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Sent: Thursday, June 16, 2011 12:44 PM
To: Lingam, Siva
Subject: CR-3 EPU LAR Submittal (Part 3 of 5)
Attachments: Attachment 6_ CR-3 EPU LAR.pdf; Attachment 7 _CR-3 EPU LAR_Non-Proprietary.pdf

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PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #309, REVISION 0

ATTACHMENT 6

AFFIDAVIT FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. (AREVA NP) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in Engineering Information Records 51-9076487-000 entitled "Crystal River Unit 3 Extended Power Uprate Technical Report," dated March 2010, 51-9150228-000 entitled "Additional Input to CR-3 EPU LR Re: TACO3/GDTACO/COPERNIC (CO-45)," dated December 2010, and 51-9163143-000 entitled, "CR-3 EPU Technical Report Revision Sections," dated June 2011 and referred to herein as "Documents." Information contained in these Documents has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. These Documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these Documents as proprietary and confidential.

5. These Documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these Documents

be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in these Documents is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in these Documents have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 15th

day of June 2011.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2014

1.0 Introduction to the Crystal River Unit 3 Extended Power Uprate Technical Report

General Overview of the CR-3 Extended Power Uprate (EPU) Technical Report

Crystal River Unit 3 (CR-3) is a Lowered Loop B&W 177 Fuel Assembly (FA) Pressurized Water Reactor. The 177 FA fleet includes Arkansas Unit 1, Oconee Units 1, 2, and 3, Davis Besse, and Three Mile Island. CR-3 is owned and operated by Progress Energy making it part of a five unit fleet that includes Robinson, Harris, and Brunswick Units 1 and 2. CR-3 is a single-unit nuclear plant that shares a coastal site in Western Florida with four coal units which, when taken together, form one of the largest generating facilities in the nation. CR-1, -2, and -3 utilize once-through cooling relying on common intake and discharge canals connected to the Gulf of Mexico.

The CR-3 EPU Technical Report (TR) is a technical summary of the results of the analyses and evaluations performed to demonstrate that the proposed increase in plant power can be safely achieved. This Technical Report is one of the principle attachments to the CR-3 License Amendment Request (LAR) 309 which requests approval, by the Nuclear Regulatory Commission (NRC), to increase the Rated Thermal Power from 2609 MWt to 3014 MWt. CR-3 has been uprated incrementally since initially being licensed at 2452 MWt. This EPU will take CR-3 slightly above the highest rated unit in the B&W fleet (currently Davis Besse at 2817 MWt). This Technical Report provides the technical basis for the associated CR-3 Facility Operating License and Improved Technical Specifications (ITS) changes. The EPU Technical Report works in concert with the other attachments to the LAR in order to provide the NRC reviewers with a comprehensive evaluation of the effects of the proposed EPU.

Progress Energy Florida, Inc. (PEF) contracted with AREVA NP, and through them, with AREVA FA and WorleyParsons (formerly Gilbert Associates, the CR-3 Architect/Engineer (A/E)) and others to perform a number of evaluations as shown below. The status of these evaluations was discussed with the NRC during various pre-application meetings from 2007 through 2011.

- A conceptual level evaluation (i.e., scoping study) of the CR-3 EPU was conducted by AREVA NP and was reviewed and accepted by PEF after resolving Technical Expert Panel (Participants included AREVA, Progress Energy (PGN), Worley Parsons and MPR) and station management comments resulting from reviews that were conducted in 2006 and 2007, respectively. As the design details progressed, station and project management personnel met regularly to finalize key decisions.

- AREVA FA is and has been the fuel provider for CR-3 since commercial operation. They evaluated the impact on fuels from both Fuel Management and interaction with Safety and Transient Analysis perspectives.

- AREVA NP is the Nuclear Steam Supply System (NSSS) vendor for CR-3 and, as such, maintains the Analyses of Record for both Safety and Transient Analysis from Reactivity, and Thermal-Hydraulic and Mass-and-Energy-release perspectives, which have been reanalyzed to reflect EPU conditions.

- WorleyParsons is working with AREVA NP to support some of the analytical work and to develop the necessary engineering change packages.

- AREVA NP produced revised source terms at EPU conditions for both onsite and offsite dose calculations which supported revisions to the CR-3 Alternate Source Term (AST) based dose calculations.
- Offsite dose calculations were also based on updated metrological data and associated Atmospheric Dispersion Factors (X/Q) provided by Murray and Trettel, Inc.
- The onsite source terms were integrated into the three dimensional site model by Sargent and Lundy and used for Vital Area and Environmental Qualification (EQ) zone and point specific dose calculations.
- AREVA NP worked with PEF and others (B&W Canada) to further evaluate structure, system and component impacts beyond the preliminary reviews conducted as part of the scoping/evaluation phases. In particular, the reactor coolant system (RCS) loop, attached piping and related structural evaluations, including confirmation of the continued applicability of Leak-Before-Break approvals, were updated.
- AREVA NP provided revised system and component evaluations including but not limited to an update to the Reactor Coolant System Functional Specifications, Reactor Vessel and Internals Materials, updated PEPSE (thermal model), updated CHECKWORKS (flow-accelerated corrosion models, etc.
- B&W Canada evaluated the impact of the EPU on the Replacement Once-Through Steam Generators and supports, and provided inputs to the appropriate accident and transient analyses performed by AREVA NP.
- Siemens, the supplier of the new CR-3 turbine-generator package, provided appropriate information to evaluate the secondary plant side from thermal-hydraulic and electrical perspectives.
- Holtec International evaluated the impact of the EPU burn-up profiles on the current spent fuel storage criticality analyses. As discussed with the NRC during September 23, 2009 and March 24, 2011 teleconferences and again at the final pre-application meeting with the NRC on April 21, 2011, this will rely on conservative ITS changes crediting sufficient boron concentrations and resulting criticality margin.
- PEF contracted with Tetra Tech, Inc. to provide the Supplemental Environmental Report (See Attachment 9). This company was chosen, in part, because they supported a similar effort associated with the CR-3 License Renewal Application which was submitted to the NRC in December 2008. Thus, the NRC Staff is familiar with much of the information contained in the Supplemental Environmental Report. Further, TetraTech, Inc. relied heavily on information provided to the State of Florida as part of their environmental review process.

The results of these evaluations have been compiled in accordance with RS-001,"Review Standard for Extended Power Uprates." The CR-3 EPU Technical Report sections are generally formatted as follows:

- Section names are not all consistent with RS-001. Instead the corresponding CR-3 specific system names are used where this brings necessary clarity.

- The Regulatory Evaluation section replicates the wording from RS-001 and applies it as appropriate in the CR-3 Current Licensing Basis (CLB) section.
- The Technical Evaluation section focuses on changes as a result of EPU. Sufficient information is supplied for context and completeness; however, efforts were made to limit repeating detailed information that is readily available (e.g., CR-3 Final Safety Analysis Report).
- The Conclusion section is presented consistent with RS-001 and NRC Safety Evaluations on previous EPU dockets, reflecting the CR-3 specific evaluation.

The Technical Report supports a comprehensive understanding of the effects of EPU on CR-3. Operating Experience was relied upon extensively; most notably the Safety Evaluations generated by the NRC Staff in support of recent PWR EPUs. Relevant requests for additional information (RAIs) on other dockets, acceptance review feedback and other related NRC Staff actions were reviewed and incorporated into this Technical Report, as appropriate.

The role of the Technical Report is to document the technical basis for the evaluation of the effects of the proposed changes necessary to implement the EPU with a sufficient level of detail to permit the NRC Staff to reach an informed determination regarding the consistency, quality, and completeness of the evaluation with respect to the areas that are within the NRC scope of review. It is recognized that the NRC will review the CR-3 EPU Technical Report in its entirety and will perform independent evaluations, calculations, and audits as deemed necessary and appropriate to reach its own conclusion concerning the effects of the proposed EPU and the continued safe operation of CR-3. The technical evaluations presented in the Technical Report include, when appropriate, discussion of the effects of the EPU on plant operating limits, functional performance requirements and design margins as well as describing the methods CR-3 used in reaching the conclusions documented in the report consistent with the guidance of RS-001.

Summary of EPU Impacts

The descriptions below follow the energy flow from its source (the fuel assembly) to the eventual delivery of electrical energy to the grid. Significant changes in plant parameters are discussed, as well as a brief description of the basis for the plant modifications that are required. Conclusions of evaluations summarized in greater detail elsewhere in the Technical Report are not justified in this section.

Fuel Assembly

The fuel assembly used for the uprate is unchanged from that currently in-service at CR-3. The Mark-B-HTP fuel design was first introduced at CR-3 in Cycle 14. The Mark-B-HTP fuel design is a 15x15 fuel assembly design for operation in a B&W 177 fuel assembly PWR reactor core. The Mark-B-HTP utilizes M5™ alloy fuel rod cladding rod end caps, guide tubes and instrument tubes. The advanced alloy M5™ provides superior corrosion resistance compared to other Zirconium alloys. The resulting alloy microstructure is highly stable under irradiation and provides superior in-reactor performance. These improvements permit higher burnup and extended power uprate of the fuel in conjunction with improved thermal and mechanical performance. In order to support the increased energy requirements at the EPU power level, CR-3 will nominally install up to 88 new fuel assemblies each reload cycle as opposed to a current nominal batch feed of 76 new fuel assemblies.

Reactor Core

The uprated core will operate at a core thermal power of 3014 MWt as compared to the current core thermal power of 2609 MWt. This represents an increase of 15.5% in core thermal power. The uprate core design will be very similar to the core design now in use at CR-3. Fuel enrichment will remain within the current limit of 5.0 wt%. Nominal refueling batch sizes (new fuel) will increase to 88 assemblies. The core cycle length will continue to be two years. The core power density will increase to support the uprate power increase resulting in a proportionally higher coolant temperature increase across the core.

Core Decay Heat

Core decay heat increases proportional to the uprate core power increase. The higher decay heat results in the need for modifications to systems that are used to remove decay heat. The following are examples of such modifications (see Appendix E for details):

- The required flow from the Emergency Feedwater (EFW) pumps increases from 550 gpm to 660 gpm for the Loss of Feedwater (LOFW) event. In order to assure sufficient flow, the EFW recirculation flow paths are isolated when EFW flow is adequate to assure reliable pump operation. The recirculation isolation eliminates its flow diversion and assures sufficient EFW flow to the SGs without otherwise modifying the system.

- To meet Appendix R requirements (to achieve safe shutdown in 72 hours with no residual heat removal available), CR-3 Abnormal Procedures require steaming through the Atmospheric Dump Valves (ADV) on both steam generators. Due to the increased decay heat, the size of the ADVs is increased.

- Both the Low Pressure Injection (LPI) Cross-Tie and ADV/Fast Cooldown System modifications (see Appendix E, Major Plant Modifications, for details) are required as a direct result of the increased decay heat load impact on design basis accident results.

Core Accident Source Term

The core accident source term was conservatively calculated based on key parameters (i.e., rated thermal power, enrichment, and burn-up) which affected the core inventory. The maximum activity for each radionuclide was selected to provide a maximum core average inventory. This is discussed further in the Technical Report; specifically Section 2.9.2, “Radiological Consequences Analyses”.

Reactor Coolant System

The RCS operation changes very little as a result of the uprate. The system operating pressure does not change and there are no planned physical modifications to the RCS or reactor vessel internals. In order to provide the necessary steam pressure at the uprate power level, the full power average coolant temperature increases 3°F to approximately 582°F. CR-3 replaced the reactor vessel head in 2003 and modified the remaining hot leg pressure boundary to eliminate or otherwise mitigate components with Alloy 600 material. The reactor coolant temperature increase across the core is proportional to the increase in uprate power. With the higher T_{AVG} and higher core temperature difference, the reactor vessel hot leg temperature (T_{HOT}) increases. The reactor vessel inlet temperature (T_{COLD}) decreases. The RCS zero-power T_{AVG} does not change from the current value. The larger change in T_{AVG} from full power to zero power results in a greater shrink in pressurizer level following a reactor trip. Nevertheless, this does not require a change in the full power pressurizer level control program which is nominally set at 220 inches.

Emergency Core Cooling System Related Modifications

The LPI system is being modified to improve performance with regard to one limiting postulated break and to enhance post-accident boron precipitation mitigation. The two trains are being cross-tied and a line from that cross-tie is being directed to the hot leg to replace the currently licensed active boron precipitation mitigation features.

Two additional modifications improve ECCS performance during postulated Small Break Loss-of-Coolant Accident (SBLOCA) scenarios. The first reduces system flow resistance by opening fixed-position throttle valves thereby increasing HPI flow. The second referred to as a Fast Cooldown System (FCS) adds an alternate steam pressure controller for each of the ADVs. The FCS is initiated by the Inadequate Core Cooling Monitoring System (ICCMS) which is an analog monitoring and actuation system being addressed in Section 2.4.2.3. This will respond to a Loss of Subcooling Margin with inadequate HPI flow (most likely associated with single failures of a pump or other elements of the system). The net effect is to more rapidly depressurize the secondary plant thereby enhancing primary to secondary heat transfer and HPI performance leading to a more rapid core flood tank actuation in certain SBLOCA transients.

These modifications are described in further detail in Appendix E and its Enclosures.

Steam Generator, Main Turbine and Main Steam

The CR-3 steam generators were replaced during the Fall 2009 refueling outage (R16). As a part of the R17 modifications, the high pressure turbine and low pressures turbines will be replaced in order to pass the additional volumetric steam. The Secondary Cooling System was significantly upgraded to support the associated increased heat loads.

New Moisture Separator Reheaters (MSRs) were installed to support EPU conditions. In addition, and in order to improve plant thermal efficiency, Moisture Separator Belly Drain Heat Exchangers were added to the plant to recapture energy from the MSR drains before being dumped to the condenser.

Turbine Bypass Valves have been replaced to increase the capacity of the steam dump system relative to full power steam flow after uprate to maintain the plant load rejection capability. The ADVs will be replaced in R17 with larger safety-related valves. See Appendix E for details.

For the uprate, static steam pressure exiting the new steam generators nozzles increases to 958 psia (from 924.4 psia) as a result of increased turbine throttle control pressure and higher feedwater and steam flow. The mass flow rate of steam increases proportionately to the power level increase to deliver the energy to the main turbine. Steam velocity in the main steam piping increases due to the steam mass flow rate increase. The higher velocity has the potential to increase flow induced vibration. Regulatory Guide 1.20, R3 suggests that appropriate components be evaluated for the potential of flow induced vibration. This was completed and documented in the applicable Sections: 2.1.9, Steam Generator Tube Inservice Inspection, 2.2.2, Pressure-Retaining Components and Component Supports and 2.2.3, Rector Pressure Vessel Internals and Core Supports. The absence of steam dryer or similar internal structures in either the Reactor Vessel or Steam Generator reduces the likelihood of flow induced vibration when compared to other NSSS designs. Nevertheless, as discussed in Section 2.12.1, “Approach to EPU Power Level and Test Plan,” a robust vibration monitoring program will be implemented to validate that experienced vibration levels will not exceed projected levels.

Main Condenser and Circulating Water

The new low-pressure turbines are designed to provide a main condenser back pressure of 2.7 inches Hg at a circulating water temperature of 75°F. The main condensers were modeled and evaluated for the new steam flows, including flow induced vibration, and found to be acceptable. The circulating water flow rate from the condenser will not increase but will reject more heat to the discharge canal shared by CR Units 1, 2, and 3. The increased heat load will be compensated by either power reductions or operation of a forced draft cooling tower installed downstream of the unit discharge to remove this increased heat load (See Supplemental Environmental Report for details).

Condensate and Feedwater

The condensate and feedwater flow rates increase proportional to the uprate power increase. Larger main feedwater booster pumps and motors will be installed and the main feedwater pumps will be replaced in R17 to deliver the required flow and pressure to the steam generators at EPU conditions. The existing feedwater startup valves are acceptable for EPU conditions. The feedwater temperature to the steam generator increases to 460°F at EPU conditions from the current conditions of 458.4°F.

The condensate pumps and motors will be modified to accommodate the increase in condensate flow in R17. Condensate flow control will be changed from the current operation, which uses variable speed magnetic drive pump couplings, to constant speed condensate pumps directly coupled to the motors. Additionally, condensate flow control will be via newly installed control valves in the condensate system. The modified condensate system increases operational margin from current conditions. See Appendix E for details.

Extraction Steam and Heater Drains

The slight increase in the temperatures, pressures, and flows in the extraction steam piping and in the various heater drains and deaerator required various modifications to assure that each drain flow path is capable of passing the additional flow. Monitoring of the various drain and level control systems will be an important part of the power escalation procedure. During this time, these systems are tuned to assure stable control. The flow accelerated corrosion model has been updated for the new conditions in these systems at uprated conditions. Certain lines will require increased monitoring, but all lines will be within the capability of the program to be monitored safely. See Appendix E for details.

Main Generator

The main generator electrical output at EPU conditions increases by approximately 180 MWe from current conditions. The generator was completely rebuilt on-site. See Appendix E for details.

Iso-Phase Bus Ducts

The Generator Step-Up (GSU) transformers were previously replaced with a capacity to meet uprate power requirements. The iso-phase bus and bus duct cooling system has also been modified to handle the additional amperage and associated heat loads.

Station Switchyards

CR-3 interface with the transmission grid is provided from the 230 kV switchyard, which is supported by three neighboring coal units (Crystal River 1, 2, and 4) and multiple offsite transmission lines. The CR-3 generator output is routed via the 500 kV switchyard and, along with the output of Crystal River Unit 5, is directed to the grid through two corridors. No changes to either switchyard are necessary to support EPU. Additionally, a grid stability study was performed to verify that while operating at the uprated power, CR-3 will not impact the reliability of the grid. A detailed report on that evaluation is provided as Appendix F, Grid Stability.

Current Licensing Basis (CLB)

The CR-3 CLB is presented primarily in the FSAR, Improved Technical Specifications, Pressure/Temperature Limits Report, and the Core Operating Limits Report. The CLB includes the application of various FSAR Criteria originally developed by the industry in the early 1960s to provide some consistent guidance to evaluate the design and performance of commercial nuclear power plants.

In 1971, the Atomic Energy Commission issued 10 CFR50, Appendix A, "General Design Criteria." The GDCs were similar to the former design criteria upon which the FSAR Criteria were based, but they were expanded to include additional design considerations. Details concerning the CR-3 design criteria are found in FSAR Section 1.4.

Each section of the EPU Technical Report contains a brief outline of the CR-3 CLB with respect to the RS-001 cited Regulatory scope.

In support of the LAR, NRC approval has been requested for changes to the CR-3 CLB. Refer to Attachment 1 for summary of these changes and Appendix E for additional design information related to plant modifications required for EPU.

Sections within the Technical Report in addition to those specified in RS-001

In order to provide a complete description of the analysis performed, the CR-3 EPU Technical Report takes advantage of the provision in RS-001 to add additional sections (additional review areas).

The following sections are in addition to the standard set

1.0 Introduction to the Crystal River Unit 3 Extended Power Uprate Technical Report

1.1 Nuclear Steam Supply System Parameters

2.2.6 Incore Instrumentation Guide Tubes

2.2.2.8 NSSS Design Transients

2.7.7	Reactor Building Ventilation Systems
2.8.5.0	Non-LOCA Analyses Introduction
2.8.7.1	Loss of Decay Heat Removal at Mid-loop
2.12.2	Transient Performance
2.14	The Effects of EPU on the Renewed Licensing and License Renewal Programs
Appendix A	Safety Evaluation Report Compliance
Appendix B	Additional Codes and Methods
Appendix C	Associated Technical Review Guidance
Appendix D	Core Boric Acid Dilution Control for CR-3 at EPU Conditions
Appendix E	Major Plant Modifications
Appendix F	Grid Stability
Appendix G	Acronyms in Addition to those in RS-001

Use of Industry Operating Experience

Both the regulators and the nuclear industry peer groups strongly advocate incorporating operating experience (OE) and lessons learned as basic inputs in design, maintenance, operating and licensing activities. The analysis and evaluations performed for the CR-3 EPU took full advantage of past EPU experiences by:

- Review of previous power uprate applications and NRC RAIs. PWR RAIs were reviewed and, where appropriate, the plant analysis or evaluations relating to the subject were reviewed against the expressed concern to provide reviewer confidence that the issue was appropriately examined. BWR RAIs were also reviewed when the RAI was related to issues other than those unique to BWRs.
- Members of the EPU staff attended a number of other utility EPU meetings, participated in regular teleconference, etc. to maintain a first-hand awareness of current issues.
- During the development of the EPU (including design changes, analyses and evaluations leading to this LAR) several public pre-application meetings were held with the NRC Staff. At these meetings, the NRC Staff ensured that CR-3 was aware of the growing body of lessons learned. These lessons learned were factored into the timing or content of various activities.
- Review of the Institute of Nuclear Power Operations communications relating to power uprates.
- Members of the EPU staff represented Progress Energy on the Nuclear Energy Institute Power Uprate Task Force and are continuing to monitor industry efforts through its successor managed by the Electric Power Research Institute. Further, key AREVA NP management were also members of

these task forces.

- During the analysis and evaluation activities, careful attention was paid to ensure that system and component operating history was considered. System and Design engineers were interviewed to ensure all pertinent information was considered in the EPU evaluations.
- Progress Energy chose to establish an Expert Panel comprised of EPU-experienced individuals within the Progress Energy fleet and industry to perform a review of the Technical Report. The comments that were generated by the Expert Panel reviews were combined with those from CR-3 engineering and other internal Progress Energy organizations to assure a high quality submittal.
- A high level Executive Oversight Committee was formed to oversee EPU project plans and progress. The committee was comprised of CR-3 site personnel, Progress Energy corporate staff and vendor senior management to assure appropriate resources were brought to bear where necessary.

Treatment of Proprietary Information referenced within the Technical Report

To enhance stakeholder participation, every effort was made to reduce the level of proprietary information necessary to support the CR-3 EPU. However, some of the methods and detailed results are proprietary to AREVA NP. These are redacted from the non-proprietary portions and are accompanied by an appropriate affidavit.

1.1 Nuclear Steam Supply System Parameters

The Nuclear Steam Supply System (NSSS) Design Parameters are the fundamental parameters used as input in all of the NSSS analyses. The NSSS Design Parameters provide the primary and secondary side system conditions (thermal power, temperature, pressure, and flow rates) that serve as the basis for all of the NSSS analyses and evaluations.

1.1.1 Technical Evaluation

Introduction

The current CR-3 NSSS design and operating parameters are summarized in FSAR Chapter 3 and are updated, as applicable, in the cycle specific Core Operating Limits Report (COLR). As a result of the EPU, the CR-3 NSSS design operating parameters have been revised as shown in Table 1.1-1 below. Table 1.1-1 provides information for the current design basis conditions as well as for various cases representing operation following the EPU. All of the information in Table 1.1-1 reflects operation with the replacement once-through steam generators (OTSGs). These parameters have been utilized, as appropriate, in the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the EPU.

Description of Analyses and Evaluations

The NSSS Design Parameters provide the Reactor Coolant System (RCS) and secondary system conditions (thermal power, temperature, pressure, and flow rates) that are used as the basis for the design transients, systems, structures, components, accidents, and fuel analyses and evaluations.

Table 1.1-1 provides the NSSS design operating parameter cases that were evaluated and serve as the basis for the EPU. Cases were evaluated at the current design basis conditions, as well as at the EPU conditions with the expected operating T_{AVG} of 582°F. Tube plugging levels ranging from 0% to 5% plugged tubes were evaluated. Measurement uncertainties and/or biases on operating parameters such as thermal power, temperature, pressure, and flow rates are not incorporated in the development of the NSSS design parameters. Rather, uncertainties are applied where appropriate in the safety analyses discussed in Section 2.8.5, Accident and Transient Analyses. The cases are summarized below.

- Case 1 gives the NSSS parameters at the current design basis operating conditions.
- Cases 2 and 3 give the NSSS parameters at the proposed EPU operating conditions (T_{AVG} = 582°F) with 0% and 5% tube plugging levels respectively.

Primary side parameters are determined using a CR-3 specific RCS hydraulics model for the FSPLIT5A computer code. There is no explicit NRC approval for the code since it is used to facilitate calculations that could be performed by hand. The code and method used to calculate these values have been successfully used to license all previous similar programs for Babcock & Wilcox (B&W) plants. The code and method use basic thermal-hydraulic calculations, along with the first principles of engineering, to generate the primary side temperatures, pressures, and flow rates shown in Table 1.1-1.

Secondary side parameters are determined using a CR-3 specific SG model for the VAGEN code. VAGEN is a one-dimensional, homogeneous equilibrium OTSG performance code that predicts mean steam temperatures. VAGEN has been used successfully for determining performance of once-through steam generators for B&W plants. Secondary side design operating parameters were calculated for Cases 1 through 3 in Table 1.1-1. These cases represent the current operating conditions as well as the proposed EPU operating conditions including various tube plugging levels.

The major input parameters and assumptions used in the calculation of the three cases established for EPU conditions are summarized by the following:

- The parameters are applicable to the current operating conditions of the OTSGs (replaced Fall 2009).
- An uprated NSSS power level of 3030 MWt was assumed for the NSSS analyses, based on a nominal, rated core thermal power of 3014 MWt + 16 MWt RCP net heat input. A power measurement uncertainty of 0.4% using the Leading Edge Flow Meter (LEFM) is applied to the nominal, rated core thermal power, as appropriate, in the safety analyses.
- A feedwater temperature (T_{FW}) of 460°F was selected for the analysis of the EPU cases.
- A steam pressure at the tube bundle exit (P_{STM}) of 968.6 psia was input to all the EPU cases (Cases 2, 3). A steam pressure of 933.3 psia at the tube bundle exit was input to Case 1, representing the current design basis operating conditions.
- RCS loop flow of greater than 88,000 gpm/loop was maintained for the analyses.
- A full-power normal operating T_{AVG} of 582°F was assumed for the analysis of the EPU cases. A T_{AVG} reduction of 7°F was also evaluated for end of cycle maneuvering.
- Steam Generator Tube Plugging levels of 0% and 5% were evaluated.
- The current assumed steam generator fouling factor of 1.0E-6 hr-ft^2-°F/BTU was maintained.

The acceptance criteria for determining the NSSS Design Parameters were that the results of the EPU analyses and evaluations continue to comply with all industry and regulatory requirements applicable to CR-3, and that they provide adequate flexibility and margin during plant operation.

Results

The CR-3 NSSS design operating parameters at current operating conditions as well as at the EPU operating conditions, have been calculated and are shown in Table 1.1-1. A simplified primary side heat balance diagram at the proposed EPU operating conditions is provided in Figure 1.1-1. This heat balance diagram illustrates the design parameters from Case 2 in Table 1.1-1.

1.1.2 Conclusion

The resulting EPU operating conditions shown in Table 1.1-1 were used as the basis for all the analytical efforts. The analyses and evaluations performed were based on the parameter sets that were most limiting, so that the analyses would support operation over the entire range of conditions specified.

