.3.6 specifies that two manual initiation channels of each one of the ESFAS functions (HPI, LPI, RB cooling, RB spray, RB isolation, and control room isolation) shall be OPERABLE during MODES 1, 2, and 3, and 4 when associated engineered safeguard equipment is required to be OPERABLE. If one or more ESFAS functions with one channel inoperable and the required action (i.e., restoration to OPERABLE status) and associated CT are not met (Condition B), Required Action B requires the plant to be place in Mode 3 within 6 hours and in Mode 5 within 36 hours. The proposed change would change the end state from Mode 5 in 36 hours to Mode 4 within 12 hours.

The ESFAS manual initiation capability allows for the operator to actuate ESFAS functions from the main control room (MCR) in the absence of any other initiation condition. This manual initiation capability is provided as a backup to automatic trip function, in the event that the operator determines that an ESFAS function is needed and has not been automatically actuated. Furthermore, this capability allows operators to rapidly initiate ESFAS functions if the trend of unit parameters indicates that ESFAS actuation will be needed.

The ESFAS manual initiation function relies on the operability of the automatic actuation logic for each component to perform the actuation of the systems. The ESFAS manual initiation channel is defined as the instrumentation between the console switch and the automatic actuation logic that actuates the end device. Other means of manual initiation, such as controls for individual ESFAS devices, may be available in the control room and other unit locations. These alternative means are not required by, nor credited to fulfill the requirements of this LCO.

In Mode 4, the RCS temperature is less than 350 °F at low RCS pressure, the loss of cooling and loss of inventory events are characterized by lower initial fuel temperatures and a lower decay heat generation rate because of the time elapsed since power operation. There are also more mitigation systems (e.g., HPI and EFW) available to respond to initiating events that could challenge RCS inventory or decay heat removal than when operating in Mode 5. In addition, all redundant functions initiated by the ESFAS can be manually initiated via individual component controls. Therefore, should an initiating event occur, the transient will proceed slowly, which provides the plant operator with adequate time to respond to the challenges to the reactor and containment systems. Also, when operating in Mode 4 with the SG in operation and the SDC system not in operation, the risk associated with SDC is avoided.

In addition, as a backup to the automatic actuation of the ESFAS, the manual initiation channels of an ESFAS function are used only in the event of the failure of the ESFAS instrumentation. In any event, there are also the manual initiation of individual ESFAS devices to provide backup to the manual initiation channels. Based on this multiple redundancy in the ESFAS initiation, the adequate response time should an initiation event occur in Mode 4, and the avoidance of risk associated with SDC operation, the NRC staff concludes that the proposed change of end state to Mode 4 is acceptable.

3.3 RCS Loops - MODE 4

LCO 3.4.6 specifies that two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be operable and one loop shall be in operation during Mode 4.

With one required loop inoperable (Condition A), Required Action A.1 requires restoration of a second loop to an operable status immediately. If the remaining operable loop is a DHR loop, Required Action A.2 requires the plant to be in Mode 5 within 24 hours. The proposed change would delete Required Action A.2 so as to allow continued operations in Mode 4.

In Mode 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to the SGs or DHR heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The purpose of this LCO is to require that two loops, either RCS or DHR, be operable in Mode 4 and one of these loops be in operation. Any one loop in operation provides enough flow to remove decay heat from the core with forced circulation. The second loop that is required to be operable provides a redundant path for heat removal. With one required operable loop inoperable, redundancy for heat removal is lost. Required Action A.1 specifies that action be initiated immediately to restore the inoperable loop to an operable status.

In the case when the inoperable loop is the RCS, the DHR loop is operable and operating, with sufficient capability for core decay heat removal. Should the operating DHR loop fail and the inoperability of the RCS loop is due to inoperable reactor coolant pump (RCP), the SGs are still available as a heat sink to provide for core heat removal through natural circulation, which, in Mode 4 can be effective for heat removal with steaming. The availability of SGs for heat removal is ensured by LCOs 3.7.4, 3.7.5, and 3.7.6, that, respectively, require that the atmospheric vent valves, EFW system, and the condensate storage tank, be operable while in Modes 1, 2, and 3, and in Mode 4 when SGs are relied upon for heat removal.

The failure of the only operable and operating DHR loop in Condition A would result in Condition B with inoperability of two required operable loops or no required loop in operation. Required Action B allows the plant to remain in MODE 4 and relies on natural circulation for decay heat removal, while only requiring (B.1) suspension of operations that would cause introduction of water into the RCS coolant with boron concentration less than that required to meet the shutdown margin (SDM) requirement of LCO 3.1.1, and (B.2) initiation of action to restore one loop to operable status and be operating, immediately. Therefore, the proposed change to allow the plant to remain in Mode 4 with one DHR loop operable and operating is justified by the required action for Condition B. If the plant is placed in Mode 5 (with 24 hours as required by Action A.2), and the operating DHR loop fails, use of natural circulation would not be effective. The RCS must be heated up to the temperature region of Mode 4 where steaming can be effective for heat removal. Also, in Mode 5, the RCP would not be operating even if restored to operable status because of net positive suction head (NPSH) considerations. Figure 2 of BAW-2441 also shows that the risk is lower operating in Mode 4 with the SG in operation than Mode 5 with SDC operation. Therefore, the NRC staff agrees with the proposed amendment to delete Required Action A.2 so that the plant would be allowed to remain in Mode 4.

3.4 Borated Water Storage Tank (BWST)

LCO 3.5.4 specifies that the BWST shall be operable, including maintaining boron concentration, water temperature, and water volume within limits, during Modes 1, 2, 3, and 4 operation. With the BWST boron concentration, water temperature, and water volume not within limits, and failure to restore the BWST to comply with the limits within the specified CTs, the required action requires that the plant be in Mode 3 within 6 hours and in Mode 5 within 36 hours. The proposed change would rearrange the existing Actions Statements by separating the conditions and required actions for exceeding the boron concentration and the temperature limits, and change the end state for failure to restore the boron concentration to within the limit within 8 hours from being in Mode 5 in 36 hours to being in Mode 4 within 12 hours. No change is made regarding the end state for failure to comply with the water temperature and volume limits.

The BWST provides a source of borated water to HPI, LPI, and RB spray pumps during accident conditions. The LCO specifies the limits on the BWST boron concentration, water volume, and water temperature to ensure that the BWST contains sufficient borated water to support the emergency core cooling system (ECCS) for core cooling and to maintain SDM. There are two limits for boron concentration in the BWST. The minimum and maximum BWST boron concentration limits, respectively, are established to (1) ensure the reactor will be maintained in a cold shutdown condition following a postulated LOCA; and (2) avoid the potential boron precipitation in the core resulting from reactor coolant boil off during the long-term cooling period following a LOCA that could result in flow channel blockage.