1.1.3 References

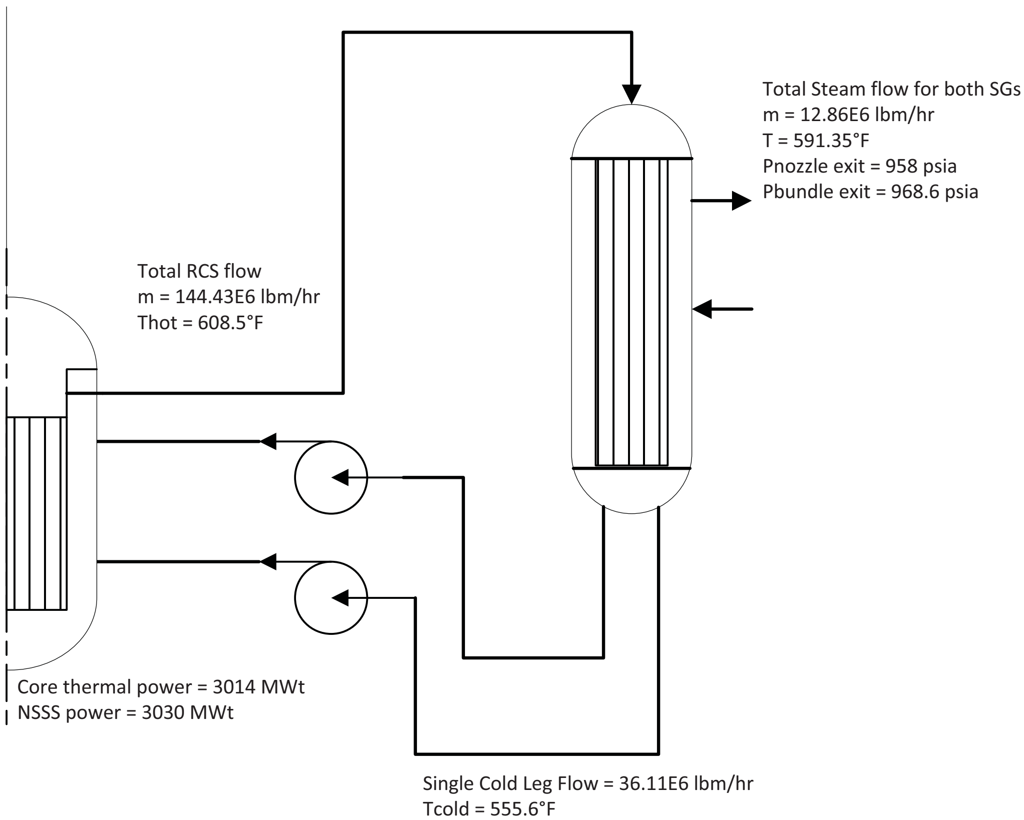
None

Table 1.1-1
NSSS Design Operating Parameters for Crystal River Unit 3

Thermal Design Parameters	Case 1	Case 2	Case 3
Total NSSS Power			
MWt	2625	3030	3030
10^6 Btu/hr	8,959	10,341	10,341
Core Thermal Power, MWt	2609	3014	3014
10^6 Btu/hr	8,904	10,287	10,287
Net RCP Heat Input, MWt	16	16	16
10^6 Btu/hr	55	55	55
RCS Vol. Flow, gpm	388,371	387,560	384,688
RCS Mass Flow, 10^6 lbm/hr	144.67	144.43	143.4
Tube Flow Area Basis (1)	Nominal	Nominal	Nominal
Reactor Coolant System			
T_{HOT}, °F	602.1	608.5	608.7
T_{COLD}, °F	555.9	555.6	555.4
T_{AVG}, °F	579.0	582.0	582.0
Pressure, psia	2170	2170	2170
Steam Generator			
Steam Temperature, °F	595.25	591.35	585.65
Steam Pressure at Nozzle Exit, psia	924.4	958	958
Steam Pressure at Bundle Exit, psia	933.3	968.6	968.6
Total Steam Flow Rate, 10^6 lbm/hr	11.02	12.86	12.95
Feedwater Temperature, °F	458.4	460.0	460.0
Steam Superheat, °F	58.98	50.63	44.93
Tube Plugging Level per SG, %	0%	0%	5%
Terminal Temperature Difference, °F (T_{HOT} – T_{STEAM})	6.9	17.2	23.1

(1) Tube flow area basis refers to the calculation of SG primary side flow area. All calculations are based on a nominal tube diameter of 0.625”. Cases 1 – 3 use the nominal tube wall thickness.

Figure 1.1-1
Simplified Primary Heat Balance Diagram at Nominal EPU Conditions



2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

2.1.1.1 Regulatory Evaluation

The Reactor Vessel Material Surveillance Program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The review in Section 2.1.1, Reactor Vessel Material Surveillance Program, primarily focuses on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule.

The NRC's acceptance criteria for the Reactor Vessel Material Surveillance Program are based on:

- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of gross rupture or significant leakage throughout its design lifetime;
- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized;
- 10 CFR 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region; and,
- 10 CFR 50.60, which requires compliance with the requirements of 10 CFR 50, Appendix H.

<u>CR-3 Current Licensing Basis</u>

As noted in the FSAR, Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have exceedingly low probability of gross rupture or significant leakage throughout its design lifetime [GDC-14]; and
- FSAR Section 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention, and FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized [GDC-31].

As described in FSAR Section 4.4.5, "Material Irradiation Surveillance," a master integrated reactor vessel surveillance program (MIRVSP) (Reference 1) has been developed to monitor the irradiation-induced material changes of the steel and weldments routinely used in the reactor vessels of B&W 177 FA plants. CR-3 participates in the MIRVSP which is described in detail in BAW-1543A, and includes provisions for the plant specific surveillance programs of each participant and for a research capsule program. The MIRVSP complies with ASTM E185-82 and also addresses the additional requirements of 10 CFR 50, Appendices G and H.

2.1.1.2 Technical Evaluation

Introduction

Reactor vessel integrity is impacted by any change in plant parameters that affect neutron fluence levels or temperature/pressure transients. The changes in neutron fluence resulting from the EPU have been evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the evaluation of the EPU impact on the CR-3 surveillance program compliance with ASTM E185-82.

Description of Analyses and Evaluations

EPU Fluence Projections

Detailed fluence calculations were performed for CR-3 at the EPU conditions. The purpose of these calculations is to show that CR-3 continues to meet the reactor vessel surveillance requirements defined in 10 CFR 50, Appendix H and the guidance provided in Regulatory Guide 1.190 relating to reactor pressure vessel surveillance programs. The fluence methodology is described in AREVA's NRC approved fluence topical report (Reference 2). It has been demonstrated in an independent benchmark experiment (listed in Reference 2) that the results of a fluence analysis that employs the methodology in Reference 2 were unbiased and had a precision well within the NRC suggested standard deviation (σ) limit of 20% from Regulatory Guide 1.190.

The fluence projections are presented in Table 2.1.1-1. Surveillance capsule fluence values for applicable CR-3 MIRVSP capsules are listed in Reference 3.

Chemistry Factor Values

The chemistry factor (CF), along with the fluence factor (FF), is used to determine the ΔRT_{NDT}. ΔRT_{NDT} is used to determine the number of capsules that are needed to meet the recommended practices of ASTM E 185-82. The CFs used in this evaluation are presented in Table 2.1.1-2. Chemistry factors are calculated based on Regulatory Guide 1.99, Revision 2, Positions 1.1 and 2.1.

Inlet Temperature

The reactor vessel inlet temperature (T_{COLD}) will change with the proposed EPU operating conditions. For various cases after the EPU, T_{COLD} ranges from 555.4°F to 555.6°F. (Section 1.1, Nuclear Steam Supply System Parameters). The bounding limits for the T_{COLD} are provided in Regulatory Guide 1.99, Revision 2, which is the basis for 10 CFR 50.61, and state that, "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Thus, the T_{COLD} must be greater than 525°F and less than 590°F for the equations and methodology of Regulatory Guide 1.99, Revision 2 to remain valid.

A CR-3 post-EPU specific surveillance capsule program, which is in compliance with ASTM E185-82, was developed in order to effectively monitor the condition of the reactor vessel materials under actual operating conditions as shown in Table 2.1.1-3. Though all capsules which CR-3 is taking credit for have been removed from their respective reactor vessels and tested, the way in which CR-3 complies with ASTM E185-82 is documented to show compliance.

Reactor vessel fluence projections were generated for the proposed EPU following the guidance of Regulatory Guide 1.190 and are presented in Table 2.1.1-1 as well as the reactor vessel fluence projections for the pre-EPU conditions. By comparison, the reactor vessel fluence projections were higher than what was used for the development of the current surveillance capsule schedule for CR-3. There is no impact on the withdrawal of the capsules because all of the capsules used to meet the intent of ASTM E185-82 have been withdrawn from their respective capsules. As presented above, T_{COLD} is maintained above 525°F and below 590°F. Therefore, the equations and results remain valid without adjustments for temperature effects.

<u>Results</u>

Table 2.1.1-3 provides the post-EPU surveillance capsule withdrawal times which meet the intent of ASTM E185-82 as well as listing the pre-EPU surveillance capsule withdrawal times from Reference 3. The five capsules making up the post-EPU surveillance program, listed in Table 2.1.1-3, have been removed from their respective reactor vessels and tested; thereby completing the requirement for capsule withdrawals in accordance with ASTM E185-82.

Five surveillance capsules are required, per ASTM E185-82, because ΔRT_{NDT} exceeds 111°C (200°F) as shown in Table 2.1.1-4.

2.1.1.3 Conclusion

CR-3 has evaluated the effects of the proposed EPU on the integrated reactor vessel surveillance withdrawal schedule and concludes that it has adequately addressed changes in neutron fluence and their effects on the withdrawal times. CR-3 further concludes that the reactor vessel capsule withdrawal schedule, as described in Table 2.1.1-3, is appropriate to ensure that the Reactor Vessel Material Surveillance Program will continue to meet the requirements of 10 CFR 50, Appendix H, and 10 CFR 50.60. Based on the above, the CR-3 material surveillance program will continue to be acceptable following implementation of the proposed EPU, and will continue to meet the requirements of FSAR Sections 1.4.9, 1.4.34, and 1.4.35. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Reactor Vessel Material Surveillance Program.

2.1.1.4 References

1. BAW-1543A, Rev. 4, “Master Integrated Reactor Vessel Surveillance Program,” 1992.
2. BAW-2241PA (Proprietary) Rev. 0, “Fluence and Uncertainty Methodologies,” 1999.
3. BAW-1543, Rev. 4, Supplement 6A, “Supplement to the Master Integrated Reactor Vessel Surveillance Program,” June 2007, NRC Accession number ML072570104.

Table 2.1.1-1 Calculated Maximum Neutron Exposure of the Reactor Pressure Vessel

Reactor Vessel Location	Material ID Heat ID	EOL 32 EFPY (1) Wetted Surface Fluence (n/cm^2), E>1.0 MeV (Pre-EPU)	EOL 50.3 EFPY Wetted Surface Fluence (n/cm^2), E>1.0 MeV (Relative to EPU Power)
Lower Nozzle Belt Forging	AZJ94 123V190	7.08E+18	1.42E+19
Inside Surface Maximum	N/A	8.03E+18	1.57E+19
Upper Shell	C4344-1	7.90E+18	1.56E+19
Upper Shell	C4344-2	7.90E+18	1.56E+19
Lower Shell	C4347-1	8.00E+18	1.57E+19
Lower Shell	C4347-2	8.00E+18	1.57E+19
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	7.08E+18	1.42E+19
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	N/A	N/A
Upper Shell Axial Weld	WF-8 8T1762	7.40E+18	1.46E+19
Upper Shell Axial Weld	WF-18 8T1762	7.40E+18	1.46E+19
Upper/Lower Shell Circ. Weld	WF-70 72105	7.73E+18	1.53E+19
Lower Shell Axial Welds	SA-1580 8T1762	6.96E+18	1.41E+19

(1) 32 effective full-power years (EFPY) is relative to 2544 MWt or 27.5 EFPY relative to EPU power).

Table 2.1.1- 2 Summary of the CR-3 Beltline Material Chemistry Factor Values Based on Regulatory Guide 1.99, Revision 2, Position 1.1. and 1.2

Material	Material ID Heat ID	Chemistry Factor	
		Position 1.1	Position 2.1
Lower Nozzle Belt Forging	AZJ94 123V190	94.0	N/A
Upper Shell	C4344-1	141.8	115.8
Upper Shell	C4344-2	141.8	N/A
Lower Shell	C4347-1	82.6	N/A
Lower Shell	C4347-2	82.6	N/A
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	167.6	N/A
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	143.9	N/A
Upper Shell Axial Weld	WF-8 8T1762	152.4	N/A
Upper Shell Axial Weld	WF-18 8T1762	152.4	N/A
Upper/Lower Shell Circ. Weld	WF-70 72105	199.3	N/A
Lower Shell Axial Welds	SA-1580 8T1762	152.4	N/A

Table 2.1.1-3 Previous and Recommended Surveillance Capsule for CR-3 to Meet ASTM 185-82

	ASTM E185-82 Capsule Program Requirements				
Plant	1.5 EFPY or Fluence > 5E+18 n/cm^2 or $\Delta RT_{NDT} \cong 50°F$ whichever comes first	3 EFPY or Fluence Midway Between First and Third Capsule	6 EFPY or ¼T EOL Fluence whichever comes first	15 EFPY or IS EOL Fluence whichever comes first	EOL or 1-2 Times EOL Fluence (Capsule may be held w/o testing)
CR-3, pre-EPU [Reference 3]	CR-3-B[b]	a	CR-3-C[b]	CR-3-D[b]	CR-3-F[b]
Post-EPU at 50.3 EFPY	CR-3-B[b]	CR-3-C[b]	CR-3-D[b]	CR-3-F[b]	BWOG A5[b]

a – Not needed per ASTM E185-82 based on ΔRT_{NDT} values

b – These capsules have been removed from their respective vessels and tested

c - At 32 EFPY relative to 2544 MWt, or 27.5 EFPY relative to EPU power).

Table 2.1.1-4 ΔRT_{NDT} Values for CR-3 Beltline Materials at 50.3 EFPY

Reactor Vessel Location	Material ID Heat ID	CF (°F)	Inside Wetted Surface Fluence at 50.3 EFPY (1) (n/cm^2), E>1.0 MeV	ΔRT_{NDT} (°F)
Lower Nozzle Belt Forging	AZJ94 123V190	94.0	1.42E+19	103.1
Upper Shell	C4344-1	115.8	1.56E+19	130.0
Upper Shell	C4344-2	141.8	1.56E+19	159.2
Lower Shell	C4347-1	82.6	1.57E+19	92.9
Lower Shell	C4347-2	82.6	1.57E+19	92.9
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	167.6	1.42E+19	183.9
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	143.9	1.42E+19	N/A
Upper Shell Axial Weld	WF-8 8T1762	152.4	1.46E+19	168.4
Upper Shell Axial Weld	WF-18 8T1762	152.4	1.46E+19	168.4
Upper/Lower Shell Circ. Weld	WF-70 72105	199.3	1.53E+19	222.7
Lower Shell Axial Welds	SA-1580 8T1762	152.4	1.41E+19	166.9

(1) Relative to EPU power

2.1.2 Pressure and Temperature Limits and Upper-Shelf Energy

2.1.2.1 Regulatory Evaluation

Pressure and temperature (P/T) limits are established to ensure the structural integrity of the ferritic components of the reactor coolant pressure boundary (RCPB) during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. CR-3's review of P/T limits covered the P/T limits methodology and the calculations for the number of effective full-power years (EFPY) specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics.

The NRC's acceptance criteria for Pressure and Temperature Limits and Upper-Shelf Energy are based on:

- GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating failure, and of gross rupture through its design lifetime;
- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specified conditions, the boundary behaves in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized;
- 10 CFR 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the reactor coolant pressure boundary; and
- 10 CFR 50.60, which requires compliance with the requirements of 10 CFR 50, Appendix G.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating failure, and of gross rupture through its design lifetime; [GDC-14]; and

- FSAR Section 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention and FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention, insofar as it requires that the RCPB be designed with margin sufficient to ensure that, under specified conditions, the boundary behaves in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized. [GDC-31]

Additionally, FSAR Sections 4.3.11.15 provides heatup, cooldown, and pressure limitation criteria for systems insofar as it requires that the technical specifications have been revised to be in accordance with Appendix G of 10 CFR 50 and Appendix G of Section III of the ASME Boiler and Pressure Vessel Code as practicable.

2.1.2.2 Technical Evaluation

Introduction

Reactor vessel integrity is potentially impacted by any change in plant parameters that affect neutron fluence levels or P/T transients, such as the EPU. The changes in neutron fluence resulting from the EPU were evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the CR-3 P/T limits and the projected values of upper shelf energy resulting from the EPU.

Description of Analyses and Evaluations

- EPU Fluence Projections

  Detailed fluence calculations were performed for CR-3 at the EPU conditions. The purpose of these calculations is to show that CR-3 continues to meet the reactor vessel surveillance requirements defined in 10 CFR 50, Appendix H and the guidance provided in Regulatory Guide 1.190 relating to reactor pressure vessel surveillance programs. The fluence methodology is described in AREVA's NRC approved fluence topical report (Reference 1). It has been demonstrated in an independent benchmark experiment (listed in Reference 1) that the results of a fluence analysis that employs the methodology in Reference 1 would be unbiased and have a precision well within the NRC suggested standard deviation (σ) limit of 20% from Regulatory Guide 1.190.

  The calculated (projected) fluence on the vessel was evaluated for the impact of the proposed EPU on the reactor vessel integrity evaluations. Surveillance capsule fluence values for the applicable CR-3 master integrated reactor vessel surveillance program (MIRVSP) capsules are listed in Reference 1. Note that as discussed in Section 2.1.1, Reactor Vessel Material Surveillance Program, all of the necessary capsules for CR-3 have been removed from the appropriate reactor vessels and will not be affected by the EPU. Fluence values are used to calculate the transition temperature shift (ΔRT_{NDT}) for development of the P/T limits.

- Inlet Temperature

  As presented in Section 2.1.1, the reactor inlet temperature ranges from 555.4°F to 555.6°F for post-EPU conditions. For the vessel inlet temperature, the acceptance criteria are from Regulatory Guide 1.99, Revision 2, which is the basis for 10 CFR 50.61, where it is stated that, "The procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement." Thus, the T_{COLD} must be greater than 525°F and less than 590°F for the equations and methodology of 10 CFR 50.61 to remain valid.

- Chemistry Factor Values

  The chemistry values of weight percent copper were used along with fluence to determine the percent decrease in upper-shelf energy (USE) at EPU end of life (EOL). The weight percent copper used in this evaluation is presented in Table 2.1.2-1.

- USE

  The unirradiated USE values for each beltline material were used as a baseline for determining the post-EPU USE. The unirradiated USE values are presented in Table 2.1.2-1, along with the projected USE (including consideration of the EPU).

  The evaluation to assess the impact of the EPU on the USE was performed in two steps. First, new USE values were calculated for all reactor vessel beltline materials using the EPU neutron fluence evaluation results and Figure 2 of Regulatory Guide 1.99, Revision 2. The results are captured in Table 2.1.2-1. Second, the determination was made if an equivalent margins analysis (EMA) needed to be performed based on a review of the USE values calculated for CR-3.

  As is the case for the current term of operation, the Charpy upper-shelf energy (CvUSE) values for some beltline welds are below 50 ft-lb, requiring an EMA for the period of the EPU operation. The methodology used to evaluate CR-3 beltline welds at EOL is consistent with the EMA methods reported in BAW-2192PA, BAW-2178PA, and ANP-10308. The evaluation demonstrates that limiting CR-3 beltline welds WF-70, WF-8, and WF-18 satisfy the acceptance criteria of Appendix K of the Section XI of the ASME Code, and therefore, provide margins of safety equivalent to those of Appendix G of ASME Section XI. It may be concluded that welds WF-70, WF-8, and WF-18 have adequate upper-shelf toughness and satisfy the requirement of Appendix G to 10 CFR 50, Section IV.A.1.a through a reactor vessel life of 50.3 Effective Full Power Year (EFPY) (relative to EPU power) for CR-3.

- P/T Limits

  The current P/T limits are based on Adjusted RT_{NDT} values for projected fluence values at 32 EFPY without an EPU.

- Applicability of P/T Limits Curves

  Based on the post-EPU reactor vessel fluence projections, the current P/T limit curves are bounding for CR-3 up to 27.5 EFPY. The P/T limit curves were developed in accordance with the requirements of 10 CFR 50, Appendix G, utilizing the analytical methods of topical report BAW-10046A, and ASME Code Section XI, Appendix G.

  Maintaining the existing P/T limits until 27.5 EFPY continues to provide conservatism for the EPU conditions. Implementation of revised P/T limits will require a separate submittal which will be made at least 12 months prior to reaching 27.5 EFPY (relative to the EPU power).

  Based on the post-EPU reactor vessel fluence projections, new P/T limit curves were developed for CR-3 at 50.3 EFPY (relative to the EPU power). The P/T limit curves were developed in accordance with the requirements of 10 CFR 50, Appendix G, utilizing the analytical methods of topical report BAW-10046A, and ASME Code Section XI, Appendix G. The 50.3 EFPY (relative to the EPU power) curves provide more operating room than the 32 EFPY (without an EPU) curves because of the use of the ASME Boiler & Pressure Vessel Code 2001 through 2003 Addenda that incorporated Code Cases N-588 and N-640.

Results

- Applicability of Heatup and Cooldown P/T Limit Curves

  Using the post-EPU reactor vessel fluence projections, the current P/T limit curves are bounding for CR-3 up to 27.5 EFPY.

- USE

  Based on the EPU fluence projections, all nozzle and plate beltline materials are expected to have a USE greater than 50 ft-lb through 50.3 EFPY (relative to the EPU power), as required by 10 CFR 50, Appendix G. However some of the beltline weld materials are expected to have a USE less than 50 ft-lb through 50.3 EFPY (relative to the EPU power). The 50.3 EFPY (relative to the EPU power) USE values, as presented in Table 2.1.2-1 were predicted using the 1/4-thickness (¼T) fluence projections.

  An EMA showing sufficient margin has been performed. The CR-3 fluences for WF-8, WF-18, and WF-70 at 48 EFPY (without an EPU) as shown in the base analysis for all of the Babcock & Wilcox (B&W) plants were lower than the CR-3 predicted fluences for WF-8, WF-18, and WF-70 at 50.3 EFPY (relative to the EPU power) so a reconciliation was performed. The base analysis demonstrated that the limiting weld was the Three Mile Island Unit 1 weld SA-1526. The EMA analysis for all service loads demonstrated that welds WF-8, WF-18 and WF-70 used in CR-3 all have higher $J_{0.1}$/ J_1 ratios compared to TMI-1 weld SA-1526. The increased fluence at 50.3 EFPY (relative to the EPU power) for CR-3 does not change the selection of the limiting TMI-1 weld SA-1526 for the evaluation. Welds WF-8, WF-18, and WF-70 used in CR-3 satisfy the acceptance criteria of Appendix K of the Section XI of the ASME Code, and therefore, provide margins of safety equivalent to those of Appendix G of ASME Section XI.

- Inlet Temperature

  The range of T_{COLD} will not be affected by the EPU in that it will be maintained above 525°F and below 590°F (see Section 2.1.1, Reactor Vessel Material Surveillance Program).

2.1.2.3 Conclusion

CR-3 has evaluated the effects of the proposed EPU on the P/T Limits for the plant and concludes that it has adequately addressed changes in neutron fluence and their effects on P/T limits. CR-3 further concludes that the current P/T limits remain valid for operation under the proposed EPU conditions up to 27.5 EFPY. Based on this, the CR-3 concludes that the current Pressure-Temperature Limits and Upper Shelf Energy will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with FSAR Sections 1.4.9, 1.4.34, and 1.4.35 in this respect following implementation of the proposed EPU. Therefore, the CR-3 finds the proposed EPU acceptable with respect to the current Pressure-Temperature Limits and Upper Shelf Energy up to 27.5 EFPY (relative to the EPU power).

2.1.2.4 References

1. AREVA NP Document BAW-2241PA (Proprietary), Rev. 0, “Fluence and Uncertainty Methodologies,” February 1999.

2. AREVA NP Document BAW-1543, Rev. 4, Supplement 6A, “Supplement to the Master Integrated Reactor Vessel Surveillance Program,” June 2007, NRC Accession number ML072570104.

Table 2.1.2-1 Predicted (50.3 EFPY) USE Calculations for all Beltline Region Materials

Material	Material ID Heat ID	Copper (wt%)	¼T Fluence (n/cm^2), E>1.0 MeV	Unirradiated USE (ft-lb)	Projected USE Decrease (%)	Projected USE (ft-lb) at 50.3 EFPY
Lower Nozzle Belt Forging	AZJ94 123V190	0.13	8.94E+18	109	21	86
Upper Shell	C4344-1	0.20	9.87E+18	88	28	63
Upper Shell	C4344-2	0.20	9.87E+18	88	28	63
Lower Shell	C4347-1	0.12	9.93E+18	119	21	95
Lower Shell	C4347-2	0.12	9.93E+18	86	21	68
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	0.23	8.94E+18	70	35	45[b]
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	0.16	N/A[a]	70	N/A	N/A
Upper Shell Axial Weld	WF-8 8T1762	0.19	9.23E+18	70	32	48[b]
Upper Shell Axial Weld	WF-18 8T1762	0.19	9.23E+18	70	32	48[b]
Upper/Lower Shell Circ. Weld	WF-70 72105	0.32	9.70E+18	70	42	41[b]
Lower Shell Axial Welds	SA-1580 8T1762	0.19	8.88E+18	70	32	48[b]
Lower Shell to Dutchman	WF-154	0.27	1.16E+17	70	24	53

a – The USE for this material was not calculated because this material is used for the outer 60% of the diameter of the weld. USE is calculated using the fluence at 25% of the way through the wall, where this weld material does not exist.

b – An equivalent margins analysis (EMA) was performed for this material since the value is below the required 50 ft-lb. The EMA has demonstrated acceptability as reported in Section 2.1.2.2.

c - Relative to EPU power

2.1.3 Pressurized Thermal Shock

2.1.3.1 Regulatory Evaluation

The Pressurized Thermal Shock (PTS) evaluation provides a means for assessing the susceptibility of the reactor vessel beltline materials to PTS events to ensure that adequate fracture toughness is provided for supporting reactor operation. CR-3's review covered the PTS methodology and the calculations for the reference temperature (RT_{PTS}) at the expiration of the licensing, considering neutron embrittlement effects.

The NRC's acceptance criteria for Pressurized Thermal Shock are based on:

- GDC-14 insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture, and of gross rupture;
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that under specified conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized; and
- 10 CFR 50.61, insofar as it sets fracture toughness criteria for protection against PTS events.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture, and of gross rupture [GDC-14]; and
- FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention and FSAR Section 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention, insofar as it requires that the reactor coolant pressure boundary be designed with sufficient margin to assure that under specified conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. [GDC-31]

2.1.3.2 Technical Evaluation

Introduction

Reactor vessel integrity is potentially impacted by any changes in plant parameters that affect neutron fluence levels or temperature and pressure transients. The changes in neutron fluence resulting from the proposed EPU have been evaluated to determine the impact on reactor vessel integrity. The assessment presented herein focuses on the RT_{PTS} evaluation and any changes resulting from the proposed EPU.

Description of Analyses and Evaluations

EPU Fluence Projections

Detailed fluence calculations were performed for CR-3 at EPU conditions. The calculated (projected) fluence on the vessel was evaluated for the impact of the proposed EPU on the reactor vessel integrity evaluations. These fluence projections are presented in Table 2.1.3-1. Surveillance capsule fluence values are also provided in Reference 1. See Section 2.1.1, Reactor Vessel Material Surveillance Program, for a discussion of CR-3's surveillance program.

Fluence values are used to calculate the transition temperature shift (ΔRT_{PTS}) in the PTS equation from 10 CFR 50.61.

Chemistry Factor Values

The chemistry factors (CFs) and the fluence factor (FF) are used to determine the ΔRT_{PTS} resulting from the proposed EPU. The CFs used in this evaluation are presented in Table 2.1.3-3.

Initial Reference Temperature, Nil-Ductility Temperature (RT_{NDT})

The initial RT_{NDT} values are the baseline reference temperature for each material and were used to determine the RT_{PTS} along with adjusted ΔRT_{NDT} and margins resulting from the proposed EPU. The initial RT_{NDT} values used in this evaluation are presented in Table 2.1.3-2.

Inlet Temperature

As presented in Section 2.1.1, the reactor vessel inlet temperature (T_{COLD}) will change with the proposed EPU. For T_{COLD}, the acceptance criteria are provided in Regulatory Guide 1.99, Revision 2, and state that the procedures are valid for a nominal irradiation temperature of 550°F. Irradiation below 525°F should be considered to produce greater embrittlement, and irradiation above 590°F may be considered to produce less embrittlement. Thus, T_{COLD} must be greater than 525°F and less than 590°F for the equations and methodology of 10 CFR 50.61 to remain valid.

Results

An evaluation of the impact of the proposed EPU on PTS was performed for CR-3. PTS calculations were performed for all the beltline materials of the CR-3 reactor vessel under the EPU conditions using the rules from 10 CFR 50.61. The results of these calculations are presented in Table 2.1.3-3. Note that the controlling beltline materials for CR-3 are the upper shell longitudinal welds, WF-8 and WF-18, with predicted RT_{PTS} values of 231.9°F. Based on the results shown in Table 2.1.3-3, all RT_{PTS} values remained below the NRC screening criteria values using the projected EPU fluence projections at 50.3 EFPY of 270°F for plates, forgings, and axial weld materials and 300°F for circumferential weld materials.

2.1.3.3 Conclusion

CR-3 has reviewed the evaluation of the effects of the proposed EPU on the PTS for the plant and concludes that the evaluation has adequately addressed changes in neutron fluence and their effects on PTS. CR-3 further concludes that the evaluation has demonstrated that the plant will continue to meet the requirements of FSAR Sections 1.4.9, 1.4.34, and 1.4.35. Therefore, CR-3 finds the proposed EPU acceptable with respect to Pressurized Thermal Shock.

2.1.3.4 References

1. BAW-1543, Rev. 4, Supplement 6A “Supplement to the Master Integrated Reactor Vessel Surveillance Program,” (), NRC Accession number ML072570104.

Table 2.1.3-1. Calculated Maximum Neutron Exposure of the Reactor Pressure Vessel at 60 Calendar Years 50.3 EFPY

Reactor Vessel Location	Material ID Heat ID	EOL 50.3 EFPY Wetted Surface Fluence (Relative to EPU power) (n/cm^2), E>1.0 MeV
Lower Nozzle Belt Forging	AZJ94 123V190	1.42E+19
Inside Surface Maximum	N/A	1.57E+19
Upper Shell	C4344-1	1.56E+19
Upper Shell	C4344-2	1.56E+19
Lower Shell	C4347-1	1.57E+19
Lower Shell	C4347-2	1.57E+19
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	1.42E+19
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	N/A
Upper Shell Axial Weld	WF-8 8T1762	1.46E+19
Upper Shell Axial Weld	WF-18 8T1762	1.46E+19
Upper/Lower Shell Circ. Weld	WF-70 72105	1.53E+19
Lower Shell Axial Welds	SA-1580 8T1762	1.41E+19

Table 2.1.3-2. Summary of the Initial RT_{NDT} Values for CR-3

Reactor Vessel Location	Material ID Heat ID	Initial RT_{NDT} (°F)
Lower Nozzle Belt Forging	AZJ94 123V190	3
Upper Shell	C4344-1	20
Upper Shell	C4344-2	20
Lower Shell	C4347-1	-10
Lower Shell	C4347-2	45
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	10
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	-5
Upper Shell Axial Weld	WF-8 8T1762	-5
Upper Shell Axial Weld	WF-18 8T1762	-5
Upper/Lower Shell Circ. Weld	WF-70 72105	-26
Lower Shell Axial Welds	SA-1580 8T1762	-5

Table 2.1.3-3. RT_{PTS} Calculations for CR-3 Beltline Region Materials at 50.3 EFPY

Reactor Vessel Location	**Material ID** **Heat ID**	**Inside Wetted Surface Fluence** (x10^{19} n/cm^2, E>1.0 MeV)	**Fluence Factor**	**Chemistry Factor** (°F)	**ΔRT_{PTS}** (°F)	**Margin** (°F)	**Initial RT_{NDT}** (°F)	**50.3 EFPY** (1) **RT_{PTS}** (°F)
Lower Nozzle Belt Forging	AZJ94 123V190	1.42	1.097	94.0	103.1	70.7	3	176.9
Upper Shell	C4344-1	1.56	1.123	115.8	130.0	17.0	20	167.0
Upper Shell	C4344-2	1.56	1.123	141.8	159.2	34.0	20	213.2
Lower Shell	C4347-1	1.57	1.125	82.6	92.9	34.0	-10	116.9
Lower Shell	C4347-2	1.57	1.125	82.6	92.9	34.0	45	171.9
Upper Shell Circ. Weld (ID 40%)	SA-1769 71249	1.42	1.097	167.6	183.9	56.0	10	249.9
Upper Shell Circ. Weld (OD 60%)	WF-169-1 8T1554	N/A	N/A	143.9	N/A	N/A	-5	N/A
Upper Shell Axial Weld	WF-8 8T1762	1.46	1.105	152.4	168.4	68.5	-5	231.9
Upper Shell Axial Weld	WF-18 8T1762	1.46	1.105	152.4	168.4	68.5	-5	231.9
Upper/Lower Shell Circ. Weld	WF-70 72105	1.53	1.118	199.3	222.7	56.0	-26	252.7
Lower Shell Axial Welds	SA-1580 8T1762	1.41	1.095	152.4	166.9	68.5	-5	230.4

(1) Relative to EPU power

2.1.4 Reactor Internals and Core Support Materials

2.1.4.1 Regulatory Evaluation

The Reactor Internals and Core Supports include SSCs that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCS). CR-3's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation.

The NRC's acceptance criteria for Reactor Internals and Core Supports are based on:

- GDC-1, and
- 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria:

- FSAR Section 1.4.1, Quality Standards, for material specifications, controls on welding, and inspection of reactor internals and core supports. [GDC-1]

Additionally, FSAR Section 4.4.1 provides criteria for material specifications, controls on welding, and inspection of reactor internals and core supports in conformance with 10 CFR 50.55a.

2.1.4.2 Technical Evaluation

<u>Introduction</u>

CR-3 is an active participant in the industry's Electric Power Research Institute (EPRI) Material Reliability Project (MRP) research program on aging related degradation of reactor vessel (RV) internals components. The MRP program has evaluated the internals of the United States (US) Pressurized Water Reactors (PWRs) including Babcock & Wilcox units for various aging degradation mechanisms during the existing and extended license periods. Based on the results of the evaluations, an industry generic RV internals aging management strategy (MRP-227) (Reference 1) for the US PWRs was developed and issued in 2008. EPRI is currently in the process of submitting MRP-227 for NRC approval. CR-3 will evaluate and implement the applicable guidelines pertinent to CR-3. A CR-3 unit specific internals aging management plan, based on the generic MRP-227 inspection and evaluation guidelines, will become the new licensing basis for the EPU cycles.

Additional analyses based on the EPU conditions have been performed to ensure that the industry generic MRP-227 inspection and evaluation guidelines, which are based on a generic evaluation for a 60-year operating lifetime, bound the CR-3 EPU conditions. The objective of the EPU evaluation is to ensure that the EPU environmental conditions (chemistry, temperature, and fluence) will not introduce any new aging effects on the RV internals components, nor change the manner in which the component aging will be managed by the aging management program. The relevant potential aging degradation mechanisms by the EPU are:

- Thermal Aging Embrittlement (TE)
- Irradiation Embrittlement (IE)
- Stress Corrosion Cracking (SCC)
- Irradiation Assisted SCC (IASCC)
- Void swelling (VS)
- Irradiation Stress Relaxation and Creep, Fatigue, and Wear

The results of the CR-3 EPU-specific assessment of these aging degradation mechanisms are summarized below.

<u>Input Parameters, Assumptions, and Acceptance Criteria</u>

The CR-3 Reactor Coolant System (RCS) water chemistry will continue to follow the latest revision of the EPRI PWR water chemistry guidelines after the EPU implementation. Potential changes in lithium/boron concentration and pH value of the RCS to optimize fuel performance after the EPU, as allowed by the EPRI PWR water chemistry guidelines, have been considered in this evaluation.

The maximum neutron dose in dpa (displacement per atom) of the CR-3 reactor internals for an operating period of 60 calendar years (49.06 Effective Full Power Year (EFPY)) including the EPU cycles was evaluated. The maximum exposure occurs on the inside surface of the baffle plates opposite the central sections of the reactor core. These maximum assumed neutron exposure values from the CR-3 analysis and MRP-229 analysis are compared in Table 2.1.4-1.

<u>Description of Analyses and Evaluations</u>

In 2008, EPRI performed a functionality analysis for the B&W-designed PWR Internals and the results of the functionality analysis are summarized in the EPRI MRP-229-Rev. 1 report (Reference 2). The MRP-229-Rev. 1 functionality analysis results were used to develop the aging management strategies for the B&W-design PWR internals and are summarized in the EPRI MRP-231-Rev. 1 report (Reference 3), which in turn provided the inspection requirements listed in MRP-227 for the B&W-designed PWR internals. Since the CR-3 EPU conditions were not considered by the functionality analysis in MRP-229-Rev. 1 and the aging management strategies in MRP-231-Rev. 1, the EPU conditions were not considered by MRP-227. Additional analyses for CR-3 have been performed to ensure that the MRP-227 generic inspection and evaluation guidelines will bound the CR-3 EPU conditions.

Thermal Aging Embrittlement (TE)

The following CR-3 RV internals items have been identified to be above the thermal embrittlement screening threshold and have augmented inspection requirements specified in MRP-227:

- Control rod guide tube (CRGT) assembly spacer castings (made from CF3M)*
- Core support shield (CSS) assembly vent valve discs (made from CF8)
- CSS assembly vent valve top and bottom retaining ring (made from Type 15-5PH)
- CSS assembly vent valve disc shaft or hinge pin (made from Type 431 SS)
- Incore monitoring instrumentation (IMI) assembly guide tube spiders (made from CF8)

*The CRGT spacer castings are categorized as “Expansion” while the rest are categorized as “Primary” by MRP-227. Implementation of the augmented inspection of the “Expansion” items such as CRGT spacer castings is not required unless problems are first found when inspecting the triggering “Primary” component(s).

These items are made of cast austenitic stainless steels (CASS), martensitic precipitation hardenable stainless steel, or martensitic stainless steel. Thermal aging embrittlement of these materials is a thermally activated process. The screening is based on whether the components are known to be susceptible to thermal embrittlement at normal PWR temperatures. The screening is not tied to any specific threshold temperature. Therefore, minor changes in operating temperature due to the EPU would not affect the thermal embrittlement screening results.

Changes in the material microstructure due to thermal embrittlement cause loss of ductility and fracture toughness, and degradation of impact properties. These changes are usually accompanied by an increase in hardness and strength. For the items listed above, their thermal embrittlement kinetics would change with the change in the exposure temperature. Under normal full power operating condition, the T_{HOT} will increase from 602.1°F to 608.7°F, the T_{COLD} will decrease from 555.9°F to 555.4°F, and the T_{AVG} will increase from 579°F to 582°F after implementing the EPU. The CRGT spacer castings and CSS vent valve items are exposed to the T_{HOT} and therefore will see a maximum temperature increase of 6.6°F after the EPU (Table 1.1-1).. The IMI guide tube spiders are exposed to the T_{COLD}, so will be unaffected by the EPU. The 6.6°F increase will slightly accelerate the thermal embrittlement process. The maximum embrittlement (such as total potential loss of fracture toughness) will be unaffected by the temperature increase. Since the decline rate in fracture toughness is the highest at first and then gradually levels off with operating time, most of the embrittlement has already taken place before the EPU implementation. The higher T_{HOT} will not cause significant changes in the thermal embrittlement process of these components after the EPU. Therefore, it is concluded that the MRP-227 requirement for thermal embrittlement will not be affected by the proposed EPU.

Irradiation Embrittlement (IE)

The RV internals component items identified by MRP-227 for augmented inspections for irradiation embrittlement are located in the core barrel assembly, and the lower grid support pad items and the IMI guide tube spiders immediately below the core barrel assembly. For a given material and composition, irradiation embrittlement correlates with the neutron dose level. The neutron flux during the EPU cycles has been analyzed and compared to the neutron flux used for the generic MRP-229-Rev. 1 functionality evaluation.

The EPU neutron flux is comparable to the MRP-229-Rev. 1 high leakage flux, but higher than the MRP-229-Rev. 1 low leakage flux. The CR-3 EPU analysis showed that the cumulative lifetime fluence including the EPU cycles for the CR-3 core barrel assembly is below the 60-year lifetime fluence for the generic B&W PWR core barrel assembly in the MRP-229-Rev. 1. The total lifetime neutron dose and the corresponding irradiation embrittlement will be bounded by the generic analysis in the MRP-229-Rev. 1. The change in neutron flux during the EPU cycles would not cause additional internals items to exceed the irradiation embrittlement screening threshold. Therefore, it is concluded that the MRP-227 requirement for irradiation embrittlement will not be affected by the proposed CR-3 EPU.

<u>Stress Corrosion Cracking (SCC)</u>

The RV internals component items identified by the MRP-227 for augmented inspection for SCC are the following high-strength bolts made from either Alloy A-286 Condition A or Alloy X-750 materials:

- Upper Core Barrel (UCB) Bolts
- Lower Core Barrel (LCB) Bolts
- Flow Distributor (FD) Bolts*
- Upper Thermal Shield (UTS) Bolts*
- Lower Thermal Shield (LTS) Bolts*
- Surveillance Specimen Holder Tube (SSHT) Bolts*

  *These bolts are categorized as “Expansion” while the UCB and LCB bolts are categorized as “Primary” by MRP-227. Implementation of the augmented inspection of the “Expansion” bolts is not required unless problems are first found when inspecting the “Primary” UCB and LCB bolts.

The CR-3 RCS water chemistry will continue to follow the latest revision of the EPRI PWR water chemistry guidelines after the implementation of the EPU. Any changes in lithium/boron concentration and pH value of the RCS to optimize fuel performance after the EPU, as allowed by the EPRI PWR water chemistry guidelines, are not expected to have any adverse effect on the SCC of the internals items. The increase of 6.6°F in the T_{HOT} after implementing the EPU will not have a significant adverse effect on SCC. However, no additional internals items will exceed the SCC screening threshold due to changes in lithium/boron concentration and minor increase in T_{HOT} after the EPU.

Of the six high-strength bolt locations susceptible to SCC, only the UCB bolts are exposed to the T_{HOT} and could be adversely affected. The other locations are exposed to the T_{COLD} and will be unaffected after the EPU.

All 120 UCB bolts at CR-3 were replaced in 1983 with modified Alloy A-286 bolts. All 120 replacement UCB bolts were inspected using ultrasonic testing (UT) and were found to be free of any rejectable indications in 1985 and again in 1996. Currently, the UCB bolts were UT inspected in the Fall of 2009 in accordance with the MRP-227 baseline inspection schedule, with no indications. Subsequent inspection requirements depend on evaluating the baseline UT inspection results of all seven B&W designed PWRs.

The temperature effect on SCC initiation and crack growth rate has been quantitatively modeled utilizing the Arrhenius equation (activation energy). However, SCC initiation is also sensitive to stress level, bolt heat chemical composition, and fabrication history. Currently, there is insufficient data to quantify the activation energy for Alloy A-286 bolt SCC initiation. With certain allowances, i.e., uncertainties associated with bolt heat chemical composition and variation in fabrication and installation torque, the effects of minor temperatures increases after the EPU is within the conditions considered by the MRP-227 inspection requirement for the UCB bolt SCC. The effect of higher T_{HOT} after the EPU will be considered in the subsequent inspection requirement for the UCB bolts. Therefore, it is concluded that the MRP-227 requirement for SCC will not be affected by the proposed EPU.

Irradiation Assisted SCC (IASCC)

The RV internals component items exceeding the IASCC screening threshold established in EPRI MRP-175 report (Reference 4) are located within the core barrel assembly due to the high fluence level. The new CR-3 fluence analysis shows that the end-of-life fluence after the EPU implementation is bounded by MRP-229-Rev. 1. Therefore, no additional internals items outside the core barrel assembly would exceed the IASCC screening threshold after the EPU. Additional analyses of the entire core barrel assembly have been performed for CR-3 to ensure that MRP-227 bounds the CR-3 EPU conditions for the IASCC in the core barrel assembly.

The CR-3 analyses take into account the effect of stress relaxation, temperature, neutron flux, and void swelling induced distortion on IASCC during the EPU cycles. The results show that, similar to the MRP-229-Rev. 1 results, the IASCC is not a concern during the lifetime for the CR-3 core barrel, baffle plates, and former plates. In addition, the IASCC susceptibility (including pre-EPU and EPU cycles) for the CR-3 baffle-to-baffle bolts, core barrel-to-former bolts, and the baffle-to-former bolts is similar to the MRP-229-Rev. 1. IASCC for the CR-3 internals after the EPU will be bounded by the MRP-229-Rev. 1 results. Therefore, it is concluded that the MRP-227 requirement for IASCC will not be affected by the proposed CR-3 EPU.

Void Swelling (VS)

The RV internals component items exceeding the void swelling screening threshold values are within the core barrel assembly due to a combination of high fluence and high temperature. The amount of void swelling in the core barrel assembly after 60 years is predicted to be insignificant by the MRP-229-Rev. 1 functionality analysis. The peak 60-year void swelling is below 5% and the average void swelling in the core barrel assembly is below 1%. Therefore, it is concluded that void swelling is not a credible concern for the B&W-design PWR internals items. As a result, there is no augmented inspection requirement specifically for void swelling in the MRP-227 for the B&W-design PWRs.

Additional analyses of the entire core barrel assembly have been performed for CR-3 to ensure that the MRP-229-Rev. 1 conclusion bounds the CR-3 EPU conditions. The analyses include the effect of fluence and temperature on void swelling during the EPU cycles. The results indicate that the higher gamma heating rates will lead to higher peak temperatures at the reentrant corners of the baffle and former plates during the EPU cycles. As a result, the predicted peak void swelling value after 60 years is slightly higher for CR-3 than that of the MRP-229-Rev. 1 (4.0% for CR-3 vs. 3.3% for MRP-229-Rev. 1) even though the total 60-year fluence is lower for the CR-3 core barrel assembly. The average void swelling in the CR-3 core barrel assembly after 60 years is below 1% and comparable to that of the MRP-229-Rev. 1. The previous conclusion on void swelling will remain valid for the CR-3 internals after the EPU, since the peak 60-year void swelling, including the EPU cycles, remains below 5%. Therefore, it is concluded that the MRP-227 inspection requirement for void swelling will not be affected by the proposed CR-3 EPU.

<u>Irradiation Stress Relaxation and Creep, Fatigue and Wear</u>

The RV internals bolted items identified by the MRP-227 for augmented inspection for irradiation stress relaxation and creep, and the associated fatigue and wear are located within the core barrel assembly. Irradiation stress relaxation and creep, fatigue, and wear are a function of fluence. The new CR-3 fluence analysis shows that the end-of-life fluence including the EPU cycles will be bounded by the MRP-229-Rev. 1 fluence. Therefore, no additional bolting items outside the core barrel would become susceptible to irradiation stress relaxation and creep after the EPU. The bolting items in the core barrel assembly will be bounded by the previous results in the MRP-227. Therefore, it is concluded that the MRP-227 requirement for irradiation stress relaxation and creep, fatigue, and wear will not be affected by the proposed CR-3 EPU.

<u>Results</u>

The results of the RV internals material aging degradation assessment showed that the materials aging degradations during the proposed EPU at CR-3 will be bounded by the conditions used for developing the EPRI MRP-227 Inspection and Evaluation Guidelines. It was also concluded that the new EPU environmental conditions (chemistry, temperature, and fluence) will not introduce any new aging effects on the RV internals components, nor will the EPU change the manner in which the component aging degradations will be managed by the CR-3 specific aging management program to be developed based on MRP-227.

2.1.4.3 Conclusion

CR-3 has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of Reactor Internal and Core Support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. CR-3 further concludes that the licensee has demonstrated that the Reactor Internal and Core Support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Reactor Internal and core Support Materials

2.1.4.4 References

1. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Rev. 0). EPRI, Palo Alto, CA: 2008. 1016596.

2. Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229-Rev. 1). EPRI, Palo Alto, CA: 2009. 1019090.

3. Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231-Rev. 1). EPRI, Palo Alto, CA: 2009. 1019092.

4. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.

Table 2.1.4-1. Comparison of Maximum Cumulative Neutron Dose in dpa (displacement per atom) in the Reactor Pressure Vessel Internals

Analysis	Operating Time	Maximum accumulated does in dpa (highest fluence location in the internals)
MRP-229	60-year (60 EFPY)	127.14 dpa
CR-3 with EPU	60-year (49.06 EFPY)	121.2 dpa

2.1.5 Reactor Coolant Pressure Boundary Materials

2.1.5.1 Regulatory Evaluation

The reactor coolant pressure boundary (RCPB) defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The CR-3 evaluation of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs.

The NRC's acceptance criteria for Reactor Coolant Pressure Boundary Materials are based on:

- GDC-1 and 10 CFR 50.55a, insofar as they require that structures, systems, and components (SSCs) important-to-safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed;
- GDC-4, insofar as it requires that SSCs important-to-safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;
- GDC-14, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture,
- GDC-31, insofar as it requires that the RCPB be designed with margin sufficient to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized, and
- 10 CFR 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.1, Quality Standards, FSAR Section 1.4.5, Records Requirements, and 10CFR50.55, insofar as they require that SSCs important-to-safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed. [GDC-1]
- FSAR, Section 1.4.9 Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture. [GDC-14]

- FSAR, Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention, and FSAR Section 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention, insofar as they require that the RCPB be designed with margin sufficient to ensure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized. [GDC-31]

Additionally, FSAR Section 4.1 provides design basis criteria for the Reactor Coolant System (RCS) insofar as it requires that SSCs important-to-safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. [GDC-4]

The CR-3 response to NRC Generic Letter (GL) 97-01 is documented in the following letters;

- FPC to NRC letter dated July 18, 2001, “Crystal River Unit 3 - Commitment Change Regarding Control Rod Drive Mechanism (CRDM) Nozzle Inspection Plans” (ML012120288).
- FPC to NRC letter (3F0199-06) dated January 14, 1999, “NRC Request for Additional Information Regarding the Response to Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations”.
- FPC to NRC letter (3F0797-03) dated July 29, 1997, “Generic Letter 97-01, ‘Degradation of Control Rod Mechanism Nozzle and Other Vessel Head Penetrations’”

CR-3 evaluated NRC Information Notice (IN) 2000-17 which resulted in plant walkdowns and visual inspections.

The CR-3 response to NRC Bulletin (BL) 2001-01 is documented in the following letters:

- FPC to NRC letter dated August 30, 2001, “Crystal River Unit 3 - Response to NRC Bulletin 2001-01, ’Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles’" (ML012490152)
- FPC to NRC letter dated November 19, 2001, “Crystal River Unit 3 – Information Requested in Item 5 of NRC Bulletin 2001-01, ‘Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles’" (ML020160467)

The CR-3 response to NRC Bulletin (BL) 2002-01 is summarized in FPC letter dated July 24, 2002, “Crystal River Unit 3 - Supplemental Information Regarding the 15-Day Response to NRC Bulletin 2002-01, ‘Reactor Pressure Vessel Head Degradation and Reactor’”. (ML022130350)

The CR-3 response to BL 2002-02 is included in FPC letter dated August 22, 2002, “Crystal River Unit 3 - Response to NRC Bulletin 2002-02, ‘Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs’” (ML022460249).

Additionally, FSAR Sections 4.3.11.15 provides heatup, cooldown, and pressure limitation criteria for systems insofar as it requires that the Improved Technical Specifications have been revised to be in accordance with Appendix G of 10 CFR 50 and Appendix G as practicable.

2.1.5.2 Technical Evaluation

Introduction

The EPU evaluation assessed the potential effect of changes in the Reactor Coolant System (RCS) chemistry (impurities), pH conditions, and EPU service temperatures on the integrity of primary component pressure boundary materials during service. The evaluation includes:

- An assessment of the potential effect of water chemistry changes on the (i) general corrosion (wastage) of carbon steel components, and (ii) stress corrosion cracking (SCC) of system austenitic stainless steel materials, and the management strategy of any issues there from.
- An assessment of the effect of change in the service temperature on (i) primary water stress corrosion cracking (PWSCC) of Alloy 600/182/82 nickel base alloys, and (ii) cast austenitic stainless steel (CASS) materials and the management strategy of any issues there from.

Input Parameters, Assumptions, and Acceptance Criteria

The CR-3 RCS water chemistry will continue to follow the latest revision of the EPRI PWR water chemistry guidelines after the implementation of the EPU. Any changes in lithium/boron concentration and pH value of the RCS to optimize fuel performance after the EPU, as allowed by the EPRI PWR water chemistry guidelines, are not expected to produce any undesirable material integrity issues.

Under normal full power operating condition, the T_{HOT} will increase from 602.1°F to 608.7°F, the T_{COLD} will decrease from 555.9°F to 555.4°F, and the T_{AVG} will increase from 579°F to 582°F after implementing EPU. Therefore, the T_{HOT} will see a maximum temperature increase of 6.6°F after the EPU (Table 1.1-1).

Description of Analyses and Evaluations

The effect of change in service conditions (temperature and water chemistry) due to the proposed EPU on the performance of the RCPB materials has been evaluated, as described below.

- General Corrosion/Wastage of Carbon Steel Components

  Experience with operating plants as well as with the guidelines provided by EPRI PWR Water Chemistry Guidelines suggest that changes in lithium concentrations with controlled boron concentrations after the EPU for optimum fuel performance do not produce any undesirable material integrity issues.

  The CR-3 Boric Acid Corrosion Program was developed in response to the recommendations of NRC Generic Letter 88-05. The Boric Acid Corrosion Program implements systematic measures to ensure that leaking borated coolant does not lead to the degradation of the leakage source or adjacent mechanical, electrical and structural components susceptible to boric acid corrosion. The Program consists of: (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of the leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) a follow-up inspection for adequacy of corrective actions. The Boric Acid Corrosion Program includes plant-specific RCPB boric acid leakage identification and inspection procedures to ensure that leaking borated coolant does not lead to degradation of the leakage source or adjacent structures, and provides assurance that the RCPB will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

Implementation of the Boric Acid Corrosion Program will provide reasonable assurance that applicable aging effects will be managed such that the components susceptible to boric acid corrosion will continue to perform their intended functions. The Boric Acid Corrosion Program will not be affected by the EPU.

- SCC of Austenitic Stainless Steels

  The two degradation mechanisms that are operative in the pressure boundary austenitic stainless steel (base and weld) materials in the RCPB are intergranular stress corrosion cracking (IGSCC) and transgranular stress corrosion cracking (TGSCC). Susceptible materials, sensitized microstructure, and the presence of oxygen are required for the occurrence of IGSCC, while the introduction of halogens such as chlorides and the presence of oxygen are prerequisites for the occurrence of TGSCC. The RCS water chemistry will continue to follow the latest revision of the EPRI PWR water chemistry guidelines after implementing the EPU. Any changes in lithium/boron concentration and pH value of the RCS to optimize fuel performance after the EPU, as allowed by the EPRI PWR water chemistry guidelines, are not expected to produce any detrimental effect on the SCC of the stainless steel components in the RCS pressure boundary.

- Alloy 600/82/182 Components at CR-3

  The CR-3 RCS water chemistry will continue to follow the latest revision of the EPRI PWR water chemistry guidelines after the implementation of the EPU. A review of operating experience regarding effects of lithium and pH on PWSCC by EPRI PWR Water Chemistry Guidelines suggests that there are no adverse effects on PWSCC from the lithium/boron concentration range. Therefore, there is no PWSCC impact from the EPU water chemistry.

  T_{HOT} will see a maximum temperature increase of 6.6°F after the EPU while the T_{COLD} is basically unchanged. The RCS pressure and the T_{SAT} in the pressurizer are unaffected by the EPU. Therefore, the PWSCC impact from the EPU would be limited to any Alloy 600/82/182 in the reactor pressure vessel (RPV) closure head, Once-Through Steam Generators (OTSGs), and hot leg piping exposed to the T_{HOT}.

  The original CR-3 RPV closure head containing Alloy 600/82/182 penetration nozzles and J-groove welds was replaced in 2003. CR-3 replaced the OTSG and hot leg in the Fall of 2009. PWSCC impact to the surge line pressurizer nozzle was mitigated in the fall of 2007. The surge line hot leg nozzle impact was mitigated in the Fall of 2009. The 12” decay heat nozzle Alloy 82/182 butt weld was mitigated in the Spring of 2008. In all cases, the replacement or mitigation is with Alloy 690/52/152. Therefore, the EPU conditions will have no impact on the Alloy 600/82/128 PWSCC in the RCS pressure boundary.

- PWSCC Susceptibility of RPV Head

  At CR-3, the original RPV closure head with Alloy 600/82/182 penetrations was replaced in 2003 with a new closure head with Alloy 690/52/152 penetrations. Laboratory and field experience to date suggests that Alloy 690 and associated Alloy 52/152 welds are resistant to PWSCC as described in EPRI MRP-111 report (Reference 1). On this basis, an increase of 6.6°F in the closure head after the proposed EPU is not expected to have any impact on the PWSCC of the Alloy 690/52/152.

  The CR-3 RPV closure head inspection program is an existing program consistent with NRC NUREG-1801, GALL Report Section XI.M11A. The CR-3 program is implemented through the plant Inservice Inspection (ISI) Program by the use of augmented inspections. The NRC mandated the use of ASME Code Case N-729-1 via paragraph (g)(6)(ii)(D) in 10CFR 50.55a. The use of Code Case N-729-1 is subject to the conditions specified in paragraph (g)(6)(ii)(D)(2) through (6) in 10CFR 50.55a.

  For CR-3 RPV head, which was replaced in 2003 with PWSCC-resistant Alloy 690 nozzles and Alloy 52/152 weld materials, the applicable ISI methods and frequency is specified in Table 1 of Code Case N-729-1, Items B4.30 and B4.40 for the RPV heads with PWSCC-resistant materials of the following:

  - B4.30, bare metal visual examination (VE) must be completed for the RPV head every third refueling outage or every five years, whichever is less.
  - B4.40, volumetric and surface examination must be completed for all nozzles and partial penetration welds not to exceed one inspection interval (nominally 10 calendar years).
  - For the CR-3 RPV head, the bare metal visual examination is per Code Case N-729-1. The initial volumetric inspection and subsequent reinspection interval for the CR-3 RPV head will not be based on the use of EDY or RIY (re-inspection years), which are normalized to a reference temperature of 600°F. Hence, the increase in T_{HOT} by 6.6°F after the EPU is of no consequence to CR-3 RPV head inspection requirement.