Upon entering into Mode 3, the core is subcritical with all rods inserted and the reactor is maintained shutdown by operating procedures and other administrative controls. Hence, in the highly unlikely event of a LBLOCA occurring while in Mode 4, all control rods will be inserted. This provides for the reactor SDM to be very conservative (the applicant indicated excess of approximately -9.0 percent $\Delta k/k$). If the boron concentration in the BWST is below the minimum boron concentration limit, the excessive SDM provided by the control rods being inserted provides margin to compensate for injected ECCS water should a LOCA occur in Mode 4. Also, deviations in boron concentration are likely to be relatively slow and small, and the boric acid addition systems would normally be available.

If the BWST boron concentration exceeds the maximum concentration limit, the concern would be boron precipitation during long-term cooling following a LOCA. However, due to low power levels associated with Mode 4, there will be ample time to establish boron dilution flow paths should the need arise. Post LOCA emergency procedures direct the operator to establish dilution flow paths in the LPI system to prevent this condition by establishing a forced flow path through the core regardless of break location. By utilizing these procedures following a LOCA, potential boric acid precipitation from the core could be avoided.

Since (1) the need for a large volume of water from the BWST in Mode 4 is due to low likelihood of LOCA events; (2) the anticipated deviations in the BWST boric acid concentrations are expected to be small; and (3) the ability to correct this deficiency is expected to be readily available, the requested change to have a Mode 4 end state would have an insignificant impact on safety, and is therefore acceptable to the NRC staff.

4.0 EVALUATION OF RISK ASSESSMENTS

BAW-2441 documents a risk-informed analysis of the proposed TS change. Probabilistic risk assessment (PRA) results and insights are used, in combination with the results of deterministic assessments, to identify and propose changes in "end-states" for all B&W plants. This is in accordance with guidance provided in Regulatory Guides (RGs) 1.174 and 1.177. The three-tiered approach, documented in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," was followed. The first tier of the three-tiered approach includes the assessment of the risk-impact of the proposed change for comparison to acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174, entitled "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." In addition, the first tier aims at ensuring that there are no unacceptable temporary risk increases during the implementation of the proposed TS change, such as when equipment is taken out of service. The second tier addresses the need to preclude potentially high risk configurations which could result if equipment is taken out of service concurrently with the implementation of the proposed TS change. The third tier addresses the application of 10 CFR 50.65(a)(4) of the Maintenance Rule for identifying risk-significant configurations, resulting from maintenance or other operational activities, and taking appropriate compensatory measures to avoid such configurations. The scope of the topical report and the NRC staff evaluation was limited to identifying changes in end-state conditions that excluded continued power operation as an acceptable end-state, regardless of the risk.

The risk assessment approach followed by BAW-2441 includes the following tasks:

- Performance of a generic qualitative risk assessment,

- Performance of a quantitative risk assessment for a pilot plant that includes the following:

  - Comparison of baseline risks between Modes 4 and 5 (i.e., risks when no equipment outages are assumed),

  - Comparison of configuration-specific risks between Modes 4 and 5 (i.e., risks when certain equipment is assumed to be unavailable),

- Performance of sensitivity studies to investigate the robustness of the results to uncertainties in data and modeling assumptions, and

- Performance of sensitivity studies to ensure that the conclusions of the quantitative assessment for the pilot plant apply also to other B&W plants.

- Use of risk insights, derived from the qualitative and quantitative generic risk assessments, in the individual TS assessments supporting each of the proposed end-state changes.

The objective of the generic qualitative risk assessment is to show that the proposed TS end-state changes result in an increase in defense-in-depth for expected initiating events. This is achieved by performing qualitative risk comparisons between cold shutdown (Mode 5) and hot shutdown (Mode 4). Such comparisons include risk parameters, such as initiating events and mitigating systems, associated with each critical safety function (e.g., reactivity control and core decay heat removal) at the various B&W plants. The objectives of the quantitative risk assessment are: (1) to substantiate the conclusion of the qualitative risk assessment by providing numerical results for a representative plant, (2) to investigate the robustness of the results regarding uncertainties in data and modeling assumptions through sensitivity analyses, and (3) to assess the applicability of the results to other B&W plants through sensitivity analyses accounting for design and operational differences. In addition, specific risk assessments were also performed for several of the proposed TS end-state changes to ensure that the specific condition causing the LCO does not increase the risk when the proposed new end-state is implemented. Finally, an integrated discussion of the risk-significance and defense-in-depth considerations is provided (using risk insights from both the qualitative and quantitative risk assessments) for each proposed TS end-state change. This discussion provides useful information that can be used by individual licensees applying for such TS changes to develop guidance in appropriate plant procedures and/or administrative controls to ensure that risk-significant plant configurations are avoided. The NRC staff's review finds that the BAW-2441 risk assessment approach is comprehensive and follows staff guidance as documented in RGs 1.174 and 1.177.

4.1 Evaluation of the Quality of the Risk Assessment

The risk impact of the proposed end-state changes was assessed subject to the following major general assumptions:

- The request is to allow Mode 4 (hot shutdown) as the end-state for all of the selected TS action statements, instead of Mode 5 (cold shutdown). However, licensees will still have the option to take the plant to Mode 5 (cold shutdown) to complete maintenance.

- An important difference between Mode 4 and Mode 5 as end-states is the transition in the mode of core cooling. In Mode 4, heat removal continues through the SGs, while Mode 5 requires the initiation of the SDC system, a transition that exposes the plant to several potential failure modes.

- Entry into the shutdown mode under consideration is for a short interval, with the primary intent being to repair a nonfunctional component and return the plant to power as soon as practical.

- The RCS remains at its nominal inventory and the RCS boundary strength is not compromised (e.g., via installation of nozzle dams).

The NRC staff finds that these assumptions adequately represent the proposed changes and can be used in PRA models to compare risks between Mode 4 and Mode 5 associated with the "short" duration repairs (i.e., repairs needed to correct the initiating condition and return to power as soon as is practical). This comparison can be made by considering only steady state risks because transition risks, as discussed later in this SE, are about equal for the two end-states, or favor Mode 4 as the end-state.