- Thermal Aging

  Thermal aging of CASS can lead to precipitation of additional phases in the ferrite and growth of existing carbides at the ferrite/austenitic boundaries that can result in loss of ductility and fracture toughness of the material. The susceptibility to thermal aging is a function of the material chemistry, aging temperature, and time at temperature. A review of the RCS pressure boundary components shows no CASS material in the RCS pressure boundary exposed to the T_{HOT}. Therefore, there will be no impact on thermal aging by the EPU.

Results

CR-3 finds that while the small increase in temperature and modified chemistry during EPU operation have a minor effect on RCS component materials, no new failure mechanisms are introduced due to the

EPU that challenge RCPB materials. Therefore, CR-3 concludes that the above listed materials will not be adversely affected in a significant manner due to the EPU. CR-3 further concludes that the RCS materials will continue to be acceptable following implementation of the proposed power uprate. Therefore, CR-3 finds the proposed power uprate acceptable with respect to RCS materials.

2.1.5.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the susceptibility of CR-3 RCPB materials to known degradation mechanisms, and concludes that it has identified appropriate degradation management programs to address the effects of changes in the coolant chemistry and operating temperature due to the EPU on the integrity of RCPB materials. CR-3 further concludes that it has demonstrated that the CR-3 RCPB materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the CR-3 Station current licensing basis requirements with respect to FSAR Sections 1.4.1, 1.4.5, 1.4.9, 1.4.34, 1.4.35, 10 CFR 50.55a, and 10 CFR 50 Appendix G. Therefore, CR-3 finds the proposed EPU at CR-3 acceptable with respect to Reactor Coolant Pressure Boundary materials.

2.1.5.4 References

1. Materials Reliability Program: Resistance to Primary water Stress Corrosion Cracking of Alloys 690, 52 and 152 in Pressurized Water Reactors (MRP-111), EPRI, Palo Alto, CA: 2004. 100980.

2.1.6 Leak-Before-Break

2.1.6.1 Regulatory Evaluation

Leak-before-break (LBB) analyses provide a means for eliminating from the design basis the dynamic effects of postulated pipe ruptures for a piping system. NRC approval of LBB permits CR-3 to: (1) remove protective hardware along the piping system (e.g., pipe whip restraints and jet impingement barriers), and (2) redesign pipe-connected components, their supports, and their internals. The CR-3 review for LBB covered: (a) direct pipe failure mechanisms (e.g., water hammer, creep damage, erosion, corrosion, fatigue, and environmental conditions); (b) indirect pipe failure mechanisms (e.g., seismic events, system overpressurizations, fires, flooding, missiles, and failures of SSCs in close proximity to the piping); and (c) deterministic fracture mechanics and leak detection methods.

The NRC's acceptance criteria for Leak Before Break are based on:

- GDC-4, insofar as it allows for exclusion of dynamic effects of postulated pipe ruptures from the design basis.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, insofar as it allows for exclusion of dynamic effects of postulated pipe ruptures from the design basis. [GDC 4]

Additionally, FSAR Section 14.2.2.5.11 discusses an exemption that was granted to CR-3 based on "Leak-Before-Break" methodology as described in Babcock & Wilcox topical report BAW-1847, Rev. 1, "Leak-Before-Break Evaluation of Margin against Full Break for Reactor Coolant System (RCS) Primary Piping of B&W Designed NSS" (References 1, 2, and 3).

2.1.6.2 Technical Evaluation

Introduction

The original structural design basis of the RCS for CR-3 required consideration of dynamic effects resulting from postulated pipe breaks and the need to incorporate protective measures for such breaks into the design. Research by the NRC and industry, coupled with operating experience, has determined that safety could be negatively impacted by placement of pipe whip restraints on certain systems. As a result, NRC and industry initiatives resulted in demonstrating that LBB criteria can be applied to RCS piping based on fracture mechanics technology.

The current structural design basis of CR-3 includes the application of LBB methodology to eliminate consideration of the dynamic effects resulting from pipe breaks in the RCS primary loop piping. The purpose of this section is to describe the evaluations performed to demonstrate that the elimination of these breaks from the structural design basis continues to be valid following implementation of the EPU, and that the primary loop piping for which CR-3 credits LBB continues to comply with the requirements of FSAR Section 1.4.40.

Description of Analyses and Evaluations

The CR-3 review of the EPU for continued LBB applicability of the RCS piping considered direct piping failure mechanisms, indirect piping failure mechanisms and deterministic fracture mechanics and leak detection methods. No new indirect pipe failure mechanisms acting on the RCS piping are being introduced as a result of the EPU. Nor are any previously analyzed indirect failure mechanisms being modified as a result of the EPU. See Section 2.2.5, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment" for evaluation of seismic qualification of equipment and Section 2.5.1.2.1, "Internally Generated Missiles" for evaluation of missiles. Additionally, RCS piping leak detection methods are not changing for the EPU.

An evaluation of direct pipe failure mechanisms and deterministic fracture mechanics was performed as part of the LBB review for the EPU. This evaluation was based on the RCS piping forces, moments, normal operating temperature, and normal operating pressure under the EPU conditions. These parameters were used as input in the evaluation. The normal EPU operating temperature range for the RCS is provided in Table 1.1-1. Also shown in Table 1.1-1, the normal operating pressure for the RCS will not change for the EPU.

The LBB evaluation was performed by comparing the "Minimum Moment" loads and "Maximum Moment" loads calculated for the EPU to those loads documented in topical report BAW-1847, Rev. 1 (Reference 1). The "Minimum Moment" loads are normal operating loads (Dead Weight + EPU Thermal Expansion) combined algebraically (with sign). The "Maximum Moment" loads are the same as the "Minimum Moment" loads except that Safe Shutdown Earthquake (SSE) is added by absolute sum. The "Minimum Moment" loads are those loads that would exist during operation when a through-wall crack might appear. These loads are used to determine the flaw size resulting in a leakage of 10 gallons per minute (gpm). The "Maximum Moment" loads are used to determine if a crack of two times the length of the leakage crack is still stable. For this analysis, the EPU "Maximum Moment" loads for each type of piping section (hot leg elbow, hot leg straight, etc.) were compared to the "Maximum Moment" loads evaluated in BAW-1847 to ensure they do not exceed those loads. At the locations where the "Maximum Moment" loads are most severe, the "Minimum Moment" loads were calculated for the EPU conditions and compared to the "Minimum Moment" loads evaluated in BAW-1847 to ensure they exceed those loads such that the 10 gpm flow rate is detectable.

In summary, to show that the LBB evaluation provided in BAW-1847 continues to be applicable following the EPU, the following two conditions must be satisfied:

- The “Minimum Moment” loads calculated for the EPU must be *greater* than the “Minimum Moment” loads documented in BAW-1847
- The “Maximum Moment” loads calculated for the EPU must be *less* than the “Maximum Moment” loads documented in BAW-1847

Reference 1 includes a qualitative assessment of thermal aging of the cast austenitic stainless steel (CASS) reactor coolant pump (RCP) inlet and exit nozzles. However, reduction of fracture toughness by thermal aging of the RCP inlet (suction) and exit (discharge) nozzles was considered quantitatively in subsequent evaluations for a generic load set to ensure that the conclusions of the LBB evaluation reported in Reference 1 remain valid for the CASS RCP nozzles. To confirm that the LBB analysis of the CASS RCP nozzles remains valid for the EPU, a load comparison has been done to show that the loads calculated for the EPU conditions are bounded by the generic loads.

Results

The piping credited for LBB has been reviewed for the EPU impact on (a) direct pipe failure mechanisms, (b) indirect pipe failure mechanisms and (c) deterministic fracture mechanics and leak detection methods. Indirect failure mechanisms and leak detection methods were determined to be unaffected by the EPU. To assess the EPU impact of direct pipe failure mechanisms and deterministic fracture mechanics an evaluation was performed which shows the EPU loads are bounded by the loads previously considered in LBB evaluations.

As done in topical report BAW-1847, the worst locations for “Maximum Moment” loads were determined first. Table 2.1.6-1 shows a comparison for each type of RCS piping section, the EPU “Maximum Moment” loads, and the BAW-1847 “Maximum Moment" loads. Table 2.1.6-1, as noted, shows the EPU “Maximum Moment” loads are acceptable when compared to the loads evaluated in topical report BAW-1847.

Next, the EPU “Minimum Moment” loads were calculated at the locations where the “Maximum Moment” loads are most severe and then checked to ensure that those loads are higher than those in topical report BAW-1847. Table 2.1.6-2 shows a comparison of the EPU “Minimum Moment” loads and “Minimum Moment” loads from BAW-1847. Table 2.1.6-2, as noted, shows that the BAW-1847 “Minimum Moment” loads remain applicable and are acceptable for LBB for the EPU conditions.

Tables 2.1.6-3 and 2.1.6.4 present a comparison of the pre-EPU and EPU “Minimum Moment” and “Maximum Moment” loads of the RCP suction and discharge nozzles. As shown in these tables, the EPU loads remain bounded by the pre-EPU loads for the RCP nozzles. That is, the EPU Minimum Moments are *greater* than the pre-EPU Minimum Moments, and the EPU Maximum Moments are *less* than the pre-EPU Maximum Moments. Thus, the pre-EPU LBB analysis of the RCP nozzles remains valid for EPU conditions.

2.1.6.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the LBB analyses and concludes that changes in primary system pressure and temperature and their effects on the LBB analyses have been adequately addressed. CR-3 further concludes that the LBB analyses will continue to be valid following implementation of the proposed EPU and that lines for which the CR-3 credits LBB will continue to meet the requirements of FSAR Section 1.4.40. Therefore, CR-3 finds the proposed EPU acceptable with respect to Leak Before Break.

2.1.6.4 References

1. BAW-1847, Rev 1, “B&W Owners Group Leak-Before-Break Evaluation of margins Against Full Break for RCS Primary Piping,” dated September 1985.

2. Safety Evaluation Report: Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, February 18, 1986.

3. License Amendment No. 89: NRC to CR-3 letter, dated May 23, 1986, “Crystal River Unit 3 - Amendment to Facility Operating License”.

Table 2.1.6-1: Comparison of Maximum Moment Loads

Piping Component	BAW-1847 Maximum Moment Loads	CR-3 EPU Maximum Moment Loads
	$M_{bending}$ (ft-kips) [(1)]	$M_{bending}$ (ft-kips)
36" ID Hot Leg Straight	3952.3	1610.9
36" ID Hot Leg Elbow	2376.5	1614.2
28" ID Cold Leg Straight	3098.0	2008.2
28" ID Cold Leg Elbow	2822.8	1553.9

NOTES:
(1) Evaluated Maximum Moment Loads come from Table 4-6 of BAW-1847

Table 2.1.6-2: Comparison of Minimum Moment Loads

Piping Component	BAW-1847 Minimum Moment Loads	CR-3 EPU Minimum Moment Loads
	$M_{bending}$ (ft-kips) [(1)]	$M_{bending}$ (ft-kips)
36" ID Hot Leg Straight	2284.0	1518.7[(2)]
36" ID Hot Leg Elbow	1010.0	1526.0
28" ID Cold Leg Straight	560.0	1789.8
28" ID Cold Leg Elbow	1278.0	1403.3

NOTES:
(1) Evaluated Minimum Moments come from Table 4-1 of BAW-1847.
(2) The EPU minimum moment is less than the BAW-1847 moment for the 36” ID Hot Leg Straight; however, as seen in Table 2.1.6-1, the EPU maximum bending moment at this location is much smaller than the BAW-1847 maximum moment, meaning that the flaw size required for a flow rate of 10 gpm is stable enough to not experience growth under the EPU maximum moment loading.

Table 2.1.6-3: Comparison of RCP Nozzle Minimum Moment Loads

RCP Nozzle	Pre-EPU Minimum Moment	EPU Minimum Moment
	$M_{bending}$ (ft-kips)	$M_{bending}$ (ft-kips)
Suction	970.9	1608.0
Discharge	881.2	1321.2

NOTES:
(1) The above loads for the suction and discharge nozzle are the result of enveloping all four RCPs.

Table 2.1.6-4: Comparison of RCP Nozzle Maximum Moment Loads

RCP Nozzle	Pre-EPU Maximum Moment	EPU Maximum Moment
	$M_{bending}$ (ft-kips)	$M_{bending}$ (ft-kips)
Suction	2482.3	2181.1
Discharge	2568.2	2220.1

NOTES:
(1) The above loads for the suction and discharge nozzle are the result of enveloping all four RCPs.

2.1.7 Protective Coating Systems (Paints) – Organic Materials

2.1.7.1 Regulatory Evaluation

Protective Coating Systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination by radionuclides and also provide wear protection during plant operation and maintenance activities. The CR-3 review covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects.

The NRC's acceptance criteria for Protective Coating Systems are based on:

- 10 CFR 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related structures, systems, and components SSCs; and
- Regulatory Guide (RG) 1.54, Revision 0, for guidance on the application of coatings in nuclear power plants.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Sections 1.6, Quality Program (Preoperational), and 1.7, Quality Program (Operational), provides quality assurance requirements for the design, fabrication, and construction of SSCs. [10 CFR Part 50, Appendix B]
- FSAR Table 1-3, Crystal River Unit 3 Quality Program Commitments, states and clarifies that the application of coating systems inside the CR-3 containment is in accordance with RG 1.54 (June 1973 version) and ANSI N101.4-1972. [RG1.54]

Additionally, FSAR Sections 1.3.2.11, 5.2.2.5 and FSAR Table 14-45 provide criteria for Protective Coatings insofar as they sets forth guidance on the application of coatings in nuclear power plants in conformance with RG 1.54.

CR-3 provided additional information pertaining to their reactor containment protective coatings program in 1998 via FPC Letter 3F1198-02 (Reference 2) in their response to GL 98-04. The use of RG 1.54 (June 1973 version) and ANSI N101.4-1972 were noted within the response to the NRC as part of the summary description of the plant-specific program. In addition, it was noted that design basis accident (DBA) qualification testing was performed in 1990 using the requirements of ASTM D3911-80 as a guideline. The CR-3 response to GL 98-04 was evaluated and accepted by the NRC (Reference 3).

In accordance with Progress Energy CR-3 Letter, dated August 30, 2005, Response to Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors, CR-3 maintains its protective coatings program consistent with its response to GL 98-04.

2.1.7.2 Technical Evaluation

Introduction

The Protective Coating System's function is to provide corrosion and erosion control and to facilitate decontamination in the event of radioactive material leakage into the containment. Failure of coating systems inside the containment before and/or during a design bases accident that requires the recirculation of water from the containment sump could degrade or fail the accident response systems' functions by clogging the sump screens.

Description of Analyses and Evaluations

The approach taken to evaluate the impact of the proposed EPU on the CR-3 Primary Reactor Containment Protective Coatings Program was to perform a review of relevant requirements documents, to review relevant CR-3 EPU containment evaluations and/or calculations (refer to Section 2.6.1, Primary Containment Functional Design), and to then assess the impact that the EPU will have on the applicable requirements and existing DBA qualification of the primary containment protective coatings in accordance with FSAR Table 1-3.

Two test reports form the basis for the DBA qualification of coatings used inside containment. The test report performed in 1990 (as noted in CR-3's response to GL 98-04) was for all of the coating products ever approved for application in the CR-3 Reactor Building. Since that time a new DBA test report has been performed in order to qualify the use of the Carboline Carboguard 2011SN surfacer topcoated with Carboline Carboguard 890N for concrete substrates. That particular test report utilized the ASTM D3911 Figure 2 pressure-temperature curve except that a peak temperature of 310°F was used in lieu of 307°F. The ASTM D3911 Figure 2 pressure-temperature curve is greater than the pressure-temperature curve used in the 1990 DBA qualification test report. The accumulated radiation dose that was used in the 1990 DBA test report was equal to 1.80 E+08 rads. The Carboguard 2011SN DBA test report utilized an accumulated dose equal to or greater than 3.00 E+08 rads. Based on the higher pressure, temperature, and accumulated dose used for the Carboguard 2011SN DBA test, this test is considered to be the most limiting DBA test. It is referred to as the most limiting DBA test report in the remainder of the write-up for Section 2.1.7, Protective Coating Systems (Paints) - Organic Materials.

The temperature profiles for the most limiting DBA test and two most limiting Design Basis Loss-of-Coolant Accident (DBLOCA) containment responses at EPU conditions are shown in Figure 2.1.7-1 (limiting case obtained from Section 2.6.1, Primary Containment Functional Design). The pressure profiles for the most limiting DBA test and DBLOCA containment response at EPU conditions are shown in Figure 2.1.7-2 (limiting case obtained from Section 2.6.1, Primary Containment Functional Design).

The containment pressure and temperature analyses for the EPU remain below the design pressure and temperature limits (refer to Section 2.6.1, Primary Containment Functional Design).

From a review of Figures 2.1.7-1 and 2.1.7-2, it can be seen that the DBA test exceeds the calculated peak pressure and temperature for the EPU. The DBA test does not envelope the predicted EPU

DBLOCA conditions in all cases but the DBA test overall is considered to be more severe. Specifically, the DBA test does not envelope the CR-3 temperature at 0.1 seconds to about 20 seconds. Additionally, the DBA test does not envelope the CR-3 pressure at 0.1 seconds to about 0.2 seconds. These differences are not considered significant. Lastly, the DBA test has a steep drop in pressure and temperature at 4,000 seconds as the CR-3 pressure-temperature curves decreases at a more gradual rate. Testing experience shows that at the point where the quick drop in pressure and temperature occurs is typically where coating failures occur. This is due to the thermal shock which challenges the coating system. This quick loss of pressure against the coating film allows the coating system to relax to some degree and if the adhesion to the substrate is not adequate, the coating system will delaminate or fall off of the surface. The more gradual decrease in pressure and temperature predicted for CR-3 under EPU conditions will have less of a thermal shock and delaminating effect. Therefore, the DBA test is considered to be a more severe test than that predicted to be experienced during a DBLOCA even though portions of the CR-3 pressure and temperature curve are not enveloped. Consequently, it provides an adequate basis for the qualification of all of the Service Level 1 coatings within the CR-3 containment for the EPU.

The total 40 year accumulated gamma and beta radiation dose (with a LOCA) for the EPU is 1.64 E+08 rads. The most limiting DBA test report shows that the coated test samples were exposed to an accumulated radiation dose of 1.80 E+08 rads which bounds the accumulated radiation dose for the EPU condition.

The Reactor Coolant Quality Specifications (FSAR Table 4-10) specifies a pH range of 6.4 to 7.8 which varies based on the lithium and boron concentration. There will be no change in this range under EPU conditions. As a result of the pH remaining unchanged at EPU conditions, the post-accident pH of containment spray at EPU conditions will remain the same as it would be under current operating conditions. Therefore, the EPU has no impact on the coatings within the containment from exposure to chemical spray.

CR-3 conducts visual inspections of coatings during each refueling outage as stated in the response to GL 2004-02. Areas with degraded coatings are prioritized, removed, and repaired as appropriate. Maintenance or repair coatings are applied using approved procedures that meet the licensing basis requirements, thereby ensuring that the DBA performance requirements are maintained. There is no effect of the EPU on these activities.

Results

The calculated peak containment pressure and temperature from a DBLOCA at EPU conditions is less than those used for the most limiting DBA test for qualification of the CR-3 containment coatings. The DBA test past peak response is considered to be a more severe test than that predicted to be experienced during a DBLOCA. The test samples for the most limiting DBA test were irradiated at a total accumulated radiation dose that was higher than that which will occur for EPU conditions. The post-accident pH of containment spray at EPU conditions is unchanged and has no impact on the coatings within the containment from exposure to chemical spray.

With regard to Protective Coating System failures, as it relates to GL 2004-02, the design basis conditions (temperature, pressure, radiation dose, and pH) for protective coatings inside containment during the

EPU conditions will not create any additional coating system failures than that postulated for the resolution of GL 2004-02. Therefore, the EPU will not impact the resolution to GL 2004-02.

2.1.7.3 Conclusion

CR-3 has evaluated the effects of the proposed EPU on Protective Coating Systems and appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. CR-3 concludes that the evaluation demonstrates that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of FSAR Sections 1.6 and 1.7. Therefore, CR-3 finds the proposed EPU acceptable with respect to Protective Coatings Systems.

2.1.7.4 References

1. Florida Power Corporation (FPC) Letter Number 3F0692-18, dated June 26, 1992, Change to Quality Program Description – Reactor Building Painting.

2. Florida Power Corporation (FPC) Letter Number 3F1198-02, dated November 6, 1998, Response to Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment."

3. USNRC Letter dated November 9, 1999, Completion of Licensing Action for Generic Letter 98-04, Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment.

Figure 2.1.7-1, Temperature Profiles for DBA Test and DBLOCA Containment Response

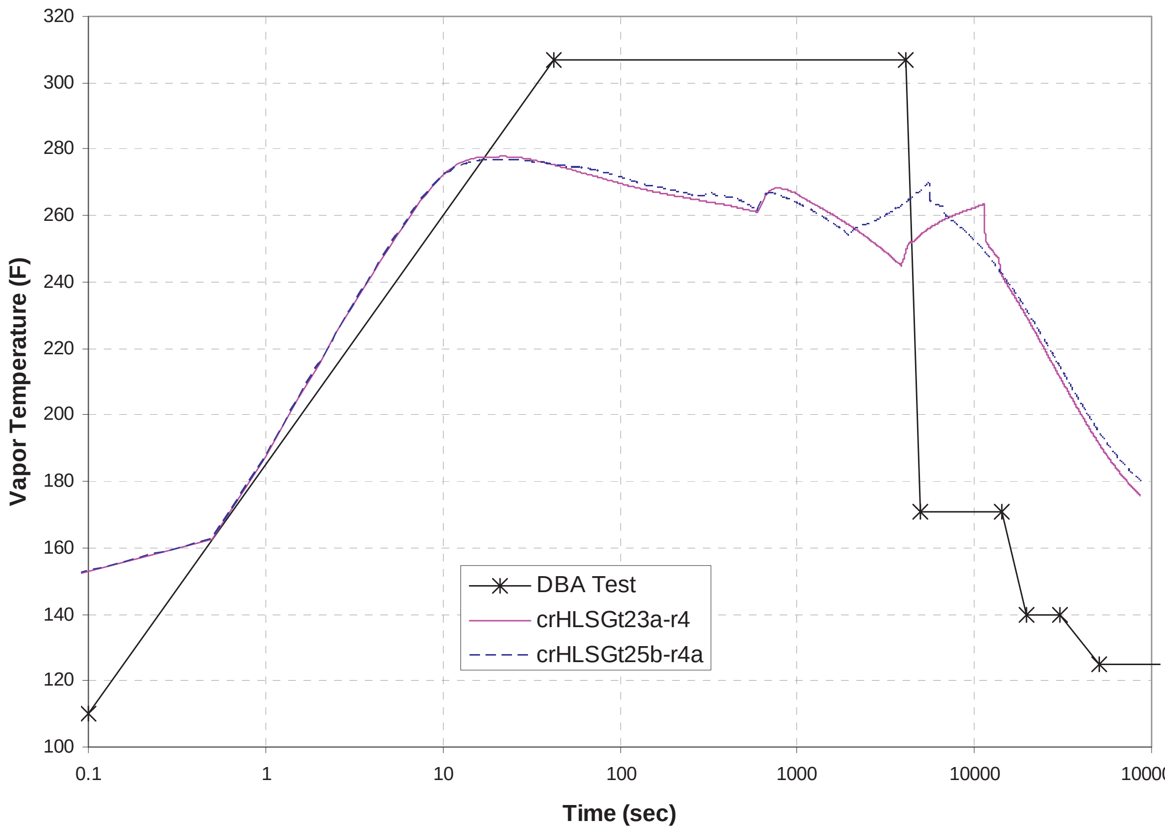
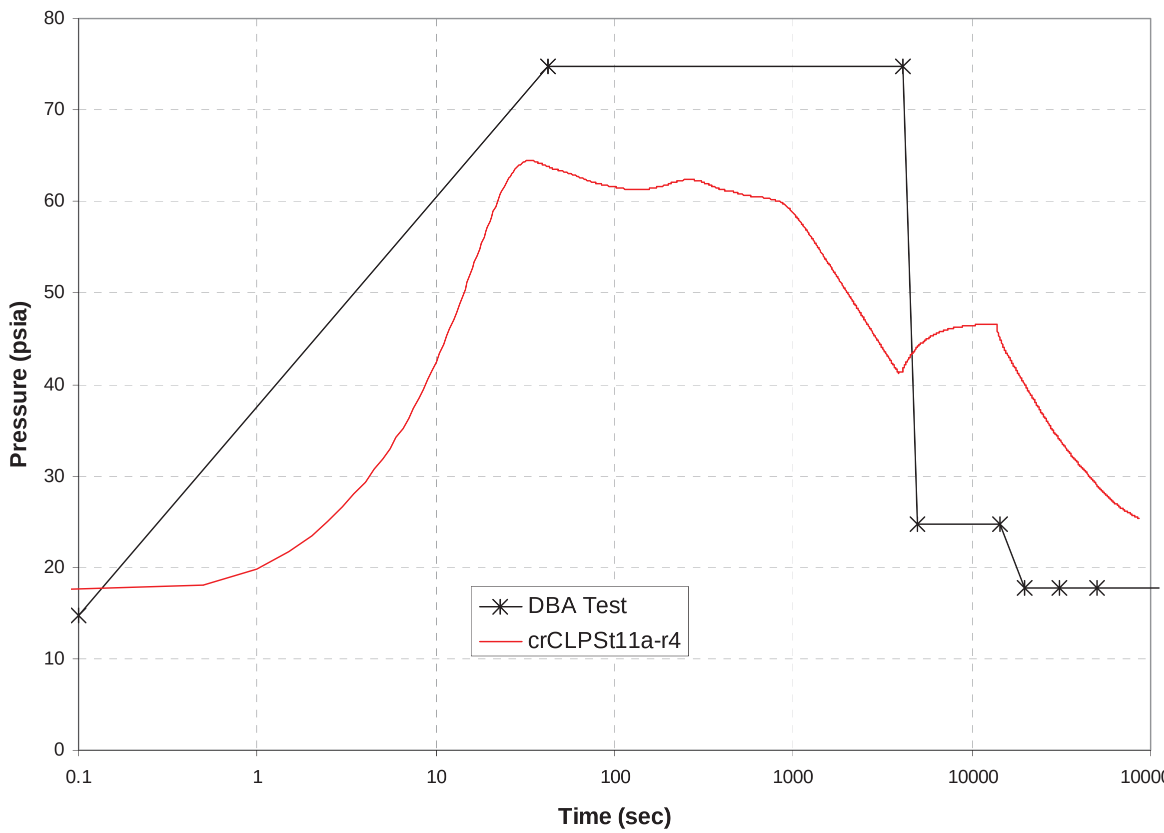


Figure 2.1.7-2, Pressure Profiles for DBA Test and DBLOCA Containment Response



2.1.8 Flow-Accelerated Corrosion

2.1.8.1 Regulatory Evaluation

Flow-Accelerated Corrosion (FAC) is a corrosion mechanism occurring in carbon steel and low alloy steel components exposed to both single and two phase flow conditions. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects in most cases cannot be achieved. Loss of material by FAC will, therefore, occur. CR-3 has reviewed the effects of the proposed EPU on FAC and the adequacy of the FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. CR-3's FAC program is based on NUREG-1334, Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning", and the guidelines in EPRI Report NSAC-202L-R3, "Recommendations for an Effective Flow-Accelerated Corrosion Program." It consists of predicting loss of material using the CHECWORKS computer code, and visual inspection and volumetric examination of susceptible components.

NRC's acceptance criteria for Flow-Accelerated Corrosion are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

<u>CR-3 Current Licensing Basis</u>

The CR-3 FAC Program is responsive to NUREG-1334, Generic Letter 89-08 and implements the guidelines in EPRI Report, NSAC-202L-R3.