The quality of the risk assessment is a very important part of any risk-informed license amendment review. In this case, both the qualitative and quantitative risk assessments must be of adequate quality and completeness to support their intended purposes. Regarding the qualitative risk assessment, the comparisons between current and proposed end-states for the various B&WOG plants must be of adequate quality and completeness to ensure confidence in the robustness of the conclusion that the proposed TS end-state changes result in an increase in defense-in-depth for expected initiating events, and that all expected initiating events were addressed in the analysis. Regarding the quantitative risk assessment, the various models (including assumptions and data) and sensitivity studies must be of adequate quality and completeness (e.g., with respect to initiating events and failure modes of the various safety systems) to provide confidence in the robustness of the conclusion that the risk will not increase if the proposed new TS end-states are approved and implemented. The NRC staff's evaluation of the qualitative and quantitative risk assessments are documented in Sections 4.2 and 4.3, respectively, of this SE.

4.2 Qualitative Risk Assessment

The qualitative risk assessment is a comparison between operation in Modes 4 and 5 at the various B&WOG plants. This comparison, which assesses qualitatively the means that exist at each B&WOG plant to maintain critical safety functions for expected initiating events, contains the following three parts:

- Assessment of critical safety functions at shutdown,

- Generic comparison of risks at shutdown, and

- Comparison of safety and operational features at shutdown among B&WOG plants.

Several critical safety functions at shutdown (reactivity control, RCS inventory control, core decay heat removal, containment integrity control, and power availability) were identified based on insights from previous risk studies. The means utilized at the B&WOG plants to perform each of the critical functions during Mode 4 (hot shutdown) and Mode 5 (cold shutdown) are discussed and used in the generic (i.e., without reference to a specific plant) comparison of risks.

In the generic comparison of risks at shutdown, Mode 4 and Mode 5 risks are qualitatively compared to each other by discussing the likelihood of the various initiating events and the availability of mitigating systems at each plant operating condition. This generic comparison of risks is complemented by a comparison of safety and operational features among B&WOG plants. Such a comparison is needed in order to ensure that the conclusions of the generic qualitative risk assessment are valid for each specific B&WOG plant.

The NRC staff finds that the qualitative risk assessment is of adequate quality and completeness to support a conclusion that the proposed TS end-state changes do not decrease defense-in-depth based on examination of the following:

- Challenges and mitigating capabilities of B&WOG plants and comparison between current and proposed end-states;

- Documentation of the various design and operational features used to mitigate shutdown accidents at B&WOG plants; and

- Proper use of results and insights from previous deterministic and probabilistic studies.

4.3 Quantitative Risk Assessment

A quantitative risk assessment of current and proposed end-states (corresponding to shutdown Modes 5 and 4, respectively) was performed for a B&WOG plant (Davis-Besse). The scope was to provide a comparison of the risks associated with either staying in Mode 4, or going to Mode 5 to carry out equipment repair. Variability in safety and operational features among B&WOG plants was addressed by a series of direct comparisons of features as well as by sensitivity studies to ensure that the conclusions of the quantitative assessment for Davis-Besse apply to all B&WOG plants.

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

- Initiating events, accidents sequences, and failures found to be significant contributors to shutdown risk in previous studies have been addressed;

- Important assumptions made and data used are reasonable;

- Important uncertainties in data and modeling assumptions were identified, and appropriate sensitivity studies were performed in order to provide confidence in the conclusions regarding the proposed TS end-states; and

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

The quantitative risk analysis was performed using PRA models of Mode 4 (on both SG and SDC) and Mode 5 for internal initiating events of the representative plant (Davis-Besse). The Davis-Besse non-power PRA models evolved from generic shutdown and transition templates, which were developed in a cooperative effort by the B&WOG (BAW-2393, “Generic Template

for Shutdown Risk Assessment," Framatome ANP, November 2001 and BAW-2415, "Generic Method to Assess Transition Risk," Framatome ANP, December 2001) to provide "top logic" that is consistent across all B&WOG plants. These generic templates were subsequently adapted by each B&WOG plant for plant-specific operational and design differences, and are used for outage risk management. The Davis-Besse models are representative for the B&WOG plants because of their generic origin. Furthermore, the use of the Davis-Besse risk estimates to compare Mode 4 to Mode 5 risks is conservative.
This conservatism stems primarily from the following plant-specific differences between Davis-Besse and other B&WOG plants:

- Davis-Besse has two safety-related turbine-driven emergency feedwater (TDEFW) pumps, and no safety-related motor-driven emergency feedwater (MDEFW) pumps. All other B&WOG plants have at least one MDEFW pump. The availability of an MDEFW pump results in a better reliability of the EFW system at very low pressures, as may be the case during plant operation in Mode 4 with SG cooling.

- At Davis-Besse, the DHR suction line valves must be opened during plant operation in Mode 4, even when cooling is provided by the SGs. The reason for this alignment is to provide sufficient relief capacity for low-temperature overpressure protection (LTOP) considerations, which rely on a DHR system relief valve located on the DHR suction line. Therefore, the Davis-Besse PRA includes a higher frequency of SDC-related loss of inventory events associated with Mode 4 on SG cooling than do other B&WOG plants.

In developing the Davis-Besse models for the various shutdown modes, several initiating events applicable to the shutdown modes of interest were considered and appropriate accident sequence models were developed. Such initiating events were selected from a broad list of postulated initiating events by screening out those events that either do not apply at shutdown, or are not risk-significant based on previous PRA insights. The success criteria for the various safety functions were derived from the full power PRA, after accounting for the reduced decay heat levels in the shutdown modes. The PRA models were quantified for Modes 3, 4, and 5 base cases (i.e., assuming no equipment outages), as well as for several other cases reflecting the LCO conditions for which an end-state change is requested. The Mode 4 (on SG cooling) and Mode 5 core damage frequency (CDF) results were used to identify important risk contributors and to investigate the sensitivity of the risk assessment results to important uncertainties in data and modeling assumptions.

The quantitative risk assessment does not include risks from external events (dominated by internal fires, internal floods and seismic events), risks associated with transitions from one mode of operation to another, or risks in terms of large early release frequency (LERF). The following qualitative arguments are made to justify not assessing such risks:

- Risks associated with external events are smaller when Mode 4 instead of Mode 5 is selected as the end-state for the following reasons:

  - Seismic events, which are equally likely in either mode, have a larger impact on the plant's accident mitigation capability during Mode 5 than during Mode 4. Although during both modes of operation there are an adequate number of seismically-designed safety systems available to mitigate accidents (e.g., EFW, DHR, ECCS, cooling water systems, and onsite standby power sources), there are more systems available during Mode 4 on SG cooling than during Mode 5. In addition, since a seismic event is very likely to result in an unrecoverable loss of offsite power event, the plant's ability to prevent core damage is higher in Mode 4 (on SG cooling) due to the availability of the TDEFW pumps.