2.1.8.2 Technical Evaluation

<u>Introduction</u>

The program is designed to ensure that FAC does not result in unacceptable degradation of the structural integrity of susceptible piping systems. The program is documented in the CR-3 FAC Monitoring Program and includes the following:

- Identifying susceptible systems
- CHECWORKS analysis
- Inspection/expansion selection criteria
- Acceptance criteria
- Repair/replacement criteria
- Corrective action
- Long term strategy

This section addresses the following FAC Program topics:

- FAC Program scope / Attributes
- Piping / component inspection
  - CHECWORKS
  - Susceptible Non-Modeled Program
  - Trending
  - Plant Experience
  - Operating Experience
  - FAC Susceptible Equipment / Pressure Vessel FAC Program
  - Engineering Judgment
  - Inspection Techniques
- Evaluation of inspection data
- Sample expansion
- Component repair / replacement

FAC Program Scope / Attributes

The scope of the FAC Program encompasses piping, pressure vessels, and storage tanks containing both single phase and two phase fluids. The following systems and equipment are screened for inclusion into the CR-3 FAC Program:

- Carbon and low alloy steel piping systems susceptible to FAC
- Vessels and tanks connected to piping systems within the scope of the FAC Program
- Portions of systems and components that have been found to exhibit wear from mechanisms other than FAC (e.g. erosive attack).

All piping and equipment for the systems identified above are considered susceptible to FAC unless excluded by one of the following screening criteria:

- **FAC Resistant Material**

  Stainless steel or low-alloy steel with chromium content equal or greater than 1.25% are FAC-resistant. Lines or systems made, or replaced with, such materials are not sufficiently susceptible to FAC to warrant further analysis. It is to be noted, however, that resistance to FAC does not ensure against other erosion/corrosion mechanisms such as cavitation or impingement. Therefore, even if components are replaced with FAC-resistant material, the root cause of wear is determined before excluding the replaced components from the inspection program.

  Wholesale replacement of a pipe segment with FAC resistant material would cause that piping to be excluded from the FAC program. However, if any one component on a line is susceptible, then the whole line was considered susceptible.

- **Superheated Steam**

  Piping transporting superheated steam with no moisture content was classified as non-susceptible. According to EPRI Report NSAC-202L-R3, FAC is known to occur only under flowing water or wet steam conditions, and has not been documented in superheated steam piping.

- **Single-Phase Piping at Low Temperature**

  Piping with single-phase flow and an operating temperature below 200°F does not experience FAC and was classified as non-susceptible. No temperature exclusion exists for two-phase lines. Other degradation mechanisms, such as cavitation, may occur at low temperatures but such mechanisms are outside the scope of this analysis.

- **Low Operating Frequency**

  Piping that operates less than 2% of the plant operating time was deemed non-susceptible. In general, such piping does not experience the amount of flow required to make FAC a legitimate concern and is excluded from further analysis in favor of piping with greater FAC susceptibility. Piping in this category includes lines with normally closed valves and those feeding or emerging from equipment that operate less than 2% of the plant operating time.

  Exceptions to this rule are made when the operating frequency may be low, but the service is especially severe.

  All lines excluded due to infrequent operation would be susceptible to FAC if operating frequency is increased above the 2% threshold. Therefore, if operating frequency is increased, these lines are reviewed for inclusion into the FAC Program.

- **Combination Single-Phase Piping with Low Temperature and Operating Frequency**

  Piping containing single-phase flow that operates less than 2% of the time above 200°F was classified as non-susceptible. This category is a derivative of the two previous susceptibility categories. This category may be used when normal flow is below the 200°F threshold but occasionally (<2% of the time) exceeds it.

- **Non-Water/Steam Piping**

  Piping that transports fluids other than water or steam, such as air or oil, are not susceptible to FAC.

- **Dissolved Oxygen Concentration**

  Lines containing water with high levels of dissolved oxygen (typically greater than 1000 ppb) are considered immune to FAC and can be excluded from further analysis. Systems normally meeting this criterion include Service Water, Circulating Water, and Fire Protection.

- **Low or No Flow**

  While instrumentation and/or sensing lines may operate continuously, they experience very limited fluid flow. Such piping does not experience the amount of flow required to make FAC a legitimate concern and is excluded from further analysis in favor of piping with greater FAC susceptibility.

- **Non-Piping**

  Systems that do not contain piping (e.g. computer system) were excluded from the FAC program.

- **Piping Removed from Service**

  Piping that has been removed from service and capped was excluded from the FAC program. Degradation mechanisms may occur in such piping, but for FAC to occur, flow is required.

The following large bore systems have been found to be susceptible to FAC through the screening process and are monitored in the FAC program:

- Feedwater
- Auxiliary Steam
- Condensate
- Heater Drains/Vents
- Extraction Steam
- Gland Steam

- Gland Seal Water
- Main Steam
- Reheat Steam
- Turbine Drain

The following small bore systems / lines are monitored in the small bore FAC program:

- Auxiliary Steam
- Extraction Steam
- Feedwater
- Gland Seal Water
- Main Steam Drains
- Heater Vents
- Gland Steam
- Reheat Steam
- Turbine Drains

The pressure vessel FAC program includes the following tanks:

- Feedwater Heater
- Deaerator
- Deaerator Storage Tank
- Moisture Separator Reheater
- Reheater Drain Tank
- Miscellaneous Drain Tank
- Steam Generator
- Condenser
- Gland Steam Condenser

Piping / Component Inspection

A component in a susceptible piping system is selected for inspection if any one of the following considerations is applicable:

- Results of lines analyzed using the Predictive Plant Model.
- Results of evaluations of lines that cannot be accurately analyzed in the Predictive Plant Model due to uncertain operating conditions. They are commonly called "Susceptible-Non-Modeled" lines. Lines with socket-welded fittings are typically not analyzed due to uncertainties in the fit-up gaps.
- Extrapolations of prior inspection results, commonly called "trending".
- Plant experience.
- Operating experience.
- FAC-susceptible equipment.
- Engineering judgment.

Results of Lines Analyzed using the Predictive Plant Model (CHECWORKS)

The EPRI CHECWORKS computer code SFA 2.2 is used to evaluate piping systems with known operating conditions that are susceptible to FAC. Parameters including operating conditions, pressure, temperature, water chemistry, and piping/component design conditions are inputs for CHECWORKS program evaluations.

The primary objective of the CHECWORKS Program is to provide predictive technology to estimate wear rates and remaining service lives of FAC susceptible components. A sampling of these components can then be periodically inspected to refine and calibrate the predictive model and to identify components with unacceptable wear. The CHECWORKS model is continually updated based on periodic non-destructive examination (NDE) inspections.

Selection of inspection locations in CHECWORKS Calibrated and Non-Calibrated lines are components with the highest predicted wear rates and the shortest remaining service life. Selection of inspection locations in New Lines are as follows:

- Select a sample from the components identified in the wear ranking as having the highest relative wear.
- Select one or more components with the shortest relative remaining service life from the time rankings.

If a component is considered susceptible to FAC but cannot be inspected, it is analytically evaluated using the CHECWORKS Pass 2 results. The analytical predictions are then compared to actual wear rate results for actually inspected, usually adjacent, components which have the same fluid conditions. These results are used to trend the un-inspected component and if possible, a visual inspection to confirm them.

Susceptible Non-Modeled

CR-3 has established a Susceptible Non-Modeled (SNM) Program to produce a prioritized ranking of all of the CR-3 FAC SNM piping. This program determines the consequence of failure, the level of FAC susceptibility, and establishes the methodology used to perform the evaluation. The results of this program are to be used as input to inspection planning and selection of inspection locations and/or as a guide for a proactive small-bore replacement plan.

SNM lines include small-bore and some large-bore piping that is not suitable for predictive modeling. Small-bore piping is not suitable for CHECWORKS modeling primarily due to the random nature of the fit-up gap between the pipe and socket-weld fittings. In addition, the extremely large linear footage of small-bore lines and the often-unknown thermodynamic conditions combine to make detailed modeling of small-bore piping inefficient and sometimes inaccurate. In addition to small-bore piping, some large-bore FAC susceptible piping is unsuitable for CHECWORKS modeling because of uncertain operating conditions.

Since quantitative methods are not available for evaluating SNM piping, a qualitative method is used. This method addresses both the failure consequence and relative FAC susceptibility to develop a prioritized ranking of lines that can be used as input to the inspection planning process.

Trending

Trending is the evaluation of past inspection data to identify components with the highest trended wear rate and shortest remaining service life which is determined by extrapolations of prior inspection results. Trended results are adjusted for changes to power level and/or chemistry where such changes will increase wear rates.

Plant Experience

CR-3 maintains record of piping components that have experienced FAC degradation which required past repair or replacement. These locations are considered in the FAC Program. Continued inspection at these points depends on the replacement material used.

Operating Experience

Components which have displayed susceptibility for FAC in other power plants are given consideration in the inspection point selection process. These “industry experience” points are typically downstream of components which cause flow restriction or otherwise add turbulence (e.g. downstream of control valves and orifices). A review of the inspection results is performed to assure plant-specific experience has been included in the selection of components for examination.

FAC-susceptible equipment / Pressure Vessel FAC Program

Because of the potential severe consequences of a vessel failure (i.e., plant shutdown and personnel injury), CR-3 has developed a program to address FAC for vessels. The primary purpose of this Pressure Vessel Program is to make recommendations as to the inspection requirements of vessels based on the new EPU operating conditions. The Pressure Vessel Program also:

- Provides background data on FAC in vessels including damage mechanisms, operating experience, EPRI guidance, and utility practices.
- Provides a methodology to determine FAC susceptibility of vessels.
- Evaluates the relative susceptibility of vessels to FAC.
- Recommends inspection locations and inspection techniques.

Engineering Judgment

Piping systems that are not analyzed using CHECWORKS have inspections at locations based upon engineering experience. A conservative number of locations is specified to ensure that any significant variations in susceptibility are represented and bounded. CHECWORKS and non-CHECWORKS selections are based on the following:

- Steam cycle systems with similar flow and operating conditions may be grouped together. Each group is considered separately when determining required inspections.
- Historical inspection data available for each system is considered.
- A qualified engineer experienced in the use of CHECWORKS reviews isometric drawings or performs walkdowns to determine inspection locations.
- In addition to the inspection of “standard” components, special consideration is given to components known to be particularly susceptible (e.g. control valves, component discharge nozzles, orifices, and steam traps).
- Databases are used to trend inspection results and evaluate wear rates for non-CHECWORKS systems.

Inspection Techniques

Ultrasonic testing (UT) provides the most quantitative wear trend data of a component to assess wall thickness and predict future wear. Therefore, UT is the primary non-destructive examination method used for detecting pipe wall thinning. For locations where ultrasonic testing cannot provide adequate results, alternate methods may be used, including radiographic testing, visual, and video examinations.

Evaluation of Inspection Data

Evaluations of UT data to establish the component’s initial thickness and determine the measured wear is very complex. Complications may include: not knowing the initial thickness of component, counter-bore, backing ring, bad measured reading, obstructions such as pipe hangers, lugs, etc. There are four methods that can be used to calculate wear of inspected components in the FAC program by using single and multiple measured inspection data.

- Point to point – where multiple thickness measurements are available
- Band Method

- Area Method
- Moving Blanket Method

These four methods are defined in the EPRI Report, NSAC-202L-R3.

The process / criteria for determining remaining service life of a component is outlined as follows:

- The minimum wall thickness (T_{MIN}), determined by the piping code, is based on the hoop stress due to internal design pressure and/or longitudinal stress due to internal design pressure plus bending moment due to dead weight.
- The predicted wall thickness (T_{PRED}) is defined as the predicted wall thickness at the next refueling outage. The following factors are considered in the calculation of remaining service life:
    - Minimum wall thickness based on code requirements
    - Current wear rate
    - Effective full power years (EFPY) until the next refueling outage
    - Line correction factor for the CHECWORKS analysis
    - Safety factor
- The wall thickness measured (T_{MEAS}) is obtained by nondestructive examination methods when the component is inspected.
- If the predicted wall thickness is greater than the code required wall thickness, the component is acceptable for continued service.
- If T_{PRED} is less than T_{MIN} or is expected to go below T_{MIN} prior to the next refueling outage, a Condition Report is generated and actions are taken to repair / replace the component.

Sample Expansion

Sample expansion is the process by which additional components are selected for inspection due to the detection of FAC-related wall thinning exceeding specified limits. The purpose of this effort is to determine the extent of the located thinning.

- If the inspections in the initial sample selection reveal unexpected wall thinning, the sample size is increased and additional inspections performed to bound the wear. The increased sample includes the following:
    - Components within two diameters downstream of a component displaying significant wear, or two diameters upstream if that component was an expander or expanding elbow.
    - A minimum of the next two most susceptible components from the CHECWORKS relative wear ranking in the same train as the piping component displaying significant wear.

- Components in other trains of a multi-train line with similar configuration as the piping component displaying significant wear.

- The expansion process will continue until no further expansion is required based on the above criteria. Expansions within a parallel line to the next most susceptible components in that parallel line is not necessary if the initial parallel train examinations are acceptable and do not require expansions.

Components which have been inspected are re-examined on a schedule consistent with the calculated component life (time to T_{MIN}). Re-examination is scheduled for the refueling outage preceding the time when T_{MIN} will be reached.

Component Repair / Replacement

The existing criteria for repair/replacement of piping are consistent with the guidelines in EPRI Report NSAC-202L-R3. The following items are considered in making component replacement decisions:

- The cost and availability of replacement fittings.
- The need for skills and procedures to weld alloy steels and clad material to carbon steel.
- The pre- and post-weld heat treatments generally required for welding “chrome-moly” fittings.
- The piping stress analysis required if a large portion of a carbon steel line is replaced with stainless steel.

The feasibility of replacing the entire system with a more wear-resistant material.

- Limits on hexavalent chromium when cutting, grinding, and welding chromium-based materials

The following items are considered in making component repair decisions:

- If repair is decided upon, the weld buildup technique is commonly used for the temporary repair of balance-of-plant piping. Interior weld buildup is generally preferred to exterior buildup; however interior weld buildup is often limited by accessibility. Temporary clamping devices are often used to make temporary repairs to balance-of-plant piping.

- If repair or replacement of a component is necessary, a strategy (e.g., replacement with a more resistant alloy) is developed so that the wear process does not continue. The strategy will focus on reducing the plant FAC susceptibility. Optimizing the inspection planning process is important, but reduction of FAC wear rates is needed if both the number of inspections and the probability of failure are to be reduced.

In order to achieve the long-term goals of reduced cost and increased safety, a strategy of a systematic reduction of FAC wear rates has been adopted. Three options are available to reduce FAC wear rates. These are:

- Improvements in materials.

- Improvements in water chemistry.
- Local design changes.

Description of Analyses and Evaluations

For piping in the following systems, the EPU will result in changes in operating pressures, temperatures, and fluid flow velocities and affect steam quality in the applicable lines:

- Extraction steam
- Heater drains, including moisture separator reheater drains
- Condensate
- Feedwater

Prior to implementing the EPU all of the CR-3 CHECWORKS models will be updated to incorporate operating pressures and temperatures, fluid velocities, and steam quality data derived from the EPU heat balances.

As indicated previously in this section, single phase systems with operating temperatures less than or equal to 200°F are excluded from the FAC program. A review of lines in the systems listed above was performed to determine if, at the EPU conditions, the operating temperature of any single phase lines / components not currently in the FAC program increased from below to above 200°F, the temperature threshold for FAC susceptibility. No current lines had temperature increasing to above the FAC susceptible 200°F. The following extraction steam lines were added to the FAC Program because the lines are no longer superheated: Extraction from high pressure (HP) Turbine to FWHE-3A (6A) and FWHE-3B (6B); Extraction from HP Turbine to Reheater 3A/B and 3C/D; Extraction from HP Turbine to moisture separator reheaters (MSR) 3A/B/C/D.

For several representative lines, which are susceptible to FAC, Table 2.1.8-1 provides a comparison of the wear rates for both the existing full power plant conditions and the EPU conditions as analytically calculated by the EPRI CHECWORKS Program. In support of the planned plant operation at the EPU power level, the actual wear rates, which would be measured in the plant, can vary from wear rates that are calculated using CHECKWORKS. However, the percent change in wear rates (generated by the CHECKWORKS model) from plant operation prior to the EPU (Cycles 1 through 17) to permanent plant operation post-EPU (Cycle 18) provides an estimate for expected changes in actual component wear rates.

The majority of the line/component wear-rates noted in Table 2.1.8-1 increased. However, several lines had wear rates that decreased or remain unchanged after the EPU. A summary of results of lines with the EPU wear rates which significantly changed is as follows:

- Heater Drains: HP Reheaters to FW Heaters 6
- Heater Drains: low pressure (LP) Reheaters to FW Heaters 5

- Extraction Steam: LP Turbines to FW Heaters 3
- Extraction Steam: LP Turbines to FW Heaters 2
- Extraction Steam: LP Turbines to FW Heaters 1

The EPU will result in an increase in steam flow and velocities in the HP Turbine gland steam leak-off piping due to the higher HP turbine exhaust pressure. Since the gland steam piping is periodically monitored as part of the existing FAC Program, any increase in long term wear of the piping due to the EPU will be identified by the existing monitoring program. The EPU may result in increased flow rates in some small bore piping (e.g., main steam drains). As discussed previously in this section, monitoring and inspection of small bore lines is based on engineering experience and judgment. Monitoring/inspection of small bore lines in accordance with the FAC Program will continue after the EPU.

Feedwater heaters shells are included in the FAC Program. Feedwater Heaters CDHE-3A and 3B have been replaced in 16R in preparation for the EPU. Monitoring per the FAC Program will be continued after the EPU.

Drain inlet nozzles and extraction steam inlet nozzles of all feedwater heaters are included in the FAC Program. Monitoring per the FAC program will be continued after the EPU.

Deaerator and Deaerator Storage Tank are included in the FAC Program. Monitoring per the FAC Program will be continued after the EPU.

An assessment of the potential for FAC within the MSRs at the EPU conditions has been performed. The MSRs have been replaced in R16 in preparation for the EPU and future monitoring will be continued by the FAC Program after the EPU.

Monitoring of the reheater drain tank and reheater flash tanks per the FAC Program will be continued after the EPU. Any changes in wear due to the EPU would be identified by the periodic inspections.

Elements of the FAC Program, including the component repair/replacement process and criteria, will continue to be utilized following the EPU.

For plant modifications associated with the EPU that will be completed in R17, impact of the modifications on the FAC Program will be addressed as part of the plant change process. For these new components and affected pre-existing components, their satisfaction of the FAC Program's inclusion/exclusion criteria following the EPU will be checked, and they will be subject to the program depending on those findings.

<u>Results</u>

Changes to the piping/equipment included in the FAC Program as a result of the EPU are within the scope of the existing program and in compliance with program criteria. As noted above, the EPU changes to the FAC Program have already been incorporated into the CR-3 FAC Program and this program will continue to monitor components to ensure their adequacy.

2.1.8.3 Conclusion

CR-3 has evaluated the effect of the proposed EPU on the FAC analysis for the plant and concludes that the review has adequately addressed changes in the plant operating conditions on the FAC analysis. CR-3 further concludes that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU is acceptable with respect to Flow Accelerated Corrosion.

2.1.8.4 References

None

Table 2.1.8-1 Comparison of Current and EPU Analytical Wear Rates for CR-3 FAC

Location	No. of Comps Analyzed	Wear Rate Change			Avg. Post Wear Rate (mils/yr)	Temperature		Steam Quality		Flow Rate		Notes
Steam Cycle Location		Avg. Change (%)	Max Change (%)	Min Change (%)		Post Temp (°F)	Temp Change (°F)	Post Quality (%)	Quality Change (%)	Post Flow Rate (Mlb/hr)	Change in Flow Rate (%)	
Condensate: FW Heaters 2 to FW Heaters 3	26	7.9	9.1	7.5	1.5	205.8	-2.3	0.0	0.0	8.79	10.6	
Condensate: FW Heaters 3 to Deaerator	88	11.5	12.4	0.0	1.4	272.4	4.5	0.0	0.0	7.79	-2.0	
Feedwater: Deaerator to FW Booster Pumps	36	-15.9	-10.9	-16.4	1.0	318.5	15.3	0.0	0.0	12.86	19.3	
Feedwater: FW Booster Pumps to FW Heaters 5	42	-11.8	-6.6	-12.5	1.2	313.4	9.9	0.0	0.0	12.86	19.3	
Feedwater: FW Heaters 5 to FW Pumps	58	6.5	12.4	0.0	2.0	379.0	8.9	0.0	0.0	12.86	19.3	
Feedwater: FW Pumps to FW Heaters 6	80	6.1	12.0	0.0	2.2	380.3	8.7	0.0	0.0	12.86	19.3	
Feedwater: FW Heaters 6 to Steam Generators	152	11.7	20.0	9.1	2.9	460.0	1.4	0.0	0.0	12.86	19.3	
Heater Drains: FW Heaters 6 to FW Heaters 5	73	2.3	10.0	-1.4	2.3	391.8	10.2	0.0	0.0	1.71	15.0	
Heater Drains: FW Heaters 5 to Deaerator	64	-0.9	4.2	-7.8	2.5	325.4	11.9	0.0	0.0	3.05	19.2	
Heater Drains: FW Heaters 3 to FW Heaters 2	74	10.3	33.3	-40.2	2.2	215.8	-2.3	0.0	0.0	0.61	23.9	
Heater Drains: FW Heaters 2 to FW Heaters 1	48	17.2	35.9	0.0	3.8	181.7	4.4	0.0	0.0	1.00	16.1	
Heater Drains: FW Heaters 1 to Condenser	38	8.8	11.8	0.0	2.1	124.3	12.0	0.0	0.0	1.50	7.2	

Table 2.1.8-1 Comparison of Current and EPU Analytical Wear Rates for CR-3 FAC

Location	No. of Comps Analyzed	Wear Rate Change			Avg. Post Wear Rate (mils/yr)	Temperature		Steam Quality		Flow Rate		Notes
Steam Cycle Location		Avg. Change (%)	Max Change (%)	Min Change (%)		Post Temp (°F)	Temp Change (°F)	Post Quality (%)	Quality Change (%)	Post Flow Rate (Mlb/hr)	Change in Flow Rate (%)	
Extraction Steam: HP Turbine to FW Heaters 6	60	0.0	0.0	0.0	0.1	467.2	-7.6	100.0	0.0	1.24	10.5	Superheated line showing no wear
Extraction Steam: HP Turbine to FW Heaters 5	10	16.8	20.8	15.2	24.1	387.2	12.1	93.0	-0.4	0.80	20.1	This line shows the greatest Average Post Wear Rate
Extraction Steam: HP Turbine to LP Reheaters	101	0.0	0.0	0.0	0.0	472.5	-1.3	100.0	0.0	0.54	32.0	Superheated line showing no wear
Reheat Steam: Moisture Separators to Condenser	122	7.1	17.6	0.0	0.8	391.0	13.6	0.0	0.0	0.68	27.8	
Heater Drains: HP Reheaters to FW Heaters 6	423	23.6	50.0	0.0	2.8	533.4	4.6	0.0	0.0	0.48	29.9	
Heater Drains: LP Reheaters to FW Heaters 5	413	30.3	100.0	0.0	3.3	471.5	11.4	0.0	-1.1	0.54	32.0	
Extraction Steam: LP Turbines to FW Heaters 3	97	338.2	350.0	320.0	13.8	277.4	4.5	98.8	-1.0	0.61	23.9	Greatest Post-EPU increase in wear rate. Large change due to decrease in steam quality. Even though the steam quality only decreased 1%, FAC wear rates increase drastically when conditions change from superheated steam to high-quality, two-phase flow.
Extraction Steam: LP Turbines to FW Heaters 2	26	20.0	25.3	15.1	4.1	214.7	1.6	77.0	-4.2	0.39	5.8	
Extraction Steam: LP Turbines to FW Heaters 1	58	36.1	40.7	27.0	4.2	177.5	5.2	72.4	-6.8	0.57	4.9	

Table 2.1.8-1 Comparison of Current and EPU Analytical Wear Rates for CR-3 FAC

Location	No. of Comps Analyzed	Wear Rate Change			Avg. Post Wear Rate (mils/yr)	Temperature		Steam Quality		Flow Rate		Notes
Steam Cycle Location		Avg. Change (%)	Max Change (%)	Min Change (%)		Post Temp (°F)	Temp Change (°F)	Post Quality (%)	Quality Change (%)	Post Flow Rate (Mlb/hr)	Change in Flow Rate (%)	
Feedwater Pumps Recirc to Deaerator	28	0.0	0.0	0.0	0.0	380.3	8.7	0.0	0.0	0.0	0.0	No measurable change. Very low flow resulting in insignificant wear rate.

2.1.9 Steam Generator Tube Inservice Inspection

2.1.9.1 Regulatory Evaluation

Steam generator (SG) tubes constitute a large part of the reactor coolant pressure boundary (RCPB). SG Tube Inservice Inspection (ISI) provides a means for assessing the structural and leaktight integrity of the SG tubes through periodic inspection and testing of critical areas and features of the tubes. The CR-3 review in this area covered the effects of changes in differential pressure, temperature, and flow rates resulting from the proposed EPU on plugging limits, potential degradation mechanisms (e.g., flow-induced vibration), plant specific alternate repair criteria and redefined inspection boundaries.

The NRC's acceptance criteria for Steam Generator Tube Inservice Inspection are based on:

- 10CFR50.55a requirements for periodic inspection and testing of the RCPB

Additional review guidance is contained in Regulatory Guide 1.121 for SG tube plugging limits, NRC Generic Letter 95-03 for degradation mechanisms, and Nuclear Energy Institute (NEI) 97-06 for structural and leakage performance criteria all of which form the basis for alternate repair criteria or redefined inspection boundaries.

CR-3 Current Licensing Basis

CR-3 requirements for periodic inspection and testing of SG tubes as part of the RCPB are based on 10CFR50.55a. The ISI program for the RCPB is discussed in FSAR Section 4.4.1. This includes a discussion of the SG ISI which is conducted in accordance with the CR-3 Steam Generator Tube Integrity Program. A program of periodic SG inspections, designed to meet the guidance of NEI 97-06, is conducted to provide assurance of acceptable SG performance. The Steam Generator Tube Inspection Program is contained in Improved Technical Specification (ITS) sections 5.6.2.10 and 5.7.2, which specifies a tube plugging criteria of 40% through wall. The guidance of NEI 97-06 applies to the Tube Integrity Program areas of: degradation assessment; inspections; tube integrity assessment; condition monitoring; operational assessment; tube plugging; leakage monitoring; secondary side integrity; water chemistry; foreign material exclusion; and, contractor oversight. Additional program guidance is provided in Regulatory Guide 1.121 and NRC Generic Letter 95-03.

CR-3 has no approved plant-specific alternate repair criteria or redefined inspection boundaries identified in ITS section 5.6.2.10.

2.1.9.2 Technical Evaluation

Introduction

SG process parameters will change as a result of the proposed EPU. Parameters that are expected to change include steam temperatures, steam pressure, steam and feedwater flows, Reactor Coolant System (RCS) temperatures, and RCS flow. The specific values are included in Section 1.1, Nuclear Steam Supply System Parameters, of this report.

CR-3 replaced steam generators in the Fall 2009 outage. The installed SGs utilize Alloy 690 Thermally Treated (TT) tubes and have a higher heat transfer capability, improved performance, and reduced susceptibility to tube wear.

Description of Analyses and Evaluations

Although the process parameter changes due to the EPU may impact the initiation and growth rates of various degradation mechanisms, these changes were considered per the assessments described below, and are incorporated into the selection of tubing inspection techniques and the extent and frequency of tubing inspections. Tube fretting wear is the principal degradation mechanism identified for Alloy 690 TT tubes. As described in Section 2.2.2.5.2.4, Steam Generators and Supports, flow-induced vibration (FIV) analysis and tube wear calculations were performed to demonstrate that the steam generator tubes continue to be adequately supported to prevent detrimental FIV and fretting wear at the EPU conditions.

The process of SG tube ISI and integrity assessment will not change as a result of the EPU. The CR-3 Steam Generator Tube Integrity Program begins with an assessment of potential degradation mechanisms and selection of applicable non-destructive examination techniques that will be used during the ISI to determine if any degradation exists. After performing the ISI, a condition monitoring assessment is performed to determine if there may have been structural or leakage integrity issues during the operating interval since the previous inspection. After employing conservative growth rates, an operational assessment is performed to ensure that structural and leakage integrity performance criteria will be met during the operating interval until the next inspection. Tubes that are not projected to meet the structural and/or leakage integrity criteria are then removed from service by plugging.

2.1.9.3 Conclusion

CR-3 has evaluated the effects of the proposed EPU on SG tube integrity and concludes that the evaluation has adequately assessed the continued acceptability of the CR-3 ITS under the proposed EPU conditions and has identified appropriate degradation management inspections to address the effects of temperature, differential pressure, and flow rates on SG tube integrity. CR-3 further concludes that SG tube integrity will continue to be maintained and will continue to meet the performance criteria in NEI 97-06 and the requirements of 10CFR50.55a following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Steam Generator Tube Inservice Inspection.