- Internal fire and flood events are equally likely to occur during Mode 4 or Mode 5, during either mode the same fire or flood event would impact the same equipment, most likely equipment located in the affected fire or flood zone. Because there are more systems available for accident mitigation in Mode 4 than in Mode 5, the plant's ability to prevent core damage is at least as good in Mode 4 as is in Mode 5.

- The only transition risk that needs to be considered in the comparison of risks between the proposed and the current end-states is the risk associated with the transition from SG cooling to SDC, using the DHR system, which occurs in Mode 4. This risk is primarily due to the likelihood of a drain-down event while the DHR valves are being aligned for SDC. This transition risk is most likely avoided when Mode 4 on SG cooling, instead of Mode 4 on SDC or Mode 5, is selected as the end-state for short duration repairs. Therefore, there is no need to assess such a risk because it supports the position that it is safer to stay in Mode 4 rather than go to Mode 5. It should be noted that for Davis-Besse there is no realignment risk associated with the transition from SG cooling to SDC because the DHR suction line valves must be opened during plant operation in Mode 4 even when cooling is provided by the SGs.

- During power operation, LERF are the result of: (1) energetic containment failure due to a high pressure core melt, (2) a containment bypass event, and (3) a core damage event occurring in combination with a non-isolated containment. Compared to power operation, Mode 4 or Mode 5 operation is associated with lower initial energy level, reduced fission product inventory level, and reduced decay heat load. Due to the combined effect of these factors, the likelihood of LERFs in Modes 4 and 5 is very low. These factors serve to provide time for the operator to respond to serious plant upsets and, consequently, they contribute to delaying the core melt progression and reducing radiation releases. Therefore, any potential increase due to changing the end-state is negligible.

BAW-2441 identified several areas of uncertainty, in both data and modeling assumptions, associated with the shutdown models that could have an impact on results and conclusions, including the following:

- Accident initiating event frequencies used in the risk analysis;
- Recovery probabilities used in the risk analysis; and
- Common cause failure probabilities used in the risk analysis.

The identified areas of uncertainty were evaluated to determine how they impact the results and conclusions of the quantitative risk assessment. Major risk insights from this evaluation, which included, whenever necessary, the performance of sensitivity studies, are documented in

Sections 4.0 and 5.0 of this SE.

BAW-2441 identified several important design and operational differences between the various B&WOG plants and the analyzed plant (Davis-Besse) used in the quantitative risk assessment. The risk impact of such differences was investigated by a sensitivity study using a generic “non-Davis-Besse” PRA model, which was developed by modifying the Davis-Besse PRA model to account for the identified design and operational differences between Davis-Besse and other B&WOG plants. The purpose of the investigation was to extend the results and conclusions of the quantitative risk assessment performed for Davis-Besse to other B&WOG product lines beyond the analyzed plant. Some major design and operational differences that were investigated are:

- Davis-Besse has separate makeup and HPI pumps, while the other B&WOG plants have combined makeup and HPI pumps. These separate pumps at Davis-Besse provide an extra measure of redundancy for “feed-and-bleed” not available at the other B&WOG plants.

- The SDC system is not aligned when the plant operates in Mode 4 on SG cooling, except for Davis-Besse. At Davis-Besse the SDC system is aligned even when the plant operates in Mode 4 on SG cooling. At Davis-Besse, this alignment is needed to implement LTOP control because the LTOP control valve is located on the DHR suction line.

- At all B&WOG plants, except for Davis-Basse, the LTOP control is an integral part of the RCS because it is implemented through the use of the power-operated relief valves (PORVs). During plant cooldown, procedures require that the PORV is reset from its normal operational value to implement LTOP control. Therefore, there are design and operational differences among plants in the means used to depressurize the RCS to initiate SDC in the case of a total loss of feedwater event.

- All EFW pumps at Davis-Besse are TDEFW pumps, while other B&WOG plants have a combination of MDEFW pumps and TDEFW pumps.

- While at Davis-Besse there are non-safety MDEFW pumps, startup feed pumps and auxiliary boilers available, this is not the case at all other B&WOG plants.

- At Davis Besse, the SG loops are raised in comparison to other B&WOG plants.

- There is variability among B&WOG plants regarding support systems. Important differences are in the number and type of emergency onsite power sources, electrical divisions, and service water loops.

The identified design and operational differences were evaluated to determine how they impact the results and conclusions of the quantitative risk assessment performed for Davis-Besse. Major risk insights from qualitative and quantitative risk evaluations are documented in Sections 4.4 and 4.5 of this SE.

The NRC staff concludes that the quality of the quantitative risk assessment, including the sensitivity studies performed to address uncertainties and differences among plants, are adequate to show that there are no significant risk increases associated with the proposed TS end-state changes for B&WOG plants.

4.4 Risk Insights from the Qualitative Risk Assessment

BAW-2441 documents a generic qualitative comparison of shutdown risks in Modes 4 and 5 that aims to show that the proposed TS end-state changes do not decrease defense-in-depth. Mode 4 and Mode 5 risks are qualitatively compared by discussing the means used to address critical functions and the availability of systems needed to mitigate likely accident initiating events. This generic risk comparison is complemented by a comparison of safety and operational features among B&WOG plants, a comparison needed to ensure that the conclusions of the generic qualitative risk assessment are valid for each B&WOG plant. It should be noted that the qualitative comparison of risks is based on a plant configuration that does not include any additional outages for maintenance beyond what is associated with the subject LCO. Comparison of risks between Modes 4 and 5 when specific maintenance outages are taking place are part of the quantitative risk assessment discussed in Section 4.5 of this SE.

Important insights regarding the various means used to accomplish critical functions and mitigate accidents occurring in Modes 4 and 5 are listed below:

- The means used to achieve reactivity control, containment integrity control, and power availability are approximately equally reliable in Modes 4 and 5. Furthermore, their reliability is not altered by invoking any of the LCO proposed for change.

- More means are available to achieve inventory control when the plant is operating in Mode 4 than when the plant is operating in Mode 5. In Mode 4, two trains of HPI and two trains of LPI are either immediately available, via automatic means, or can be placed in operation, via operator action from the control room. In Mode 5, HPI may not be available and one LPI train is unavailable because it is aligned for SDC. Even though breaks and SG tube rupture events are unlikely in Mode 5, the availability of inventory control systems is important to mitigate inadvertent RCS draining events, which are more likely during plant alignment for SDC and in Mode 5 operation than in other modes of operation.

- More means are available to perform the core decay heat removal critical function while the plant is operating in Mode 4 on SG cooling than when it is aligned to the SDC system (either in Mode 4 or in Mode 5) for decay heat removal. In Mode 4 (on SG cooling), in addition to the main feedwater system, the condensate system and the EFW system can be used to remove heat from the reactor core. In the unlikely event of a total loss of feedwater, there are reliable means (e.g., use of PORVs and pressurizer vents) to depressurize the RCS and initiate SDC (at Davis-Besse this is not necessary because the plant is aligned to the SDC system for LTOP pressure control). In Mode 4 (on SDC) or in Mode 5 operation, heat is removed by either one of the two trains of the DHR system, which provides the SDC function. However, closure of a single valve in the SDC suction line will terminate core cooling. Also, when the SDC system is being aligned to the RCS, there is a possibility of inadvertent RCS draining caused by inappropriate valve alignments, as there is a possibility of draining while on SDC due to human errors during maintenance activities. Although it is possible to return to SG cooling (in either the forced or natural circulation mode), the ability to immediately use the SGs for core heat removal decreases as cooldown progresses and it may take some time to re-establish RC conditions that support SG cooling. Some of the reasons are: (1) the long time that may be required to restart the RCPs for forced circulation, (2) the potential unavailability of the SGs when the plant is in Mode 5, and (3) the significant core heatup necessary to develop appropriate hydraulic heads for natural circulation.

Potentially significant accident initiating events at shutdown and available mitigating systems were evaluated to establish the acceptability of Mode 4 (on SG cooling) end-state as the default action for the identified TSs. Important insights are:

- All potentially risk significant initiating events that can occur while the plant is operating at shutdown Mode 4 (on either SG cooling or SDC) and Mode 5, are those associated with insufficient removal of decay heat and insufficient inventory.

- In Mode 4 (on SG cooling), initiating events causing insufficient decay heat removal via the SGs or insufficient inventory are represented (or subsumed) by the following:

  - Loss of feedwater;
  - Loss of offsite power (LOOP);
  - Loss of one or more power buses;
  - Loss of cooling water;
  - Loss of instrument air;
  - Floods in pump rooms
  - Loss of inventory outside the RB, when on SDC or going on SDC (Davis Besse); and
  - Loss of inventory inside the RB.

- In Mode 5 and 4 (on SDC), initiating events causing insufficient decay heat removal via the DHR system or insufficient inventory are represented (or subsumed) by the following:

  - Loss of running DHR train;
  - LOOP;
  - Loss of one or more power buses;
  - Loss of cooling water (either component water or SWS);
  - Floods in rooms where the DHR pumps, the CCW, or the SWS pumps are located;
  - Loss of inventory outside the RB; and
  - Loss of inventory inside the RB.

- The risk impact of LOCAs, as pressure driven initiating events, are not as significant in Modes 4 and 5 as they are in Mode 1. The major contributor to this initiator is loss of inventory caused by incorrect valve lineups. Since incorrect valve lineups are more likely during Mode 5 operation, the risk associated with LOCAs will be smaller if Mode 4 (on SG cooling) is adopted as the end-state.

- LOOP is an important initiating event in both Modes 4 and 5 with approximately the same frequency. Therefore, their risk impact is lower when there is more redundancy and diversity of the mitigating systems, as is the case when the plant is operating in Mode 4 (on SG cooling).

- Loss of feedwater in Mode 4 (on SG cooling) and loss of the operating DHR train in the SDC mode in Modes 4 and 5 are important initiating events of the same order of magnitude frequency. Since there is much more redundancy and diversity of the mitigating systems when the plant is operating in Mode 4 on SG cooling, the risk impact associated with the loss of feedwater initiating event (occurring in Mode 4 on SG cooling) is lower than the risk impact associated with the loss of the operating DHR train initiating event (occurring in Mode 4 on SDC and in Mode 5).

- Loss of cooling water is an equally important initiating event in both Modes 4 and 5 with approximately the same frequency and risk impacts.

A comparison of risk important safety and operational features among B&WOG plants was made to show that the conclusions of the generic qualitative risk assessment are valid for each of the B&WOG plants. The differences in risk-important safety and operational features among B&WOG plants, discussed in Section 4.0 of this SE, do not change the conclusions of the qualitative risk assessment in favor of establishing Mode 4 (on SG cooling) as the preferred end-state for the following reasons:

- Although there are some differences among B&WOG plants regarding the means used for inventory makeup and heat removal at high pressures, all B&WOG plants have such features. Therefore, the conclusion that more means are available to perform the core decay heat removal critical function when the plant operates in Mode 4 on SG cooling than when the plant is aligned to the SDC system (in either Mode 4 or Mode 5), is valid for any plant.

- Although there are some differences in the means available to depressurize the RCS among the various B&WOG plants, the conclusion that for accidents initiated in Mode 4 (on SG cooling) the reactor can be depressurized reliably so that SDC can be used, is valid for any plant.

- Although there are some differences among B&WOG plants regarding the means used for inventory makeup and heat removal at low pressures, these differences do not change any conclusions because they impact Mode 4 and Mode 5 risks at a specific plant equally; this also true for differences among B&WOG plants regarding support systems.

The above listed insights lead to the conclusion that, in general, plant operation in Mode 4 on SG cooling (hot shutdown) offers at least the same robustness to plant upsets as operation in Mode 5 (cold shutdown). The insights gained from the quantitative risk study (listed below) substantiate this conclusion.

4.5 Risk Insights from the Quantitative Risk Assessment

The scope of the quantitative risk assessment was to compare the core damage risks associated with either staying in Mode 4 (on SG cooling), or going to Mode 5 to carry out equipment repairs. This comparison was made for each of the LCO cases for which an end-state change is proposed, and for which the equipment of interest is modeled in the PRA as well as for the non-LCO case (base case). The results are summarized in Table 2 of this SE. "LCO-specific" quantitative risk assessments were not performed for some of the proposed LCO cases for which an end-state change is proposed, because they have a negligible or intangible contribution to CDF (e.g., LCOs involving boron concentration and containment). For each of the cases for which "LCO-specific" quantitative risk assessments were performed, CDF values were assessed for both the current end-state (i.e., Mode 5) and the proposed end-state (i.e., Mode 4 on SG cooling). It should be noted that the assessed CDF values are yearly values (i.e., they are an estimate of the risk associated with plant operation at the current and proposed end-states for an entire year). In addition to these two CDF values, the percent change (always a reduction) in CDF due to changing the end-state from Mode 5 to Mode 4 is also listed in Table 2 for each of the analyzed cases.