2.1.9.4 References

None

2.1.10 Steam Generator Blowdown System

2.1.10.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of steam generator (SG) tubes. The functions of the SG Blowdown System are to provide removal of SG secondary-side impurities and assist in maintaining acceptable secondary-side water chemistry in the SGs. The CR-3 review of the SG Blowdown System focused on the effects that the proposed EPU will have on the functional performance of the system and covered the ability of the SG Blowdown System to remove particulate and dissolved impurities from the SG secondary side during normal operation, including anticipated operational occurrences (main condenser inleakage and primary-to-secondary leakage).

The NRC's acceptance criteria for Steam Generator Blowdown System are based on:

- GDC-14, insofar as it requires that the Reactor Coolant Pressure Boundary (RCPB) be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are applicable CR-3 specific criteria:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture [GDC 14];

2.1.10.2 Technical Evaluation

Introduction

The SG Blowdown System at CR-3 is only used for secondary side water chemistry control at plant power levels below 15%. At power levels at or above 15%, the respective isolation valves are maintained closed with the Blowdown System secured. Unlike recirculating steam generators, the once through steam generator design does not employ continuous blowdown at power levels above 15% as a secondary steam generator water chemistry control method.

Description of Analyses and Evaluations

The SG Blowdown System and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluations were performed for thermal power of 455 MWt.

CR-3 has a once through steam generator (OTSG) which only uses the Blowdown System during start-up to 15% power. The ability of the SG Blowdown System to remove particulates and dissolved impurities during normal operation is not part of the design basis for CR-3. The condensate polishers remove significant quantities of Na, Cl, K, sulfate, and organic acids from the condensate during power operation. Above 15% power the Secondary Water Chemistry Program monitors appropriate parameters and responds to levels or trends above pre-determined levels. The Secondary Water Chemistry Program is based on EPRI and NEI guidance "PWR Secondary Water Chemistry Guidelines – Revision 7", EPRI 1016555, February 2009. "Steam Generator Program Guidelines", NEI 97-06, Revision 2, May 2005.

Results

For the EPU, at reactor power up to 15%, the normal SG blowdown flow will be unchanged. The OTSG impurities are not expected to be significantly higher at EPU conditions. The CR-3 blowdown system is isolated and removed from service at power levels at or above 15%. There is no impact of the EPU on SG blowdown or its effectiveness in controlling steam generator secondary side chemistry below 15% power.

At EPU conditions, for plant power levels at or above 15%, the operating temperature in the OTSGs, SG blowdown tank and interconnecting piping and valves increase slightly due to the higher Tave. However, as noted in FSAR Table 4-4, the existing secondary side design pressure and temperature of the OTSGs (1050 psig / 600°F) remain bounding for EPU conditions since these values are based on the no-load operating conditions which do not change as a result of the EPU. Therefore, the design conditions for the SG blowdown piping and components connected to the OTSGs also remain bounded and unchanged for EPU conditions.

Monitoring of effluent release due to SG blowdown will be the same with EPU conditions.

2.1.10.3 Conclusion

The effects of the proposed EPU on the SG Blowdown System have been evaluated for changes in system flow and impurity levels. CR-3 concludes that SG Blowdown System will continue to be acceptable and will continue to meet the requirements of FSAR Section 1.4.9 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to SG Blowdown System.

2.1.10.4 References

None

2.1.11 Chemical and Volume Control System

2.1.11.1 Regulatory Evaluation

The Chemical and Volume Control System (CVCS) functions are performed by two separate systems at CR-3. The first is the Chemical Addition (CA) System, which includes the Liquid Sampling System, and the second is the Makeup and Purification (MU) System. These systems provide a means for (a) maintaining water inventory and quality in the Reactor Coolant System (RCS), (b) supplying seal-water flow to the reactor coolant pumps (RCPs) and pressurizer auxiliary spray, (c) controlling the boron neutron absorber concentration in the reactor coolant, (d) controlling the primary water chemistry and reducing coolant radioactivity level, and (e) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the emergency core cooling system (ECCS) in the event of postulated accidents. The CR-3 review of CVCS focused on the safety-related functional performance characteristics of CVCS components.

The NRC's acceptance criteria for the Chemical and Volume Control System are based on:

- GDC-14 insofar as it requires that the Reactor Coolant Pressure Boundary (RCPB) be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; and
- GDC-29 insofar as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of anticipated operational occurrences (AOOs).

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR, Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in the FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. [GDC-14]

Additionally, FSAR Section 7.1 provides criteria for protection systems insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs [GDC-29].

2.1.11.2 Technical Evaluation

Introduction

The CVCS functions are performed by two separate systems at CR-3. The first is the Chemical Addition (CA) System, which includes the Liquid Sampling System, and the second is the Makeup and Purification (MU) system. Each of the systems mentioned above provide both normal and accident support for the reactor. The CA System is described in FSAR Section 9.2 and the MU System is described in FSAR Section 9.1.

The CA and MU Systems provide a means for:

- Water Quality and RCS Inventory Control for normal operation.

  Water quality of the RCS is maintained by the CA and MU systems through the use of various filters, demineralizers, and chemicals. Filters are used to remove particulate debris from the RCS letdown flow that would be detrimental to the system. Demineralizers are used to control specific ionic debris in the RCS by use of specialized resins that have been designed to have an affinity for cations, anions, or both. Various chemicals are used to control different aspects of the RCS water quality.

  The MU System serves to control reactor coolant inventory and boric acid concentration in the RCS through "letdown" and "makeup" of borated water. The letdown and makeup process also accommodates thermal expansion and contraction of the reactor coolant (RC) temporarily during plant startup and shutdown.

- Seal Injection Flow and Auxiliary Pressurizer Spray for normal operation and accident conditions.

  The MU System provides seal injection flow to the RCPs during normal operation and some accident scenarios. Two RCP seal injection filters of the disposable cartridge type are installed in parallel to prevent particulate matter from entering the RCP seal cavities. One filter is normally used.

- Control of the Boric Acid Neutron Absorber Concentration in the RCS for normal operation.

  CR-3 operates with the highest concentration of boric acid in the core at beginning of life (BOL). Throughout the operating cycle the concentration is diluted using the letdown, makeup, and purification processes to maintain reactor power. During the dilution cycle, boric acid concentration is reduced in the core using a bleed and feed cycle, whereby RC fluid is siphoned off and replaced with demineralized water or filtered through the deborating ion exchanger and re-injected to the RCS using the MU pumps. The FSAR describes several safeguards that are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant.

  Boric acid is also used during emergency (accident) conditions to reach cold shutdown via the emergency borate function (See FSAR Sections 4.2.5.5 and 9.2).

- Control of the Primary Water Chemistry and Reduction of Radioactivity for normal operation.

Two mixed-bed purification demineralizers are installed upstream of the MU post-filters. The demineralizers are boric acid saturated and used to remove reactor coolant impurities other than boron. Since the reactor coolant may be contaminated with fission and corrosion products, the demineralizer resins will remove certain radioactive impurities.

- Reactor Coolant System Cleanup and Inventory Maintenance.

  To perform this function, a continuous bleed and feed process is maintained between the RCS and the MU System. Water is extracted off the RCS loop piping and is directed through coolers to reduce the temperature and therefore prevent damage to the downstream demineralizer resins. The letdown water then flows through a pressure-reducing orifice. Depending on the sampled water chemistry, the extracted RCS fluid can be directed through the demineralizer vessel that contains the appropriate ionic debris filtering resin (deborating resin, mixed bed resin, or cation resin). Inline filters ensure particulate debris is filtered from the flow to prevent damage to system components. After being filtered and passed through the appropriate demineralizer vessel, the flow is returned to the RCS by the MU pumps. Additionally, makeup to the RCS can be performed using the boric acid and demineralized feeds to the makeup tank to ensure system volumes are maintained.

- Supplying high-pressure injection flow to the Emergency Core Cooling System (ECCS) in the event of postulated accidents.

  The ECCS function is discussed in Section 2.8.5.6.3, Emergency Core Cooling System and Loss of Coolant Accidents.

Description of Analyses and Evaluations

The CA and MU Systems have been evaluated to ensure their capability to provide the required safety functions as defined above following the EPU. The evaluation is performed for operation at the EPU conditions.

Under the EPU, the RCS average coolant temperature (T_{AVE}) will increase by approximately 3°F to 582°F; however the T_{COLD} will decrease by less than 1°F. The RCS pressure is unchanged by the EPU. The Letdown flow does not change as a result of the EPU.

The EPU operating conditions that could potentially affect the function of the CA and MU components include the increase in core power and allowable range of RCS full-load temperatures. The increase in core power and the allowable range of RCS full-load design temperatures may also affect the design bases requirements related to the core reload boron requirements.

Letdown Coolers:

The letdown coolers cool the RCS letdown flow from the RCS to prevent damage to demineralizer resin and other downstream components. The letdown coolers use the Nuclear Services Closed Cooling Water (SW) System as a cooling medium. The EPU letdown flow temperature will decrease slightly as stated above. The maximum required heat rejection will subsequently decrease. Therefore, no changes are required as a result of the EPU.

Seal Return Coolers:

There are two RCP seal return coolers each sized to remove the heat in the MU pump recirculation flow and the heat picked up in passage through the RCP seals. Heat in the RCP seal return coolers is rejected to the SW System. Normally one cooler is in operation. The decrease in T_{COLD} may require the flow adjustments to the seal return coolers to ensure the tubeside ΔT is maintained within the range defined in FSAR Table 9-2. Flow adjustments to achieve the acceptance criteria of FSAR Table 9-2 are covered by operator procedure. The EPU requirements are within the capability of the existing system and no changes are required.

Charging, Letdown, and RCS Makeup (Boration, Dilution, Purification):

As discussed previously, the net effect on the performance of Letdown, and Makeup and purification functions is essentially unchanged as a result of the EPU. Since the demineralized water and boric acid feed lines are downstream of the letdown heat exchangers, the EPU has no effect on the performance of these actions.

Makeup capability is a function of process fluid pressure and temperature. As stated there will be no change in the EPU RCS pressure, and consequently the required discharge head of the MU pumps. The pumps, therefore, will continue to produce the same flow but with a decreased RCS T_{COLD} and fluid to the pump inlets temperature at or less than the current temperature. Hence, the existing MU pumps are adequate and no changes are required.

Core reactivity control to maintain or restore Axial Power Imbalance (API) operating limits (ITS 3.2.3) is limited by the rate of Reactor Coolant System (RCS) boration or dilution through the Makeup (MU) System Tank, MUT-1, to control the controlling rod group position. The EPU API operating limits are expected to be more restrictive than the pre-EPU limits as shown in Section 2.8.2, Nuclear Design. Improved response time in core reactivity control will assist in maintaining or restoring API operating limits and will be achieved by decreasing the dilution/mixing time by installing a bypass line at MUT-1. The bypass line will be used to reduce the batch dilution / mixing time and will be controlled by a new 3-way valve where the operator may select makeup flow from MUT-1 or from the bypass line at the Main Control Board. The 3-way valve will be installed downstream of the existing batch controller and post filters to ensure batch processing remains unchanged.

Boric Acid Storage Tanks (BAST) and Borated Water Storage Tank (BWST)

The design of the BAST and BWST is unaffected by the EPU as the change in core power and temperatures are physically separated from these components. The minimum boron concentration in the BWST will increase from 2270 ppm to 2600 ppm to support the EPU. Requirements for cold shutdown will be satisfied by either increasing the minimum borated water concentration and/or volume in the BAST(s). The required concentration/volume relationship along with minimum solubility temperature limits will be discussed in an update to Section 4.2.5.5 of the FSAR and controlled by station surveillance procedures. The increase for the BWST boron concentration is discussed in Section 2.8.2, Nuclear Design.

Results

The evaluations of the CA and MU Systems that provide RCS letdown and makeup performance show the CA and MU Systems are acceptable at the EPU conditions, with no plant changes required. Accordingly, the performance of the following CA and MU functions (which are accomplished via letdown, makeup, and purification) are acceptable at the EPU conditions with no plant changes:

- The boration capability is addressed during the Core Operating Limits Report (COLR) process currently incorporated into the CR-3 Technical Specifications for each core re-load cycle
- The CA and MU support functions provided by the sampling system and waste disposal system are not affected by the change in RCS conditions resulting from the EPU
- The performance of the CA and MU components including valves and piping that support containment isolation are not affected by change in RCS design parameters resulting from EPU.
- A bypass line around MUT-1 will allow an increased response time in core reactivity by decreasing the batch settling time within MUT-1.

2.1.11.3 Conclusion

CR-3 has evaluated the effects of the proposed EPU on the systems performing the chemical addition and volume control functions (i.e., the Chemical Addition and Makeup Systems) and concludes that the evaluation adequately addresses changes in temperature of the reactor coolant and their effects on the CA and MU systems. CR-3 further concludes that the CA and MU systems will continue to be acceptable and will continue to meet the requirements of FSAR Sections 1.4.9 and 9.2.1 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Chemical Addition and Makeup Systems.

2.1.11.4 References

None

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

2.2.1.1 Regulatory Evaluation

SSCs important to safety could be impacted by the pipe whip dynamic effects of a pipe rupture. CR-3 conducted a review of pipe rupture analyses to ensure that SSCs important to safety are adequately protected from the effects of pipe ruptures. The review covered (1) the implementation of criteria for defining pipe break and crack locations and configurations, (2) the implementation of criteria dealing with special features, such as augmented inservice inspection (ISI) programs or the use of special protective devices such as pipe whip restraints, (3) pipe whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe whip dynamic effects, and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe whip or jet impingement loadings.

The NRC's acceptance criteria for Pipe Rupture Locations and Associated Dynamic Effects are based on:

- GDC-4, which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, Missile Protection, which requires SSCs important to safety to be designed to accommodate the dynamic effects of a postulated pipe rupture [GDC-4]

Additionally, FSAR Section 14.2.2.5.11 discusses the approved methodology for "Leak-Before-Break" described in Babcock & Wilcox topical report BAW-1847, Rev. 1, "Leak-Before-Break Evaluation of Margin against Full Break for RCS Primary Piping of B&W Designed NSS," which was granted as an exemption (References 1, 2, and 3).

2.2.1.2 Technical Evaluation

<u>Introduction</u>

CR-3 has utilized leak-before-break (LBB) to demonstrate that certain high energy lines were designed, constructed, and analyzed so as to have a negligible probability of failure as part of their design basis. Following the application of LBB, the remaining pipe breaks in the mechanical design basis of the Reactor Coolant System (RCS) occur in the surge line, core flood line, decay heat line, main steam line and main feedwater line. The licensing basis for Pipe Rupture Locations and Associated Dynamic Effects for CR-3 is identified in the FSAR sections 5.4.4.1 and 5.4.4.2.

Refer to Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, for discussion of plant design for protection from piping failures outside containment.

High energy lines are defined in Table 2.2.1-1 as operating conditions A. They are protected from full break effects (including pipe whip, jet impingement, pressurization, flooding and environmental effects) based on analyzed piping stresses.

Description of Analyses and Evaluations

The following discussion relates to postulated pipe breaks in the RCS not credited with LBB. (LBB is discussed in Section 2.1.6, Leak-Before-Break). The high energy attached piping systems of the RCS and the surge line were evaluated to address EPU conditions. Since there is no change in piping configuration and stresses remain within code allowable values for the RCS attached piping and the surge line, the evaluations for the RCS piping did not result in any new or revised break locations, and the design basis break locations remain valid for EPU.

Hydraulic pipe rupture loads for the design basis high energy line break (HELB) locations in the RCS were evaluated considering EPU conditions. These loads are a result of asymmetric cavity pressure (ACP), jet impingement, thrust, internal forces generated from changes in area and flow direction, and internal pressures acting on components.

The following pipe ruptures are located at the nozzle end attachments, and were included as part of the RCS structural evaluation:

- Core Flood Line Break
- Decay Heat Line Break
- Surge Line Break at Hot Leg Nozzle
- Intermediate Surge Line Break
- Single Feedwater Line Breaks (East and West side of Steam Generator)
- Double Feedwater Line Break
- Main Steam Line Breaks (East and West side of Steam Generator)

The impact on the RCS loop loss of coolant accident (LOCA) hydraulic forcing functions due to the EPU is addressed in Section 2.8.5.6.3.3, Technical Evaluation – LOCA Forces. These loads were used in the structural evaluation of the RCS piping, components and supports.

Those systems defined by operating condition A are protected from full break effects (pipe whip, jet impingement, pressurization, flooding and environmental effects) based on analyzed pipe stresses. For piping whose operating condition is defined by A, B and C, protection is provided from the effects of cracks (including jet impingement, flooding and environmental conditions) at the most adverse locations for all pipes. For lines defined by operating condition D, no protection for postulated pipe rupture is considered.

The evaluation completed at EPU conditions for pipe breaks outside containment considered the zones within the plant, which contain systems required for safe shutdown and/or systems required to mitigate the effects of the postulated breaks.

Affected Balance-of-Plant (BOP) piping (non-Class 1) systems were evaluated to address EPU operating conditions. Applicable pipe rupture criteria were reviewed and changes to piping system stress levels resulting from EPU were reconciled against the design basis calculations. The evaluations performed for these piping systems did not result in any new or revised break locations, and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for EPU.

Based on the evaluations performed for the EPU noted above, the following were demonstrated:

- Existing criterion for defining pipe break and crack locations and configurations is unaffected by the EPU.
- Existing pipe whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe whip dynamic effects continue to meet the acceptance criteria for the EPU conditions.
- Existing design of SSCs remains acceptable to protect safety related SSCs from the effects of pipe whip and jet impingement loading for the EPU.

The design features for CR-3 that protect safety related structures, systems and components from the consequences of postulated piping failures both inside and outside containment as described in FSAR Sections 5.4.4.1 (for inside containment) and 5.4.4.2 (for outside containment) remain valid for EPU.

Results

The results of the reanalysis (considering EPU conditions) of the applicable HELB locations not credited with LBB confirm that the stresses and fatigue usage of the surge line and RCS attached piping remain within the design basis allowables and thus there are no additions or changes to the HELB locations inside the containment building. (See Section 2.2.2.1, NSSS Piping, Components and Supports, for more detailed discussion of the structural evaluations performed for NSSS piping, components and supports.) Additionally, the reanalysis showed the RCS primary piping, surge line, and RCS attached piping capability to withstand effects associated with high-energy pipe rupture is maintained.

For BOP piping (non-Class1) systems, the evaluations for EPU conditions did not result in any new or revised break locations, and the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for EPU. The BOP piping (non-Class 1) and support systems continue to meet their licensing basis.

2.2.1.3 Conclusion

CR-3 has reviewed the evaluations related to determination of Pipe Rupture Locations and Associated Dynamic Effects and concludes that the evaluations have adequately addressed the effects of the proposed EPU. CR-3 further concludes that the evaluations have demonstrated that those SSCs important to safety will continue to meet the requirements of FSAR Section 1.4.40 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the determination of Pipe Rupture Locations and Dynamic Effects.

2.2.1.4 References

1. BAW-1847, Rev 1, “B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping” dated September 1985

2. Safety Evaluation Report: Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, February 18, 1986.

3. License Amendment No. 89: NRC to CR-3 letter, dated May 23, 1986, "Crystal River Unit 3 - Amendment to Facility Operating License".

Table 2.2.1-1 Criteria Applicability Based on Operating Temperature/Pressure Limits

Operating Condition	Temperature °F	Pressure (psig)
A	Greater than 200	Greater than 275
B	Greater than 200	Less than or Equal to 275
C	Less than or Equal to 200	Greater than 275
D	Less than or Equal to 200	Less than or Equal to 275

2.2.2 Pressure-Retaining Components and Component Supports

2.2.2.0.1 Regulatory Evaluation

CR-3 has reviewed the structural integrity of Pressure-Retaining Components (and their supports) designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section III, Division 1. The CR-3 review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The CR-3 review covered (1) the analyses of flow-induced vibration and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The CR-3 review also included a comparison of the resulting stresses and cumulative fatigue usage factors (CUFs) against the code-allowable limits.

The NRC acceptance criteria for Pressure-Retaining Components and Component Supports are based on:

- 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed;

- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions;

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;

- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and

- GDC-15, insofar as it requires that the Reactor Coolant System (RCS) be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.1, Quality Standards, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed. [GDC-1] The quality control and quality assurance program for CR-3 operational activities is described in FSAR Section 1.7;

- FSAR Section 1.4.2, Performance Standards, insofar as it requires that the SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions [GDC-2];

- FSAR Section 1.4.23, Protection Against Multiple Disability for Protection Systems, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents [GDC-4];

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture [GDC-14]; and

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS be designed with margin sufficient to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation. [GDC-15]

Additionally, FSAR Sections 4.1.3 and 4.4.1 provide criteria for insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed in conformance with 10 CFR 50.55a . [10 CFR 50.55a] [GDC-1]

Additional information on piping and supports can be found in FSAR Sections 1.3.2.8 and 1.3.2.12.

2.2.2.0.2 Technical Evaluation

During the installation of replacement once through steam generators (OTSGs), the piping, components, and component supports of the RCS loops were analyzed and structurally qualified. These analyses were conservatively done at the EPU conditions so that the RCS would be structurally qualified for both the pre-EPU and post-EPU plant configurations for design conditions (normal, upset, emergency and faulted in accordance with ASME Section III Criteria). As such, the subsections of 2.2.2, Pressure-Retaining Components and Component Supports, justify that the RCS loop is structurally qualified for the EPU by showing either that: (a) the loads on the RCS loop under the EPU conditions are less than the loads analyzed prior to the replacement OTSG installation (and therefore, prior to the EPU), or (b) for cases where the EPU loads were higher than original OTSG loads, an ASME Code stress/fatigue analysis was performed to demonstrate computed stresses and CUFs are within allowable values.

This Section is an introduction for the subsequent sections (2.2.2.1 through 2.2.2.8). The Technical Evaluation portion will be presented in each of the individual sections as appropriate.

2.2.2.0.3 Conclusions

CR-3 has reviewed the structural integrity of pressure-retaining components and their supports. For the reasons set forth in Sections 2.2.2.1 through 2.2.2.8, CR-3 concludes that the evaluations adequately and appropriately address the effects of the proposed EPU on these components and their supports. Based on the above, CR-3 further concludes that the evaluation has demonstrated that Pressure-Retaining Components and their Supports will continue to meet the requirements of 10 CFR 50.55a and FSAR Sections 1.4.1, 1.4.2, 1.4.9, and 1.4.23 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the structural integrity of the Pressure-Retaining Components and Component Supports.

2.2.2.1 NSSS Piping, Components, and Supports

2.2.2.1.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation.

2.2.2.1.2 Technical Evaluation

Introduction

The NSSS piping, which is the Reactor Coolant System (RCS) piping, consists of two heat transfer piping loops (loops A and B) connected in parallel to the reactor vessel (RV). Each loop contains two circulating pumps called the reactor coolant pumps (RCP) and a once-through steam generator (OTSG). Each RCS loop consists of five legs: the hot leg from the RV to the OTSG, the two lower cold legs from the OTSG to the RCPs, and two upper cold legs from the RCPs to the RV. The system also includes a pressurizer, surge line, connecting piping, and the instrumentation for operational control. The pressurizer is connected to the A loop hot leg. Auxiliary system piping connections into the RCS piping are provided as necessary. The RCS piping system is supported by the primary equipment supports of the RCS, namely the RV supports, the OTSG supports, the RCP supports, and the pressurizer supports.

The NSSS Piping, Components, and Supports are described in the FSAR sections 3.1, 4.1, 4.2, 4.3, 4.4, and 4.6.

Description of Analyses and Evaluations

As discussed in Section 2.2.2, the current design basis structural analysis was performed for the RCS loop due to replacement of the OTSGs, but also considered the proposed EPU conditions. This analysis employed the same methodology as used in the pre-OTSG replacement (pre-EPU) structural analyses. Specifically, the following analyses were performed and, where necessary, reanalyzed with EPU parameters:

- RCS loop loss-of-coolant accident (LOCA) analysis using EPU Loop LOCA hydraulic forces
- RCS loop piping stresses
- RCS loop displacements at auxiliary piping line connections to the centerline of the RCS loop at branch nozzle connections and impact on the auxiliary piping systems
- Primary equipment nozzle loads
- RCS loop piping system leak-before-break (LBB) loads for LBB evaluation
- Pressurizer surge line piping analysis including the effects of thermal stratification
- RCS loop primary equipment support loads (RV, OTSG, and RCP)

Changes due to the EPU of the following three basic sets of input parameters were evaluated for their impact on the structural integrity of the NSSS Piping, Components, and Supports.

- NSSS operating conditions – Design parameters for 3014 MWt Core Thermal Power as shown in Table 1.1-1 were used in the thermal analysis of the RCS loop and used in the evaluation for the pressurizer surge line. The RCS hydraulics model used as input to the structural evaluations was modified to predict RCS conditions using the higher core power. The cases considered in the hydraulic analysis are described in Section 1.1, Nuclear Steam Supply System Parameters.

- NSSS Design Transients – The impact on design transients due to the changes in full-power operating temperatures for the EPU program is addressed in Section 2.2.2.8, NSSS Design Transients. The post-EPU design transients were considered in the stress and fatigue evaluations of the RCS primary piping.

- Loop LOCA hydraulic forcing functions forces – The impact of the EPU on the Loop LOCA hydraulic forcing functions is addressed in Section 2.8.5.6.3.3, Technical Evaluation – LOCA Forces. LOCA loadings were generated for postulated surge line, decay heat line, main steam line, main feedwater line and core flood line breaks. These loads were applied to the loop model and the resulting stresses/fatigue usages were evaluated.

The acceptance criteria for the RCS primary loop piping are based upon the USAS B31.7, Nuclear Power Piping, Draft USA Standard (Dated February 1968) Code for Pressure Piping, including June 1968 Errata.

The pressurizer surge line was reconciled to the ASME B&PV Section III, Subsection NB 1986 Code, and includes the fatigue evaluation and the effects of thermal stratification as stipulated in NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, December 20, 1988. Thus, the acceptance criteria for the pressurizer surge line thermal stratification analysis are per the ASME B&PV Code, and are as specified in the current design basis in BAW-2127 “Final Submittal in Response to NRC Bulletin 88-11” (Reference 1).

The effects of flow induced vibration were considered for the NSSS Piping and Components. Based on a negligible change in the RCS flow rate (see Table 1.1-1), the NSSS Piping and Components are unaffected by the proposed EPU conditions with respect to flow induced vibration.

RCS Loop Analysis

The operating parameters, design transients and LOCA forcing functions that will change due to the EPU were reviewed for impact on the existing RCS loop piping and consequent impact to the auxiliary lines attached to the RCS loop centerline at the RCS loop branch nozzle connections. The RCS loop piping has been re-analyzed considering these changes for all applicable deadweight, thermal expansion, operational basis earthquake (OBE), safe shutdown earthquake (SSE) and LOCA load cases and appropriate combinations thereof.

The deadweight analysis for the EPU was performed considering the weight of the RCS loop piping, components, supports (as necessary), water weight and other component attached weights (main/emergency feedwater header and nozzles). The deadweight analysis inputs have not changed as a result of EPU.

The thermal analysis evaluated the RCS loop at temperatures bounding those provided in Table 1.1-1. Specifically, T_{HOT} was evaluated at 611°F and T_{COLD} was evaluated at 553°F which envelopes the various cases listed in Table 1.1-1.

The seismic analysis was performed for both OBE and SSE using the response spectrum method. The SSE and OBE input response spectra for the EPU parameters are the same as used in the current licensing basis analysis.

LOCA and pipe rupture analyses were performed using the internal forcing functions, asymmetric cavity pressures, jet impingement loads and thrust time histories generated from RCS hydraulic analyses considering EPU conditions. The postulated pipe ruptures considered were the surge line, decay heat line, main steam line, main feedwater line and core flood line breaks (see Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, for additional discussion of pipe rupture locations).

For the pressurizer surge line piping, the impact of the design transients with respect to the thermal stratification and fatigue analysis is controlled by the ΔT between the pressurizer temperature and the hot-leg temperature. The controlling ΔTs for the pressurizer surge line are associated with the plant heatup and cooldown events which are not affected by EPU conditions. The operating conditions and the design transients affected by the EPU have an insignificant effect on the pressurizer surge line analysis. The effects of thermal stratification on the surge line piping are not negatively impacted by the EPU since the ΔT between the hot leg and the pressurizer is decreasing for the EPU (based on T_{HOT} increasing for the EPU and pressurizer temperature staying the same). Therefore, the EPU has no adverse impact on either the thermal stratification or the fatigue analysis for the pressurizer surge line, and the results of BAW-2127 remain valid.