Important results and insights from the quantitative risk assessment, which substantiate the conclusions of the qualitative risk assessment by providing numerical results, are listed below:

- The CDF estimates, reported in Table 2 of this SE, support the requested end-state change. These estimates show that staying in Mode 4 (on SG cooling), rather than going to Mode 5 to carry out equipment repairs, does not have any adverse effect on plant risk and may actually lead to significant risk reduction. This conclusion is supported by the following:

  - When no equipment is taken out (base case), the Mode 5 CDF is about 1.2E-5/year, while the Mode 4 (on SG cooling) CDF is 3.4E-6/year (an approximately 71 percent reduction).

  - When equipment associated with the proposed changes is taken out of service, the Mode 4 (on SG cooling) CDF is lower than the Mode 5 CDF, ranging from a reduction of about 6 percent for LCO 3.4.6 (one RCS loop inoperable) to a reduction of about 92 percent for LCO 3.8.9 (ac distribution subsystem inoperable). This indicates that, for outages involving the LCOs proposed for end-state change, the end-state change may lead to significant risk reductions.

- The accident sequences that dominate the risk in Mode 4 (on SG cooling) are initiated by a LOOP event, with subsequent failure of onsite standby power sources causing loss of all primary and backup core cooling options (i.e., SG cooling, DHR system cooling, and "feed-and-bleed" cooling). Other accident sequences that are significant contributors to risk in Mode 4 (on SG cooling) are initiated by a loss of RCS inventory outside of the RB, with subsequent failure of the operator to take action to stop the drain before DHR suction (the backup cooling method) is lost, and failure to initiate "feed-and-bleed" cooling.

- The major contributing accident sequences to the risk in Mode 5 and 4 (on SDC) are initiated by loss of the operating DHR train and LOOP events. In addition, there is significant contribution from accident sequences initiated by a loss of CCW event, which affects the function of the DHR and other systems, and, to a lesser extent, by accident sequences initiated by a loss of RCS inventory outside of the RB event.

Table 2 Comparison of CDF Between Mode 4 and Mode 5 End-States (Davis-Besse Model) for Proposed Changes.

Technical Specification	Condition	Mode 5 CDF/yr	Mode 4 CDF/yr	Decrease in CDF/yr (percent)
Base Case	No LCO	1.2E-5	3.4E-6	71
3.7.7 (CCW)	A. One CCW train inoperable	4.8E-4	7.6E-5	84
3.7.8 (SWS)	A. One SWS train inoperable	4.8E–5	5.6E-6	88
3.8.1 (AC Sources)	A. One offsite circuit inoperable B. One emergency diesel generator (EDG) inoperable C. Two offsite circuits inoperable D. One offsite circuit and one EDG inoperable E. Two EDGs inoperable	1.1E-5 1.1E-4 8.7E-4 9.8E-5 1.2E-3	1.4E-6 9.2E-5 1.1E-4 8.9E-5 1.1E-3	87 16 87 9 8
3.8.4 (DC Sources)	A. One train battery chargers inoperable B. Batteries on one train inoperable C. One DC subsystem inoperable -other	1.5 E-5 1.6E-4 1.4E-4	3.6E-6 4.8E-5 4.6E-5	76 70 67
3.8.9 (AC/DC Distribution)	A. AC distribution subsystem inoperable B. AC vital bus inoperable C. DC distribution subsystem inoperable	1.1E-3 5.6E-5 1.4E-4	8.6E-5 4.4E-5 4.6E-5	92 21 67
3.8.7 (Inverters)	A. One Inverter inoperable	5.6E-5	4.4E-5	21
3.4.6 (RCS loops Mode 4)	A. One RCS loop inoperable	4.8E-3	4.5E-3	6

- The dominant contributors to risk when the plant is in the proposed end-state (Mode 4 on SG cooling), are associated with failures of redundant or diverse means of performing a safety function, such as failures that affect normal cooling (feedwater) and also backup and emergency cooling methods. The most common reasons for these failures are LOOP and failure of the EDGs, which affects SG cooling as well as backup DHR and “feed-and-bleed” cooling. When the reason for being in the LCO is inoperability of one train of a safety system (such as EDG, batteries, CCW, or SWS), then common cause failure of the remaining train(s) is usually an important contributor to risk. In general, failures that dominate the risk are associated with equipment that has already been recognized as an important contributor by the TS, and for which operational requirements and guidance (e.g., compensatory measures) are in place. In addition, implementation guidance for the proposed end-state changes should be developed to ensure that insights and assumptions made in the risk assessment are properly reflected in the plant-specific configuration risk management program (CRMP) at the participating B&WOG plants.

- The conclusion that staying in Mode 4 (on SG cooling), rather than going to Mode 5 to carry out equipment repairs, does not have any adverse effect on plant risk and may actually lead to significant risk reduction, can be extended to the proposed LCO cases for which an end-state change is proposed without performing “LCO-specific” quantitative risk assessments. Although no “LCO-specific” quantitative risk assessments were performed for LCO cases having a negligible or intangible contribution to CDF (such as LCOs involving boron concentration or containment), the results of the non-LCO cases (base cases) and the insights from the qualitative risk assessments support this conclusion.

Based on the results of the quantitative risk assessment for Davis-Besse, one can conclude that, in the analyzed cases, it is safer to stay in Mode 4 (on SG cooling) than to go to Mode 5 (cold shutdown) to carry out equipment repair. This conclusion has been extended to all other B&WOG plants through a PRA sensitivity study, which accounts for the pertinent differences between Davis-Besse and the other plants. For this sensitivity study, a “non-Davis-Besse” PRA model was developed and used to re-quantify the risk associated with Mode 4 and 5 end-states. The “non-Davis-Besse” PRA model was developed by identifying the important differences in design and operational features among B&WOG plants, and by changing the Davis-Besse model to reflect these differences. Conservative or bounding assumptions were made, as necessary, so that the “non-Davis-Besse” PRA model could be used to extend the conclusion reached for Davis-Besse to all other B&WOG plants. In addition, the robustness of such a conclusion has been investigated by performing sensitivity studies to assess the impact of uncertainties in data and modeling assumptions.

The “non-Davis-Besse” PRA model was developed from the Davis-Besse PRA model by making the following changes, which reflect important differences in design and operational features between Davis-Besse and other B&WOG plants, that can have a significant impact on the results and conclusions of the risk assessment:

- Deleted separate makeup pumps and added combined makeup and HPI pumps.

- Reduced the Mode 4 (on SG cooling) initiating event frequency for loss of RCS inventory outside of the RB because all B&WOG plants with the exception of Davis-Besse do not align to the SDC for LTOP in Mode 4 while the SG are used for core cooling.

- Replaced one of TDEFW pumps with a MDEFW pump.

- Deleted credit for the auxiliary boiler.

- Added a model for the means to depressurize the RCS in order to initiate SDC for the case of total loss of feedwater (conservatively assumed that only the PORVs can be used to depressurize the RCS).

- Deleted the non-safety-related MDEFW pumps, startup feedwater pumps, and auxiliary boiler available at Davis-Besse but not at all other B&WOG plants.

- Accounted for the difference between the raised loop and the lowered-loop SG on the human recovery model.

The “non-Davis-Besse” PRA model was used to quantify and compare the core damage risks associated with either staying in Mode 4 (on SG cooling), or going to Mode 5 to carry out equipment repairs. This comparison was made for each of the LCO cases for which an end-state change is proposed, and for which the equipment of interest are modeled in the PRA as well as for the non-LCO case (base case). The results are summarized in Table 3 of this SE. Important insights from the assessment of the applicability of the Davis-Besse results to other B&WOG plants are listed below:

- Any changes in CDF that result from such design and operational differences would not impact the conclusion reached for Davis-Besse regarding the proposed TS end-state change. Therefore, the conclusion that staying in Mode 4 (on SG cooling), rather than going to Mode 5 to carry out equipment repairs, does not have any adverse effect on plant risk and may actually lead to significant risk reduction, is valid for all B&WOG plants. This finding is supported by the following:

  - When no equipment is taken out (base case), the Mode 5 CDF is about 5.2E-5/year while the Mode 4 (on SG cooling) CDF is 1.4E-5/year (an approximately 73 percent reduction).

  - When equipment associated with the proposed changes is taken out of service, the Mode 4 (on SG cooling) CDF is lower than the Mode 5 CDF, ranging from a reduction of about 3 percent for LCO 3.4.6 (one RCS loop inoperable) to a reduction of about 95 percent for LCO 3.7.7 (one CCW train inoperable). This indicates that, for outages involving the LCOs proposed for end-state change, the change would not increase risk but may lead to significant risk reductions.

- The accident sequences that dominate the risk in Mode 4 (on SG cooling) are similar to the Davis-Besse case. LOOP initiated accident sequences, with subsequent failure of onsite standby power sources causing loss of all primary and backup core cooling options (i.e., SG cooling, DHR system cooling, and “feed-and-bleed” cooling), continue to be major contributors to risk. Accident sequences involving loss of RCS inventory outside RB are larger contributors to risk than they are at Davis-Besse, due to the lower probability of successful recovery because of the smaller SG inventory available to drain back into the RCS in a lowered-loop plant. Also, loss of feedwater accident sequences are more important than they are at Davis-Besse because there is no credit for the backup non-safety feedwater pump and the auxiliary boiler in the "non-Davis-Besse" model.

Table 3 Comparison of Core Damage Frequency Between Mode 4 and Mode 5 End States (Non-Davis-Besse Model) for Proposed Changes.

Technical Specification	Condition	Mode 5 CDF/yr	Mode 4 CDF/yr	Decrease in CDF/yr (percent)
Base Case	No LCO	5.2E-5	1.4E-5	73
3.7.7 (CCW)	A. One CCW train inoperable	3.7E-3	1.7E-4	95
3.7.8 (SWS)	A. One SWS train inoperable	8.8E–5	3.0E-5	66
3.8.1 (AC Sources)	A. One offsite circuit inoperable B. One EDG inoperable C. Two offsite circuits inoperable D. One offsite circuit and one EDG inoperable E. Two EDGs inoperable	6.4E-5 1.4E-4 6.0E-4 1.5E-4 1.2E-3	1.2E-5 1.0E-4 4.0E-5 1.0E-4 1.1E-3	81 29 93 33 8
3.8.4 (DC Sources)	A. One train battery charger inoperable B. Batteries on one train inoperable C. One DC subsystem inoperable-other	6.3 E-5 2.1E-4 1.9E-4	1.4E-5 1.3E-4 1.1E-4	78 38 42
3.8.9 (AC/DC Distribution)	A. AC distribution subsystem inoperable B. AC vital bus inoperable C. DC distribution subsystem inoperable	2.7E-3 9.4E-5 1.9E-4	2.4E-3 7.1E-5 1.1E-4	11 24 42
3.8.7 (Inverters)	A. One inverter inoperable	9.4E-5	7.1E-5	24
3.4.6 (RCS loops - Mode 4)	A. One RCS loop inoperable	3.3E-3	3.2E-3	3

- The accident sequences that dominate the risk in Mode 5 and 4 (on SDC cooling) are similar to the Davis-Besse case. The major contributing accident sequences to the risk are initiated by loss of the operating DHR train, which includes LOOP and loss of CCW initiating accident sequences, that affect the function of the DHR among other systems, and by accident sequences initiated by a loss of RCS inventory outside of the RB. The latter category of accident sequences are larger contributors to risk than they are at

Davis-Besse due to the higher initiating event frequency of the loss of RCS inventory events in the “non-Davis-Besse” model associated with the alignment of SDC.

- As in the case for Davis-Besse, the dominant contributors to risk when the plant is in the proposed end-state (Mode 4 on SG cooling) are associated with failures of redundant or diverse means of performing a safety function, such as failures that affect normal cooling (feedwater) and also backup and emergency cooling methods. Therefore, the insight that failures that dominate the risk are associated with equipment which have already been recognized as important contributors by the TS and for which operational requirements and guidance (e.g., compensatory measures) are in place, is valid for all B&WOG plants.

These insights indicate that the results of the quantitative risk assessment are robust, and that the conclusions of both the qualitative and quantitative risk assessments do not change when the impact of design and operational differences among B&WOG plants is considered.

Important insights from the investigation of the robustness of the results to uncertainties in data and modeling assumptions, through sensitivity studies, are listed below.