The impact of the EPU-related changes to operating parameters, design transients and LOCA forcing functions on the RCS loop primary components and component supports is addressed in Section 2.2.2.3 for the Reactor Vessel and Supports, Section 2.2.2.5 for the Steam Generators and Supports, Section 2.2.2.6 for the RCPs and Supports, and Section 2.2.2.7 for the Pressurizer and Supports.

Results

Based on the evaluations performed for the EPU program NSSS Design Parameters, NSSS design transients and RCS loop LOCA hydraulic forcing functions, the current RCS loop piping will continue to meet the current design basis structural acceptance criteria for the EPU.

The RCS loop piping stress results for the EPU conditions are provided below in Table 2.2.2.1-1. These results summarize the maximum stress ratios (ratio of allowable stress) and fatigue usage factor for all large bore RCS piping (hot legs, upper cold legs and lower cold legs).

The RCS loop calculated stresses were combined in accordance with the ASME Code. The results of the RCS loop analyses confirm that the RCS loop piping stresses and fatigue usage factors meet the acceptance criteria and are acceptable for the EPU.

The RCS loop components and supports have also been evaluated considering the EPU parameters and are discussed in Section 2.2.2.3 for the Reactor Vessel and Supports, Section 2.2.2.5 for the Steam

Generators and Supports, Section 2.2.2.6 for the RCPs and Supports, and Section 2.2.2.7 for the Pressurizer and Supports.

The applicable RCS loop piping loads considering the EPU were provided for evaluation and confirmation of LBB. This evaluation is discussed in Section 2.1.6, Leak-before-Break.

The impact of the EPU program parameters on RCS piping displacements at the intersection of the centerline of the RCS piping and the auxiliary line piping system branch nozzle connections have been evaluated. This evaluation has shown that RCS attached piping systems will continue to meet their structural acceptance criteria and remain acceptable following the proposed EPU.

For the pressurizer surge line, the impact of the design transients with respect to the thermal stratification and fatigue analysis is controlled by ΔT between the pressurizer temperature and the hot-leg temperature and has been evaluated. The controlling transients for the pressurizer surge line are associated with the plant heatup and cooldown events, and are not affected by the EPU. Also, the environmental effects of fatigue on the pressurizer surge line have been previously evaluated and the results of those evaluations remain valid for EPU. Therefore, the current design basis pressurizer surge line analysis results, including the effects of thermal stratification, are applicable for the EPU and meet the acceptance criteria for the EPU program.

2.2.2.1.3 Conclusion

See Section 2.2.2 for Conclusion.

2.2.2.1.4 References

1. BAW-2127, Final Submittal in Response to NRC Bulletin 88-11 “Pressurizer Surge Line Thermal Stratification and Supplement 2 (Rev. -02)”

Table 2.2.2.1-1: Maximum RCS Piping Stress Ratios and Fatigue Usage

ASME Code Equation (NB-3650)	Calculated Stress (ksi)	Allowable Stress (ksi)	Ratio of Calculated Stress to Allowable Stress
Primary Stress Intensity (Equation 9)	24.59	36.18	0.68
Primary Plus Secondary Stress Intensity Range (Equation 10)	63.7	60.3	1.056 (1)
Simplified Elastic-Plastic Check (Equation 12)	39.6	60.3	0.656
Simplified Elastic-Plastic Check (Equation 13)	45.5	58.2	0.781
Max Cumulative Fatigue Usage Factor	0.038		

Notes:

(1) The equation 10 stress ratio is greater than 1.0; however, in accordance with the ASME Code (NB-3650) this is shown acceptable by meeting the criteria of the simplified elastic-plastic analysis (Equations 12 and 13 stress ratios less than 1.0) and thermal stress ratcheting checks.

2.2.2.2 Balance of Plant Piping, Components, and Supports

2.2.2.2.1 Regulatory Evaluation

See Section 2.2.2, Pressure-Retaining Components and Component Supports, for Regulatory Evaluation.

2.2.2.2.2 Technical Evaluation

Introduction

This section covers the piping and supports that are not included in Section 2.2.2.1, NSSS Piping, Components and Supports. Section 2.2.2.1 covers the reactor coolant loop piping and supports up to the Reactor Coolant System (RCS) loop class break. This section covers all non RCS loop piping and supports (hereafter referred to as Balance of Plant (BOP) Piping, Components and Supports), whether inside or outside containment. The BOP systems were evaluated to assess the impact of operating temperature, pressure and flow rate changes that will result due the implementation of EPU. The CR-3 design code of record for the BOP piping is Code for Pressure Piping ANSI/USAS B31.1.0 - 1967 (FSAR Section 5.4.4). The piping and supports for the following BOP systems were evaluated for EPU conditions:

- Auxiliary Steam
- Circulating Water
- Condensate
- Core Flooding
- Decay Heat Closed Cycle Cooling
- Decay Heat Removal
- Emergency Feedwater
- Extraction Steam
- Feedwater
- Heater Drains
- Heater Vents
- Turbine Auxiliary Systems
- Main Steam
- Makeup and Purification
- Nuclear Services Closed Cycle Cooling Water

- Nuclear Services Sea Water Cooling
- Radioactive Waste Disposal
- Reactor Building Spray
- Sampling System
- Secondary Services Closed Cycle Cooling
- Spent Fuel Cooling

Description of Analyses and Evaluations

System operation at EPU conditions generally results in increased pipe stress levels and pipe support and equipment loads when those SSCs experience higher operating temperatures, pressures or flow rates.

Pre-EPU and EPU operating data (operating temperatures, pressures, and flow rates) were obtained from heat balance diagrams, calculations, and other reference documents. Thermal "change factors" were determined, as required, to compare changes in thermal operating conditions. The thermal "change factors" were based on the ratio of the EPU to pre-EPU operating temperature. The thermal change factor is:

$$(T_{EPU} - 70^{o}F) / (T_{pre\text{-}EPU} - 70^{o}F)$$

Based on the magnitude of the calculated thermal change factors, the following engineering activities were performed and/or conclusions reached:

- For thermal change factors less than or equal to 1.00 (that is, the pre-EPU condition envelopes or equals the EPU condition), the piping system was concluded to be acceptable for EPU conditions.
- For thermal change factors greater than 1.00, an additional evaluation was performed to address the specific increase in temperature, in order to determine piping and support acceptability.

Operating pressure increases due to EPU were minor and mostly affected systems related to the main power cycle (main steam, feedwater, condensate, extraction steam, heater drains). Since the pipe stress evaluations for piping systems at CR-3 have used the system design pressure in accordance with USAS B31.1, the small increases in operating pressures were acceptable, provided the EPU operating pressure remains within the current design pressure of the system.

Flow rate increases due to EPU occurred in systems related to the main power cycle. For the BOP piping, fluid transients were considered in systems with relief valves or in-line valves with fast acting closure. Fast acting valves are defined by Electric Power Research Institute (EPRI) Guidelines as any valve with a closure time of 1 second or less (Reference 1). Flow rate increases and their impact on potential flow-induced fluid transient loads were evaluated for the main steam and auxiliary steam piping systems. The evaluation of the main steam and auxiliary steam piping systems addressed the system flow rate increase and its impact on fluid transient loads (e.g., steam hammer loads) resulting from a turbine stop valve closure event. The review of the feedwater system addressed the flow rate increase and its impact on fluid transient loads (i.e., water hammer loads) resulting from valve closure/feedwater pump trip events. The Emergency Feedwater System and portions of the Heater Drain System were also evaluated for fluid transient events (i.e., water hammer loads) resulting from rapid valve closure. The remaining piping systems potentially impacted by EPU do not contain any fast closing valves and thus will not introduce any significant flow induced transients into the systems. Hence, flow rate increases due to EPU for the remaining piping systems were determined not to require evaluation of transient type loading events.

There was no change to seismic inputs (amplified response spectra) or loads resulting from EPU. The existing seismic design basis (FSAR Sections 5.1.2 and 5.4.5.2) for all piping and supports remains valid and unaffected by EPU. Hence, BOP piping and support seismic loadings will continue to meet the CR-3 current licensing basis with respect to the requirements of FSAR Section 1.4.2.

All piping analyses and support evaluations, whether performed via hand evaluations or computer analyses, were consistent with the requirements of the design code of record, USAS B31.1.0 – 1967. The following computer programs were used in performing the EPU piping and pipe support evaluations. These programs are used to calculate stresses and loads using the appropriate equations from the USAS B31.1 criteria. Using an approved Quality Assurance (QA) program, the software has been verified and validated and shown to be accurate. Thus the programs are appropriate for use in QA category 1 Nuclear Safety Related applications.

- AutoPIPE – This program performs rigorous pipe stress analysis to the requirements of multiple codes, including USAS B31.1.0 – 1967.
- GT-STRUDL – This program is a general purpose structural analysis software used to qualify pipe supports.
- RELAP5/MOD2-B&W – This program is used to analyze fast fluid transients, including water/steam hammer. Specifically, the program was used to develop the force-time histories that are imposed due to the closing of fast acting valves.
- SHRLUG1 – This program is used to perform the local stress evaluation of integral welded attachment (rectangular lug on straight pipe) per Code Case N-318-5.
- STANCH1 – This program is used to perform the local stress evaluation of integral welded attachment (round hollow on straight pipe) per Code Case N-392-3.
- ELBUG1 – This program is used to perform the local stress evaluation of integral welded attachment (rectangular lug on curved pipe).
- ELBSTAN1S – This program is used to perform the local stress evaluation of integral welded attachment (round hollow on elbow pipe).

For BOP Piping, Component and Support Systems that required detailed evaluations to reconcile EPU operating parameters, a summary of revised stress levels is provided in Table 2.2.2.2-1. The results presented include existing stress levels (i.e., pre-EPU), revised pipe stress levels for EPU conditions, allowable stress for the applicable loading condition, and the resulting design margin for each piping analysis that was evaluated to reconcile EPU conditions. The design margin provided is based on the ratio of the calculated stress divided by the allowable stress.

Other evaluations of issues that potentially impact BOP Piping, Components and Supports are addressed in the following CR-3 EPU Technical Report Sections.

- Protection against dynamic effects of pipe whip and discharging fluids is discussed in Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and in Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.

- Protection against internally generated missiles and turbine missiles is discussed in Section 2.5.1.2.1, Internally Generated Missiles and Section 2.5.1.2.2, Turbine Generator, respectively.

The implementation of EPU will result in higher flow rates for several piping systems. Piping systems experiencing these higher flow rates will be reviewed for potential vibration issues. Potentially affected piping will be included as part of the startup testing program related to the overall implementation of EPU. Refer to Section 2.12.1, Approach to EPU Power Level and Test Plan, for discussion of the vibration monitoring program.

<u>Results</u>

The results of the piping evaluations for those BOP Systems affected by the changes to the pressure, temperature, and flow rate operating parameters are shown in Table 2.2.2.2-1, Stress Summary at EPU Conditions. This Table provides a summary of existing stress levels (i.e., pre-EPU), revised pipe stress levels for EPU conditions, and the resulting design margin for each piping analysis that required detailed evaluation to reconcile EPU conditions. Piping systems not specifically listed in Table 2.2.2.2-1 were evaluated, but did not require detailed evaluation (i.e., no significant operating parameter increases due to EPU) to reconcile EPU conditions. The stress results reported in Table 2.2.2.2-1 have incorporated thermal expansion and fluid transient increases, as applicable, that were reconciled as part of the EPU evaluations. Table 2.2.2.2-2, Pipe Support Summary at EPU Conditions, provides a summary of those supports that require modification due to EPU conditions. As stated in Appendix E, Major Plant Modifications, support modifications can be performed under the 10 CFR 50.59 process.

The piping stress evaluations conclude that piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure, and flow rate effects resulting from EPU conditions, although pipe support modifications and additions are required to accommodate the revised loadings due to EPU.

For the Main Steam Header (Loops A-1, A-2, B-1, and B-2) and the steam side Emergency Feedwater (EFW) piping lines to the steam driven turbine Emergency Feedwater Pump, including overlap piping, the piping system can withstand the steam hammer loads associated with EPU conditions (resulting from a turbine stop valve closure event). The results of the pipe support evaluations, for these lines that are impacted by EPU, conclude that certain pipe supports require modifications to accommodate the revised loadings (See Table 2.2.2.2-2). The pipe support modifications generally involve upgrading/strengthening existing components, such as, increasing the size of existing welds, installing higher capacity struts/snubbers, and adding steel frame members. CR-3 expects to perform additional refined analyses in accordance with the design and licensing basis which may reduce the scope of these Main Steam and EFW System pipe support modifications.

The evaluations of the branch lines off of the Main Steam Header (two 10-inch/12-inch Turbine Bypass lines, the four 6-inch lines to Moisture Separators, the 3-inch/4-inch branch piping to the Gland Steam Regulator, the 5-inch Auxiliary Steam line to the Desuperheater and the 5-inch Auxiliary Steam line to the Deaerator, all were shown to be acceptable for the revised loadings due to EPU including the increased flow conditions due to the steam hammer loadings.

Previous evaluations of the Feedwater System did not require the evaluation of flow induced transients as there are no existing fast acting valves or equipment that would induce a transient (waterhammer) loading into the system. Feedwater pump trips do not generate water hammer type transient loadings into the system due to the slow run down time (5 to 7 seconds) of the pumps. Modifications to the Feedwater System in conjunction with the new EPU conditions do not add fast acting valves or equipment that would require new evaluations for any type of transient loads. Modifications are being completed to the Feedwater Booster Pumps and Condensate System for EPU. As a part of the CR-3 Engineering Change (EC) process, these changes will properly evaluate the design to ensure new flow transients will be mitigated.

The Emergency Feedwater System flow is increasing for EPU conditions and contains fast acting valves. The flow rates used in the previous evaluations, preformed for the waterhammer loads associated with fast closure of these valves, envelopes the new EPU flow rates.

The flow increase associated with portions of the Heater Drain System was also evaluated for water hammer loads. The increased transient loads associated with the closure of the fast acting valves on this system were found to be acceptable.

Operating pressure increases due to EPU were minor. The impact on the piping system was evaluated for increases in pressure and no piping modifications were required.

Some of the piping systems underwent plant modifications to support EPU conditions during Refueling Outage R16 (e.g., valve changes for better flow control, new heat exchangers, and pump upgrades). For these changes, the piping and support design and evaluations were performed to address the effects of pressure, temperature, and flow transients (i.e., water hammer) as part of the EC package associated with the specific plant modification. As part of the modification process, 10 CFR 50.59 screens/evaluations were performed and the design / analyses for piping systems were developed to be consistent with the CR-3 design basis (FSAR Section 5.4.4) including flow transients. The resulting new piping and support configurations are suitable for their application.

The additional modifications to support EPU conditions will be performed during R17. For these changes, the piping and support design and analysis considers the effects of pressure, temperature, and flow transients (i.e., water hammer) as part of the engineering change package associated with the specific plant modification. The associated design and analyses will be consistent with the CR-3 design basis (FSAR Section 5.4.4) including flow transients resulting in new piping and support configurations suitable for their application.

In summary, the BOP Piping, Components and Support Systems, with modifications outlined in Table 2.2.2.2-2, will continue to meet the CR-3 current design basis for EPU conditions.

2.2.2.2.3 Conclusion

See Section 2.2.2, Pressure-Retaining Components and Component Supports, for the Conclusion.

2.2.2.2.4 References

1. EPRI TR-106438 2856-03, "Water Hammer Handbook for Nuclear Plant Engineers and Operators," Final Report, May 1996.

Table 2.2.2.2-1 Stress Summary at EPU Conditions

Piping Analysis Description	Loading Condition	Existing Stress (psi) (Note 1A)	EPU Stress (psi) (Note 1B)	Allowable Stress (psi)	EPU Stress Ratio (Note 2)
Main Steam Inside Containment Loop A-1	Occasional (with OBE)	17,667	13,690(*)	18,000	0.76
	Occasional (with SSE)	22,140	17,773(*)	27,000	0.66
Main Steam Inside Containment Loop A-2	Occasional (with OBE)	12,210	10,109(*)	18,000	0.56
	Occasional (with SSE)	13,110	13,409(*)	27,000	0.50
Main Steam Inside Containment Loop B-1	Occasional (with OBE)	17,434	14,362(*)	18,000	0.80
	Occasional (with SSE)	22,432	18,366(*)	27,000	0.68
Main Steam Inside Containment Loop B-2	Occasional (with OBE)	13,255	12,219(*)	18,000	0.68
	Occasional (with SSE)	16,156	12,955(*)	27,000	0.48
Main Steam Outside Containment Loop A-1 (Containment Penetration 106 to Turbine Stop Valve MSV-8 Nozzle Connection)	Occasional (with OBE) (Penetration 106 to Isolation Valve MSV-412	9,506	14,420	18,000	0.80
	Occasional (with OBE) (Isolation Valve MSV-412 to Turbine Stop Valve MSV-8	7,796	10,474	18,000	0.58
	Occasional (with SSE) (Penetration 106 to Isolation Valve MSV-412	15,027	18,426	27,000	0.68

Table 2.2.2.2-1 Stress Summary at EPU Conditions

Piping Analysis Description	Loading Condition	Existing Stress (psi) (Note 1A)	EPU Stress (psi) (Note 1B)	Allowable Stress (psi)	EPU Stress Ratio (Note 2)
	Occasional (with SSE) (Isolation Valve MSV-412 to Turbine Stop Valve MSV-8	10,982	14,341	27,000	0.53
Main Steam Outside Containment Loop A-2 (Containment Penetration 105 to Turbine Stop Valve MSV-5 Nozzle Connection)	Occasional (with OBE) (Penetration 105 to Isolation Valve MSV-411	8,955	17,165(*)	18,000	0.95
	Occasional (with OBE) (Isolation Valve MSV-411 to Turbine Stop Valve MSV-5	7,076	14,788(*)	18,000	0.82
	Occasional (with SSE) (Penetration 105 to Isolation Valve MSV-411	13,429	19,996(*)	27,000	0.74
	Occasional (with SSE) (Isolation Valve MSV-411 to Turbine Stop Valve MSV-5	8,954	17,178(*)	27,000	0.64
Main Steam Outside Containment Loop B-1 (Containment Penetration 201	Occasional (with OBE)(Penetration 201 to Isolation Valve MSV-413	8,616	9,352(*)	18,000	0.52

Table 2.2.2.2-1 Stress Summary at EPU Conditions

Piping Analysis Description	Loading Condition	Existing Stress (psi) (Note 1A)	EPU Stress (psi) (Note 1B)	Allowable Stress (psi)	EPU Stress Ratio (Note 2)
to Turbine Stop Valve MSV-7 Nozzle Connection)	Occasional (with OBE) (Isolation Valve MSV-413 to Turbine Stop Valve MSV-7	15,370	10,775(*)	18,000	0.60
	Occasional (with SSE) (Penetration 201 to Isolation Valve MSV-413	13,048	19,472(*)	27,000	0.72
	Occasional (with SSE) (Isolation Valve MSV-413 to Turbine Stop Valve MSV-7	26,193	14,334(*)	27,000	0.53
Main Steam Outside Containment Loop B-2 (Containment Penetration 107 to Turbine Stop Valve MSV-6 Nozzle Connection)	Occasional (with OBE) (Penetration 107 to Isolation Valve MSV-414	9,102	14,164(*)	18,000	0.79
	Occasional (with OBE) (Isolation Valve MSV-414 to Turbine Stop Valve MSV-6	9,370	10,351(*)	18,000	0.58
	Occasional (with SSE) (Penetration 107 to Isolation Valve MSV-414	14,241	17,885(*)	27,000	0.66

Table 2.2.2.2-1 Stress Summary at EPU Conditions

Piping Analysis Description	Loading Condition	Existing Stress (psi) (Note 1A)	EPU Stress (psi) (Note 1B)	Allowable Stress (psi)	EPU Stress Ratio (Note 2)
	Occasional (with SSE) (Isolation Valve MSV-414 to Turbine Stop Valve MSV-6	16,026	12,529(*)	27,000	0.46
MS Supply for EFTB-1 from MSV-56 to Support MSH-191	Occasional (with OBE)	13,746	14,464(*)	18,000	0.80
MS Supply for EFTB-1 from MSV-56 to Support MSH-191	Occasional (with SSE)	17,908	18,725(*)	27,000	0.69
MS Supply for EFTB-1 from MSV-55 to Support MSH-191 and EFTB-1	Occasional (with OBE)	12,802	12,898(*)	18,000	0.72
MS Supply for EFTB-1 from MSV-55 to Support MSH-191 and EFTB-1	Occasional (with SSE)	16,440	17,406(*)	27,000	0.64
Branch Line from MS Loops A-1 and B-1 to Gland Steam Regulator	Sustained + Resultant Transient Force		11,014	18,000	0.61
Auxiliary Steam Branch Line from Loop A-1 to Desuperheater	Sustained + Resultant Transient Force		15,281	18,000	0.85
Auxiliary Steam Branch Line from Loop B-1 to Deaerator	Sustained + Resultant Transient Force		13,897	18,000	0.77

Table 2.2.2.2-1 Stress Summary at EPU Conditions

Piping Analysis Description	Loading Condition	Existing Stress (psi) (Note 1A)	EPU Stress (psi) (Note 1B)	Allowable Stress (psi)	EPU Stress Ratio (Note 2)
Branch line from Loop B-1 to MSR 3B	Sustained + Resultant Transient Force		12,734	18,000	0.71
Branch line from Loop A-2 to MSR 3D	Sustained + Resultant Transient Force		9,035	18,000	0.50
Branch line from Loop A-1 to MSR 3A	Sustained + Resultant Transient Force		12,278	18,000	0.68
Branch line from Loop B-2 to MSR 3C	Sustained + Resultant Transient Force		17,416	18,000	0.97
Turbine Bypass Valves MSV-9/10 to Condenser	Sustained + Resultant Transient Force		10,723	18,000	0.61
Turbine Bypass Valves MSV-11/14 to Condenser	Sustained + Resultant Transient Force		9,110	18,000	0.51

NOTES:

(*) Reflects EPU conditions with support modifications as reported in Table 2.2.2.2-2

1. A) The existing stress outside containment does not include steam hammer stress
   B) The EPU Stress includes steam hammer stress loads

2. EPU Stress Ratio is based on the ratio of EPU stress divided by the allowable stress

Table 2.2.2.2-2 Pipe Support Summary at EPU Conditions

Piping Analysis Description	Support Number (MSH-___)	Required Modification
Main Steam Outside Containment Loop A-2	125	Replace snubber and associated components with higher rated components.
	128	Replace snubber and associated components with higher rated components.
	218	Rod and associated components experience uplift: add weld to support.[1]
	239	Replace snubber and associated components with higher rated components.
Main Steam Outside Containment Loop B-2	27B	Replace rod and associated components with higher rated components. Use a strut to support two-way loads.
	28	Replace rod and associated components with a strut.
	117	Increase size of base plate and anchor bolts.
	119	Replace snubber and associated components with higher rated components.
	121	Replace snubber and associated components with higher rated components.
	124	Replace snubber and associated components with higher rated components.
Main Steam Outside Containment Loop B-1	13B	Replace rod and associated components with a strut.
	122A	Replace snubber and associated components with higher rated components.
	215	Rod and associated components experience uplift: add weld to support.[1]

Table 2.2.2.2-2 Pipe Support Summary at EPU Conditions

Piping Analysis Description	Support Number (MSH-___)	Required Modification
	225	Replace snubber and associated components with higher rated components.
	226-1 & 226-2	Replace snubber and associated components with higher rated components.
Main Steam Inside Containment Loop A-2	164	Replace snubber and associated components with higher rated components.
Main Steam Inside Containment Loop A-1	152	Remove Support.
	157	Replace Constant Spring Hanger with one that allows more travel.
Main Steam Inside Containment Loop B-2	141	Redesign with stronger components. Potential exists to remove the support.
	147	Replace snubber and associated components with higher rated components.
	150	Replace snubber and associated components with higher rated components.
	144	Replace Constant Spring Hanger with one that allows more travel.
	145	Replace Constant Spring Hanger with a higher range and lengthen pin-to-pin distance to reduce excessive swing angle.
Main Steam Inside Containment Loop B-1	136	Remove Support.
	140	Increase pin-to-pin distance to reduce excessive swing angle on this constant spring hanger.
MS Supply for EFTB-1 from	179	Adjust spring cold load setting
	181	Replace rod and associated components with a strut.

Table 2.2.2.2-2 Pipe Support Summary at EPU Conditions

Piping Analysis Description	Support Number (MSH-___)	Required Modification
MSV-56 to Support MSH-191	182	Redesign support to restrain uplift
	183	Replace rod and associated components with a strut.
	185	Adjust spring cold load setting
	186	Replace Constant Spring Hanger with a higher range and lengthen pin-to-pin distance to reduce swing angle.
	190	Redesign support for higher load capacity.
	191	Redesign support for higher load capacity.

Notes.

1. This support is a rod type that is typically not used to carry compressive (uplift) forces, however, for the evaluation of steam hammer, uplift forces were identified. This is acceptable under CR-3 Piping Analysis design guidelines

2.2.2.3 Reactor Vessel and Supports

2.2.2.3.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation

2.2.2.3.2 Technical Evaluation

Introduction

The CR-3 reactor vessel (RV) is described in the FSAR, Section 4.2.2.1. The RV, as the principal component of the reactor coolant system (RCS), contains the heat-generating core and associated supports, controls, and instrumentation, and coolant circulating channels. Primary outlet and inlet nozzles provide for the exit of heated coolant and its return to the RV for recirculation through the core. The CR-3 RV consists of a cylindrical shell, a cylindrical support skirt, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. All coolant inlet and outlet, core flooding, and control rod drive mechanism (CRDM) nozzles are located above the elevation of the top of the core. The design of the RV is in accordance with the requirements of the 1965 Edition, with Addenda through Summer 1967, of the ASME Boiler and Pressure Vessel Code, Section III for Class A Vessels.

The EPU review of the RV and supports only considered in detail the RV support skirt and primary coolant inlet and outlet nozzles. These are the limiting structural components of the RV and are the components which carry the majority of the design loads experienced by the RV due to the EPU. Other RV components such as the heads, shell and host of small nozzles are only impacted by the EPU to the extent by which they are affected by a change in thermal transients of the RCS.

Description of Analyses and Evaluations

The RCS operating parameters which could potentially impact the structural integrity of the reactor vessel and support skirt are the RCS temperature, pressure and flow rate. As stated in Section 1, the RCS operating pressure will not change for the EPU and Table 1.1-1 shows that for EPU, T_{HOT} will increase from 602.1°F to a maximum of 608.7°F, Tcold will decrease from 555.9°F to 555.4°F and the RCS flow rate will be essentially unchanging. Furthermore, the nuclear steam supply system (NSSS) design transient parameters discussed in Section 2.2.2.8, NSSS Design Transients, were considered in the EPU evaluations. The rate of change of the RCS temperature per unit time are unchanging for the EPU (i.e. the overall shape of the transient curves remains the same for the EPU). Also, peak temperatures during transients are changing negligibly such that there is no credible increase in thermal expansion loading or decrease in Code stress allowables. Lastly, as stated in Section 2.2.2.8, NSSS Design Transients, no changes have been made to the allowable number of design cycles defined for each transient. Thus, there are no changes to the NSSS transients that would negatively affect the RV or support skirt. Thus, the RCS operating parameters which could potentially impact the RV and support skirt are essentially unchanging and, therefore, the RV (including all of its subparts such as the shell, bottom head, closure head, CRDM nozzles, core flood nozzle, incore instrumentation nozzles, etc.) and support skirt are not impacted by the EPU.

As discussed in Section 2.2.2, Pressure-Retaining Components and Component Supports, the current design basis structural analysis was performed for the RCS loop due to replacement of the once-through steam generators (OTSGs), but also considered the proposed EPU conditions. This analysis employed the same methodology as used in the pre-OTSG replacement (pre-EPU) structural analyses. Loads for the RV nozzles and support skirt were calculated for the EPU considering deadweight, thermal, seismic and pipe rupture effects. A set of loads was determined for the RV nozzles and support skirt based on the loads previously analyzed and qualified prior to the steam generator replacement (referred to in the Table as “design” loads). The RV nozzles and support skirt were then qualified for EPU using the load comparison method. That is, the EPU loads for each nozzle/support were compared to the corresponding design loads. Where the EPU loads are less than the design loads, the current design basis stress analysis remains applicable.

RV Support skirt

The only support for the RV is the cylindrical support skirt. The following loading combination cases from the current design basis analysis for the RV support skirt were considered:

- Case 1: Deadweight (DW), Thermal (TH), and Operating Basis Earthquake (OBE)
- Case 2: DW, TH, and Safe-Shutdown Earthquake (SSE)
- Case 3: DW, SSE, and LOCA

Loads on the support skirt were calculated considering the EPU conditions and compared to the design loads (i.e., the loads qualified prior to OTSG replacement) for the above loading combination cases.