- Accident initiating event (IE) frequencies were calculated based on a combination of operating experience and data from previous PRA studies. Because some of these frequencies are important contributors to risk, the sensitivity of the risk assessment results to values assumed for these frequencies was investigated. The following sensitivity studies, associated with IE frequencies, were performed:

  - The IE frequency for the loss of inventory event (both inside and outside the RB at Davis-Besse while the plant is operating in Mode 4 (on SG cooling) was increased by a factor of 5. Since Davis-Besse aligns the RCS to SDC in Mode 4, the Mode 4 pressure is not very different than the Mode 5 pressure. For this reason, the same IE frequencies for the loss of inventory event are used for both Modes 4 and 5 in the baseline risk assessment (8.4E-3/year for loss of inventory inside the RB and about 1.3E-2/year for loss of inventory outside the RB. This sensitivity study was performed to investigate the robustness of the risk assessment results and conclusions to the slightly higher RCS pressure associated with the Mode 4 end-state. It was shown that the results and conclusions are valid, even if the frequencies for the loss of inventory event (both inside and outside the RB are conservatively assumed to be significantly worse in Mode 4 than in Mode 5.

  - The Mode 4 (on SG cooling) IE frequency for loss of inventory inside the RB at B&WOG plants with lowered-loop SG design (as compared to Davis-Besse) was increased by a factor of 5. This sensitivity study was performed to investigate the robustness of the risk assessment results and conclusions to the assumed loss of inventory IE frequency in conjunction with the lowered-loop SG design. In the lowered-loop plants, less SG inventory is available to drain back into the RCS. This results in shorter boil-off times to core uncovery and affects recovery probabilities. Since these plants do not pre-align the RCS to the SDC system in Mode 4 (on SG cooling) and use the PORVs for LTOP, the factor of 5 was applied only to the loss inventory frequency inside the RB. It was shown that the results and conclusions are valid, even if the frequency of the loss of inventory event inside the RB is conservatively assumed to be significantly worse in Mode 4 than it is in Mode 5.

- The IE frequency for transients occurring in Mode 4 and Mode 5 were increased by a factor of 5. This sensitivity study was performed to investigate the robustness of the risk assessment results and conclusions to transient initiating events, which are major contributors to risk since they involve loss of the operating decay heat removal method, such as loss of feedwater, LOOP, loss of CCW, and loss of SWS events. The investigation focused on initiating events that have the potential to affect Mode 4 (on SG cooling) risk more than Mode 5 risk. It was shown that the results and conclusions of the quantitative risk assessment regarding the comparison between Mode 4 and Mode 5 end-state risks are valid, even when uncertainties in the transient IE frequencies are considered.

- All failure to recover probabilities (in both the Davis-Besse and the “non-Davis-Besse” PRA models), were increased by a factor of 10 (were changed to 1.0 if greater than 0.1). The results of this sensitivity have shown that the Mode 4 versus Mode 5 end-state comparisons are robust, even when considering an order of magnitude increase of the recovery probability values to account for uncertainties.

- The failure to trip the RCPs following a transient IE, to prevent RCP seal LOCA, was changed to 1.0 (screening probability). The results of this sensitivity study have shown no sensitivity to this human error probability.

- All CCF probabilities, in both the Davis-Besse and the “non-Davis-Besse” PRA models, and both the Mode 4 and Mode 5 end-states, were increased by a factor of 3. The results of this sensitivity study have shown that the Mode 4 versus Mode 5 end-state comparisons are robust and demonstrate that uncertainty in CCF data does not affect the conclusion of the quantitative risk assessment.

These insights indicate that the results of the quantitative risk assessment are robust and that the conclusions of both the qualitative and quantitative risk assessments do not change when uncertainties in data and modeling assumptions are considered.

The NRC staff believes that the above listed insights substantiate the generic conclusion that plant operation in Mode 4 (hot shutdown) offers at least the same robustness to plant upsets as operation in Mode 5 (cold shutdown).

4.6 Conclusions

The NRC staff’s review finds that the BAW-2441 risk assessment approach is comprehensive and follows 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 (documented in RG 1.177) 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 significant temporary risk increases, as defined by RG 1.177 criteria, associated with the implementation of the proposed TS end-state changes.

- Avoidance of Risk-Significant Configurations (Tier 2). The performed risk analyses, which are based on single LCOs, have shown that there are no high risk configurations associated with the proposed TS end-state changes. The reliability of redundant trains is normally covered by a single LCO. When multiple LCOs occur, which affect trains in several systems, the plant's risk-informed CRMP, implemented in response to the Maintenance Rule 10 CFR 50.65(a)(4), will ensure that high risk configurations are avoided. As part of the implementation of the 10 CFR 50.65(a)(4) program, licensees are expected to include guidance in appropriate plant procedures and/or administrative controls to preclude high risk plant configurations when the plant is at the proposed end-state. The NRC staff finds that such guidance is adequate 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 whenever a risk-informed TS is entered.

The generic risk impact of the proposed end-state mode change was evaluated subject to the following assumptions:

- The entry into the proposed end-state is initiated by the inoperability of a single train of equipment, or a restriction on a plant operational parameter, unless otherwise stated in the applicable technical specification;

- The primary purpose of entering the end-state is to correct the initiating condition and return to power as soon as is practical.

- Implementation guidance for the proposed end-state changes should be developed to ensure that insights and assumptions made in the risk assessment are properly reflected in the plant-specific CRMP at the participating B&WOG plants.

These assumptions are consistent with typical entries into Mode 4 for short duration repairs, which is the intended use of the TS end-state changes.

The NRC staff concludes that, in general, going to Mode 4 (hot shutdown) instead of going to Mode 5 (cold shutdown) to carry out equipment repairs does not have any adverse effect on plant risk and may actually reduce risk. Therefore, the NRC staff finds that the risk information provided by BAW-2441 supports the requested change.

5.0 CONCLUSION

BAW-2441, Revision 2, proposed to change the end state of the selected LCOs from Mode 5 to Mode 4. The NRC staff has reviewed the technical and risk assessments that provide justification to the proposed changes to the end-states for selected LCOs. Based on the above evaluations, the NRC staff concludes the proposed changes to these LCOs are acceptable.

6.0 REFERENCES:

1. Letter from James F. Malley, B&W Owners Group, to US Nuclear Regulatory Commission, “Request for Review of BAW-2441, Revision 2, ‘Risk Informed Justification for LCO End-State Changes,’” January 19, 2004, NRC:04:001. ADAMS Accession No. ML040260016

2. Letter from Howard Crawford, B&W Owners Group, to US Nuclear Regulatory Commission, “Response to Request for Additional Information for BAW-2441, Revision 2, ‘Risk Informed Justification for LCO End State Changes,’” February 17, 2005, NRC:05:010. ADAMS Accession No. ML050560034

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