RV Inlet and Outlet Nozzles

A similar loading comparison was performed for the RV inlet and outlet nozzles. The following loading combination cases from the current design basis analysis for the RV inlet and outlet nozzles were considered:

- Case I: DW and OBE
- Case 2: Deadweight (DW), TH, and OBE
- Case 3: DW and SSE
- Case 4: DW, TH, and SSE
- LOCA
- $(DW) + (SSE^2 + LOCA^2)^{1/2}$

Loads were calculated at the nozzle ends considering the EPU conditions and compared to the design loads for the above loading combination cases.

Results

RV Support skirt

Tables 2.2.2.3-1 (forces) and 2.2.2.3-2 (moments) present the comparison of the RV support skirt forces and moments calculated for the EPU against the design forces and moments. As seen in Tables 2.2.2.3-1 and 2.2.2.3-2, the RV support skirt forces/moments for the EPU are less than the design forces/moments. Thus, the support skirt remains acceptable following the EPU.

RV Inlet Nozzles

Tables 2.2.2.3-3 through 2.2.2.3-6 presents the comparison of the RV inlet nozzle moments calculated for the EPU to the design moments. Only the resultant moments at the nozzle ends are compared since stresses for the nozzle are governed by the loadings and section modulus at the end of the nozzle. For all four inlet nozzles the EPU moments are less than the design moments and, therefore, acceptable for the EPU.

RV Outlet Nozzles

Tables 2.2.2.3-7 through 2.2.2.3-8 presents the comparison of the RV outlet nozzle moments calculated for the EPU to the design moments. Only the resultant moments at the nozzle ends are compared since stresses for the nozzle are governed by the loadings and section modulus at the end of the nozzle. For both outlet nozzles the EPU moments are less than the design moments and, therefore, acceptable for EPU.

As shown in the tables below, the loads calculated for the EPU conditions remain bounded by the loads used in the current design basis analysis to calculate stress and fatigue for the pre-EPU conditions. Therefore, the RV support skirt and all inlet/outlet nozzles continue to meet their stress and fatigue acceptance criteria and are acceptable for the EPU.

2.2.2.3.3 Conclusion

See Section 2.2.2 for Conclusion.

2.2.2.3.4 References

None

Table 2.2.2.3-1: RV Support Skirt—Comparison of Forces

Load Combination	Horizontal Force			Vertical Force		
	Design (kips)	EPU (kips)	Ratio	Design (kips)	EPU (kips)	Ratio
Case 1 (DW+TH)+OBE	225.7	162.76	0.72	2182.6	2152.63	0.99
Case 2 (DW+TH)+SSE	394.7	261.68	0.66	2282.2	2232.80	0.98
Case 3 (DW) + $(SSE^2 + LOCA^2)^{1/2}$	5210.0	2086.55	0.40	6542.0	3350.64	0.51

Table 2.2.2.3-2: RV Support Skirt—Comparison of Moments

Load Combination	Torsional Moment			Bending Moment		
	Design (ft-kips)	EPU (ft-kips)	Ratio	Design (ft-kips)	EPU (ft-kips)	Ratio
Case 1 (DW+TH)+OBE	3370.8	1683.81	0.50	4840.1	4539.85	0.94
Case 2 (DW+TH)+SSE	4399.6	2729.86	0.62	9384.8	7149.79	0.76
Case 3 (DW) + $(SSE^2 + LOCA^2)^{1/2}$	57417.0	3275.57	0.06	119167.0	61193.57	0.51

Table 2.2.2.3-3: Northeast RV Inlet Nozzle –Comparison of Moments

Load Case	Design Moment (ft-kips)	EPU Moment (ft-kips)	Ratio
Case 1 (DW+OBE)	824.5	683.7	0.83
Case 2 (DW+TH+OBE)	1426.3	1309.0	0.92
Case 3 (DW+SSE)	1047.1	785.1	0.75
Case 4 (DW+TH+SSE)	1691.1	1410.5	0.83
LOCA	1874.6	1259.3	0.67
DW + SRSS (SSE, LOCA)	2147.2	1842.0	0.86

Note: RV Inlet Nozzles labeled Northeast, Northwest, Southeast or Southwest with respect to Plant North with the pressurizer connected to the Northwest loop.

Table 2.2.2.3-4: Northwest RV Inlet Nozzle–Comparison of Moments

Load Case	Design Moment (ft-kips)	EPU Moment (ft-kips)	Ratio
Case 1 (DW+OBE)	847.5	538.5	0.64
Case 2 (DW+TH+OBE)	1530.7	1497.1	0.98
Case 3 (DW+SSE)	1001.0	631.8	0.63
Case 4 (DW+TH+SSE)	1683.8	1590.4	0.94
LOCA	1874.6	1259.3	0.67
DW + SRSS (SSE, LOCA)	2125.1	1705.6	0.80

Table 2.2.2.3-5: Southwest RV Inlet Nozzle–Comparison of Moments

Load Case	Design Moment (ft-kips)	EPU Moment (ft-kips)	Ratio
Case 1 (DW+OBE)	824.5	689.6	0.84
Case 2 (DW+TH+OBE)	1426.3	1296.9	0.91
Case 3 (DW+SSE)	1047.1	797.2	0.76
Case 4 (DW+TH+SSE)	1691.1	1404.5	0.83
LOCA	1874.6	1259.3	0.67
DW + SRSS (SSE, LOCA)	2147.2	1848.2	0.86

Table 2.2.2.3-6: Southeast Inlet Nozzle–Comparison of Moments

Load Case	Design Moment (ft-kips)	EPU Moment (ft-kips)	Ratio
Case 1 (DW+OBE)	847.5	472.0	0.56
Case 2 (DW+TH+OBE)	1530.7	1344.0	0.88
Case 3 (DW+SSE)	1001.0	593.7	0.59
Case 4 (DW+TH+SSE)	1683.8	1465.7	0.87
LOCA	1874.6	1259.3	0.67
DW + SRSS (SSE, LOCA)	2125.1	1616.7	0.76

Table 2.2.2.3-7: North RV Outlet Nozzle–Comparison of Moments

Load Case	Design Moment (ft-kips)	EPU Moment (ft-kips)	Ratio
Case 1 (DW+OBE)	980.9	884.4	0.90
Case 2 (DW+TH+OBE)	1830.6	1558.4	0.85
Case 3 (DW+SSE)	1742.5	1097.9	0.63
Case 4 (DW+TH+SSE)	2556.1	1771.8	0.69
LOCA	4116.2	3309.9	0.80
DW + SRSS (SSE, LOCA)	4469.8	3993.9	0.89

Note: RV Outlet Nozzles labeled North or South with respect to Plant North with the pressurizer connected to the Northwest loop.

Table 2.2.2.3-8: South RV Outlet Nozzle—Comparison of Moments

Load Case	Design Moment (ft-kips)	EPU Moment (ft-kips)	Ratio
Case 1 (DW+OBE)	980.9	820.3	0.84
Case 2 (DW+TH+OBE)	1830.6	1487.0	0.81
Case 3 (DW+SSE)	1742.5	1019.8	0.59
Case 4 (DW+TH+SSE)	2556.1	1686.5	0.66
LOCA	4116.2	1863.4	0.45
DW + SRSS (SSE, LOCA)	4469.8	2513.9	0.56

2.2.2.4 Control Rod Drive Mechanism and Supports

2.2.2.4.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation.

2.2.2.4.2 Technical Evaluation

Introduction

The Control Rod Drive Mechanism (CRDM) positions the control rod within the reactor core and indicates the location of the control rod with respect to the reactor core. The CRDM nozzles are installed on the reactor vessel head with partial penetration welds. The CRDM consists of the motor tube housing, lead screw, rotor assembly, buffer and a vent cap assembly. The position indicator is a separate structure outside the motor tube. Main coolant fills the pressure containing parts of the drive mechanism. Thus, the pressure vessel component of the CRDM assembly constitutes a portion of the reactor coolant pressure boundary. The pressure boundary of the CRDMs and all the components of the control rod drive system are designed as Seismic Category I equipment. The CRDMs are supported by the nozzles in the reactor vessel head and prevented from experiencing excessive lateral deflection due to horizontal loading by a lateral support.

The CRDMs are described in the FSAR sections 3.1.2.4.3 and 3.2.4.3

Description of Analyses and Evaluations

Each control rod drive assembly is sealed with a flange and gasket connecting the CRDM nozzle to the CRDM housing to prevent leakage of reactor coolant water. The pressure-containing components of the Type ‘C’ CRDMs are designed to meet the requirements of the ASME B&PV Code, Section III for Class A Vessel appurtenances 1965 Edition, with Addenda through Summer 1967. As described in the FSAR, the CRDMs are required to be capable of withstanding the seismic loadings within the stress limits for seismic Class I equipment.

The current design basis stress analysis for the Type ‘C’ CRDMs specifies an operating pressure and temperature of 2185 psig and 608°F, respectively. The CRDMs were evaluated for the EPU using the nuclear steam supply system (NSSS) operating parameters of Section 1.1, Nuclear Steam Supply System Parameters, Table 1.1-1, and the NSSS design transients of Section 2.2.2.8, NSSS Design Transients. As stated in Section 1.0, the Reactor Coolant System (RCS) operating pressure will not change as a result of the EPU. Additionally, the maximum EPU T_{HOT} value is given in Table 1.1-1 as 608.7°F. Thus, the operating temperature to which the CRDMs are qualified in the current design basis stress analysis is a maximum of 0.7°F less than the EPU temperature. The minimal increase in temperature results in a negligible change in thermal expansion. However, the CRDMs are free to thermally expand in the vertical direction and, as such there is no increase in thermal stress resulting from the EPU. Further, this small increase in temperature does not significantly affect the ASME Code stress allowables to which the CRDMs are qualified. Thus, there are no changes to the EPU operating conditions that would negatively impact the qualification of the CRDMs.

In addition to operating conditions, the changes in NSSS design transients (discussed in Section 2.2.2.8, NSSS Design Transients) are also evaluated for the EPU since CRDM stress/fatigue can be affected by changes to any of the following characteristics of design basis transients: (1) rate of temperature change vs. time, (2) peak temperature or (3) allowable number of design cycles. Review of the NSSS design transients confirmed the rate of change of the RCS temperature for the various transients are unchanging for the EPU (i.e. the overall shape of the transient curves remains the same for the EPU). Also, peak temperatures are changing negligibly such that there is no increase in thermal expansion loading or Code stress allowables. Lastly, as stated in Section 2.2.2.8, no changes have been made to the allowable number of design cycles defined for each transient. Thus there are no changes to the NSSS transients that would increase stress or fatigue in the CRDMs or supports.

Regarding the CRDM supports, besides the CRDM nozzles (which are addressed in Section 2.2.2.3, Reactor Vessel and Supports), the only CRDM support is a lateral support on the reactor vessel service structure which prevents excessive lateral deflection due to horizontal loading. The proposed EPU will not introduce any increase in horizontal loadings since (1) seismic loading is unaffected by the EPU and (2) the CRDM lateral supports are qualified for large bore pipe rupture loadings (breaks in the primary piping). Large break loads are considered in the design basis analysis of the CRDM support. However, due to implementation of leak-before-break (LBB), the design break for the CRDMs became a 14" diameter Core Flood Line Break as opposed to a 28" or 36" diameter primary pipe break. The stress margin gained from reducing the pipe break size far exceeds the stress margin lost due to an increase in LOCA loadings due to the EPU. Therefore, the CRDM supports will continue to meet their acceptance criteria following the EPU.

The effects of flow induced vibration were considered for the CRDMs and supports. Based on a negligible amount of RCS flow in the upper head region of the reactor vessel, there are no FIV concerns for the CRDMs.

Results

Since the normal operating pressure is unchanging for the EPU and the increase in thermal stress due to exceeding the currently qualified operating temperature of 608°F by 0.7°F is negligible, the CRDMs will continue to meet their Code acceptance criteria for the new operating conditions due to the EPU. Additionally, because there are no changes to the NSSS transients that would increase stress or fatigue of the CRDMs nor any changes to the allowable number of design cycles that would impact the CRDMs, the previously analyzed transients remain applicable for the EPU. Therefore, no new stress or fatigue analyses are required for the CRDMs due to the EPU and the existing analyses remain acceptable.

Similarly, no new analysis is required for the CRDM lateral supports since there will be no increase in CRDM horizontal loadings as a result of the EPU. As such, the existing analyses remain acceptable.

2.2.2.4.3 Conclusion

See Section 2.2.2 for Conclusion.

2.2.2.4.4 References

None.

2.2.2.5 Steam Generators and Supports

2.2.2.5.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation

2.2.2.5.2 Technical Evaluation

Introduction

The CR-3 steam generators, supplied by B&W Canada, were installed during the Fall 2009 outage. The steam generators have been evaluated for operation at EPU conditions specified in Section 1.1, Nuclear Steam Supply System Parameters, Table 1.1-1. The steam generators are described in the FSAR Sections 4.2.2.2 and 4.3.4. Evaluation of the steam generators has demonstrated continued compliance with applicable regulatory and industry structural integrity and thermal-hydraulic performance requirements following the implementation of EPU. The steam generators were also evaluated to demonstrate that failure due to tube vibration and wear would not occur. The evaluations considered an EPU full-power core thermal power level of 3014 MWt (nuclear steam supply system power level of 3030 MWt) and steam generator tube plugging (SGTP) over the range from 0 to 5%, including design transients.

Description of Analyses and Evaluations

Various computer codes have been used for EPU qualification of the Once Through Steam Generators (OTSGs) (refer to Appendix B). All computer codes used to perform any new or re-analysis for EPU have been Commercial Grade Dedicated by B&W in compliance with its Quality Assurance Program for Nuclear Products, which meets 10CFR50 Appendix B and 10CFR21 requirements.

The steam generator and supports evaluation was performed in four separate, but coordinated, portions:

- Supports
- Structural Integrity
- Thermal-Hydraulic
- Tube Vibration and Wear

2.2.2.5.2.1 Steam Generator Supports Evaluation

The steam generator supports of the NSSS, as described in FSAR Sections 4.2.6.3 and 4.2.6.6, were evaluated for EPU conditions. The reactor coolant system (RCS) piping loads on the steam generator supports due to the parameters associated with the EPU as discussed in Section 2.2.2.1, NSSS Piping, Components, and Supports, were reviewed for impact on the existing RCS steam generator supports design-basis with the original steam generators installed. The RCS piping loads on the steam generator supports due to deadweight, thermal expansion, design basis earthquake (DBE), and maximum hypothetical earthquake (MHE) loading cases are obtained from the evaluation for the EPU program as described in Section 2.2.2.1, NSSS Piping, Components, and Supports. The loss of coolant accident (LOCA) and the pipe break analyses from the current design basis remain valid for EPU conditions.

The steam generator supports loads are evaluated for EPU conditions based on the design loads data from the current design basis, and the steam generator support loads obtained from the evaluation for the RCS piping analyses as described in Section 2.2.2.1, NSSS Piping, Components, and Supports.

<u>Input Parameters, Assumptions, and Acceptance Criteria</u>

The acceptance criteria for the RCS piping steam generator supports, as indicated in the FSAR Section 4.2.6.3, are based upon the Level D loading identified in the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code, Section III, Subsection NF. The steam generator base supports are designed to the 1998 Edition through 2000 Addenda and the upper lateral supports are designed to the 1989 edition, no Addenda.

The steam generator support loads from the RCS piping analyses as described in Section 2.2.2.1, NSSS Piping, Components, and Supports, and the current design basis steam generator support loads are used to calculate the design loads available for the EPU program for the steam generator supports. The design basis for the RCS piping analysis was updated to consider the EPU design parameters. The steam generator upper lateral supports and the steam generator lower base support are evaluated for the loading values for the EPU program.

<u>Results</u>

The load and design values are summarized and are tabulated in Table 2.2.2.5.2.1-1 for the loading combinations, as specified in the acceptance criteria in the Code of Record, FSAR Section 4.2.6.3, and as evaluated in the current design basis.

2.2.2.5.2.2 Steam Generator Structural Integrity Evaluation

The structural integrity evaluation of the steam generator was performed for EPU service loading conditions. The stresses, stress intensity ranges, and fatigue usage factors in the steam generators for the EPU conditions were determined by reconciling the original design basis analyses for the non-EPU condition against the EPU conditions. Results of the reconciliation are presented below.

The acceptance criteria for the EPU conditions were based on demonstrating continued compliance with the structural criteria in the ASME B&PV Code Section III, Class 1, Subsections NB and NF. These acceptance criteria are the same as those used for the original design basis analyses of the steam generators.

The internal components, which are not part of the pressure boundary, are not governed by the ASME B&PV Code. However, the ASME B&PV Code, Section III, Class 1, Subsections NB and NF were adopted as guidelines for performing the structural analysis of these components.

The scope of the reconciliation was the entire steam generator pressure boundary:

- internal pressure boundary attachments (shroud lug and shroud ring supports);
- external pressure boundary attachments (lifting trunnion and threaded hole for upper lateral support structure);
- lower base support; and
- all internal components.

It was determined that the steam generator pressure boundary and internal components continue to remain in compliance with the structural criteria of the ASME B&PV Code, Section III, Class 1, Subsections NB and NF and, therefore, are structurally adequate for operation at the EPU conditions.

<u>Input Parameters and Acceptance Criteria</u>

The structural evaluation was performed for EPU conditions identified in Table 1.1-1 with 35°F of superheat at full load conditions. This superheat condition will be maintained up to 5% tube plugging. The design loads and transients for Steam Generator Structural Integrity Evaluation are shown in Table 2.2.2.5.2.2-1, Structural Integrity Loads and Transients.

There were no changes to the external nozzle and attachment loads for EPU conditions, except for the inlet nozzle thermal loads. However, it was demonstrated that the existing inlet nozzle thermal loads are bounding. Thus, the original design basis external loads were used in the EPU reconciliation analyses. The seismic loads on the steam generators and internal components for EPU conditions establish the basis for qualification. The tube to shell temperature differences, steam flow loads on tubes, and pressure drop across the tube support broached plates were reanalyzed for EPU conditions. Large Break Loss of Coolant Accident (LBLOCA) and Main Steam Line Break (MSLB) under EPU conditions were shown to be bounded by existing pre-EPU conditions. The bolt preloads for bolted pressure boundary openings were established on the basis of leak proofing the joints, and thus remain valid for EPU conditions. All EPU transients identified in the Table 2.2.2.5.2.2-1, Structural Integrity Loads and Transients, were used in the reanalysis of all pressure boundary components including tubes and supports to demonstrate their acceptability for EPU conditions.

Continued compliance with the current steam generator design basis analysis is the acceptance criteria for the structural analysis for the EPU conditions. For the structural evaluation of the pressure boundary components, the acceptance criteria from ASME B&PV Code, Section III, Class 1, Subsections NB and NF components continued to remain applicable. Excessive plastic deformation is prevented by limits on the acceptable primary stresses. Plastic instability and incremental collapse are prevented by limits on the acceptable primary plus secondary stresses. High-strain, low-cycle fatigue is prevented by limits on the total stresses and their cycles. Satisfaction of these limits demonstrates continued compliance with the current design acceptance criteria and, therefore, the adequacy of the steam generator design for operation at the EPU conditions for the 40-year design life. As stated below, these criteria were met and are shown in the tabulated analysis outputs listed in the Results section that follows.

The steam generator internal components, other than the tubes, are not part of the pressure boundary and, therefore, are not governed by the ASME B&PV Code. However, ASME B&PV Code Section III, Class 1, Subsections NB and NF were adopted as guidelines for performing the structural analysis of these components. These components were reviewed and it was determined that they satisfy the ASME B&PV Code requirements for components not requiring an analysis for cyclic operation. As a result, a fatigue analysis was not performed for the internals. The tube support broached plates and supporting tie rods, however, were analyzed for fatigue since they are the most highly loaded of all the internal components due to significant loads and cycles associated with potential water slap cleaning operations.

For primary stresses on steam generator pressure boundary components, a reconciliation analysis, based on a detailed comparison of pressure loads, and external nozzle loads where applicable, was completed and demonstrated that the original design basis analyses remain valid for the EPU conditions.

For primary plus secondary stress ranges and fatigue usage factors on steam generator pressure boundary components, new analyses, using the EPU transients and the ANSYS finite element models completed for the original design basis analyses, were completed and demonstrated that stress ranges and fatigue usage factors are still below the appropriate ASME limits for the EPU conditions.

A new tube flaw size analysis in accordance with the requirements of NRC Regulatory Guide (RG) 1.121 and the Electric Power Research Institute (EPRI) Steam Generator Integrity Assessment Guidelines – EPRI Final Report 1012987 was completed for the EPU conditions.

The design basis analysis demonstrating protection against non-ductile fracture is unaffected by the EPU since the lower temperature operation steam pressure remained unchanged. As a result, this current design basis analysis remains bounding for EPU conditions.

For steam generator internal components, a reconciliation analysis, based on a detailed comparison of loads, was completed and demonstrated that the original design basis analyses remain valid for the EPU conditions.

Results

The most critical results from the structural evaluation of the steam generator pressure boundary are presented in Tables 2.2.2.5.2.2-1 to 2.2.2.5.2.2-6. A summary of Structural Integrity Loads and Transients is presented in Table 2.2.2.5.2.2-1. A summary of stresses for the Design conditions for Primary Side components and Secondary Side components is provided in Table 2.2.2.5.2.2-2. A summary of the stress range and fatigue results using simplified linear analysis is provided in Table 2.2.2.5.2.2-3 for Primary and Secondary side components. A summary of the stress range and fatigue results using simplified elastic-plastic analysis is provided in Table 2.2.2.5.2.2-4. A simplified elastic-plastic analysis is performed when the results from simplified linear analysis exceeded the limit. Results for Level C and Level D conditions are provided in Tables 2.2.2.5.2.2-5 and 2.2.2.5.2.2-6, respectively

The steam generator internal components are largely unaffected by the EPU and are not governed by the ASME Code. Nevertheless, the results of the internals stress analysis are included in Tables 2.2.2.5.2.2-7 to 2.2.2.5.2.2-9. Table 2.2.2.5.2.2-10 lists the allowable volumetric tube flaw sizes according to RG 1.121 and the EPRI Steam Generator Integrity Assessment Guidelines – EPRI Final Report 1012987.

2.2.2.5.2.3 Steam Generator Thermal-Hydraulic Evaluation

Thermal-hydraulic analyses have been completed for the OTSGs at the EPU conditions with a NSSS power level of 3030 MWt. The analyses determined the steam generator thermal-hydraulic characteristics and inventories, and provided input used to evaluate the potential for tube wear and flow-induced vibration (FIV). The results of this effort show that the steam generators have satisfactory thermal-hydraulic performance for the EPU operating conditions.

Input Parameters, Assumptions, and Acceptance Criteria

The operating conditions for the EPU are provided in Table 1.1-1 and were used in the following analyses. All of the significant thermal-hydraulic input parameters are listed below. Performance was determined for both start-up and end of life (EOL) conditions of the steam generator. Table 2.2.2.5.2.3-1 describes the start-up and EOL scenarios.

The thermal-hydraulic performance evaluation consisted of a steady-state heat balance using B&W Canada's THEDA-2 nuclear code. THEDA-2 is a general purpose program used to simulate the thermal-hydraulic performance of bleed flow type OTSGs. THEDA-2 handles boiling, single and two phase conditions throughout the OTSG. The program uses an efficient multi-grid solution algorithm suitable for both 2-D and 3-D models. There is ample grid refinement which allows detailed flow modeling which is necessary for subsequent flow induced vibration analysis. The THEDA-2 program predicts steam temperature at the bundle outlet, bundle pressure drop and the static pressure at the steam nozzle outlet. The program also predicts the secondary side inventory in the tube bundle, the downcomer, and in the steam annulus region.

THEDA-2 calculated pressures are used to determine the operating range and start-up range water level.

Start-up and EOL cases were considered in the analysis.

The thermal-hydraulic acceptance criteria for EPU conditions are:

- Superheat ≥ 35° F
- Secondary Side Inventory ≤ 68,059 lbm

<u>Results</u>

Table 2.2.2.5.2.3-2 compares thermal-hydraulic attributes at original replacement OTSG conditions (2584 MWt) to EPU conditions. Results show all thermal-hydraulic attributes were demonstrated to be acceptable for operation at the EPU conditions.

The secondary side pressure loss between the feedwater inlet and steam outlet nozzles increases by approximately 16 psi for the EPU conditions. However, this is considered acceptable when compared with the main feedwater (FW) pump capacity. The primary side pressure drop is essentially unchanged, other than coolant density effects, since the primary coolant flow remained unchanged for the EPU conditions. Results show all the thermal-hydraulic parameters were within expected ranges for the EPU for tube plugging up to 5.2% based on thermal design flow. With best estimate RCS flow, the maximum allowable tube plugging increases to 10.9%.

2.2.2.5.2.4 Steam Generator Tube Vibration and Wear Evaluation

FIV analysis and tube wear calculations were performed to demonstrate that the steam generator tubes continue to be adequately supported to prevent detrimental FIV and fretting wear at the EPU conditions. It was demonstrated that the tubes continue to not be susceptible to fluid-elastic instability and that the accumulated tube wear over the life of the steam generators remains acceptable.

<u>Input Parameters, Assumptions and Acceptance Criteria</u>

The assessment of FIV and tube wear was performed for the Beginning of Life and EOL conditions described in the section above. The bounding case scenario corresponds to the EOL conditions at full power (3030 MWt) with 35°F superheat. The EOL conditions effectively bounded the start-up conditions for FIV responses.

The applicable EPU condition acceptance criteria are:

- critical velocity ratio < 1.0 precluding fluid-elastic instability; and
- accumulated tube wear over a 40-year life < 40% nominal tube wall thickness.

FIV of the steam generator tubes at the EPU conditions was performed for the potential FIV mechanisms of fluid-elastic instability, vortex shedding (VS) resonance, and random turbulence (RT) excitation for the worst tube, which is the tube located at the maximum distance from the centerline of the tube bundle, where the gap velocities are highest.

In the tube wear analysis, a work rate was determined from the integral average of the normal contact force multiplied by the sliding distance over the tube-to-support interface for each mode shape. This work rate was then converted into a wear volume and an equivalent wear depth based on wear coefficients for the tube and support materials. The predicted maximum tube wear after a 40 year design life is 38.2% through wall.

Results

For the EPU conditions, the maximum calculated critical velocity ratio for fluid-elastic instability is 0.817 which satisfies the < 1.0 acceptance limit used for assessing FIV.

The maximum expected tube wear at EPU conditions was also assessed. The predicted maximum tube wear after a 40 year design life is 38.2 % through wall.

Result Tables

The design loads available for the EPU program for the steam generator upper lateral supports and the steam generator base support are evaluated and summarized in Table 2.2.2.5.2.1-1 and Table 2.2.2.5.2.2-8 for the EPU program. In all cases, the calculated loads are less than the design allowable values for the loading combinations for EPU. The EPU stress values are either equal to or less than the stress margin values in the current design basis.

The following are used in the Tables within this section:

- Pm = primary membrane stress intensity
- Pb = primary bending stress intensity
- PL = local membrane stress intensity
- σ = direct stress
- $P_{external}$ = external pressure
- τ = shear stress

2.2.2.5.3 Conclusion

See Section 2.2.2 for Conclusion

2.2.2.5.4 References

None

Table 2.2.2.5.2.1-1
Steam Generator Supports Load Cases

Load Case	Normal	Upset	Faulted	Limit
Deadweight Minimum	Deadweight + Thermal	Deadweight + Thermal + DBE	Deadweight + Thermal + $(MHE^2 + HELB^2)^{0.5}$	
Upper Lateral Supports (ULS)	163.71 kips	(See Note 1)	737.86 kips	1000 kips[3]
Base Support Anchor Bolts[2]	N/A	N/A	52.3 ksi	58 ksi

NOTES

1. DBE loads are 1/2 the MHE loads and the upper lateral supports were designed to faulted load case.
2. Base support bolts are not ASME B&PV Code NB/NF components. Their values represent the worst case bolt removal scenarios, recognizing that all current bolts may not be usable.
3. Note that the original ULS connection to wall (rear bracket and anchor bolts) were designed for a minimum load of 3430 kips, which is greater than the maximum faulted load. Therefore, the connection to the building is acceptable for the loads calculated in the structural loading analysis.

Table 2.2.2.5.2.2-1
Structural Integrity Loads and Transients

Loading Conditions	Service Loads/Combinations [1]	ASME Service Stress Limit Level
Design	• Deadweight • Operating Basis Earthquake (OBE) • Design Pressure • Design Temperature • Design Flow • Thermal • Internal Design Mechanical Loads	Design
Normal	• Deadweight • Thermal • Internal Normal Mechanical Loads • Normal Condition Transients [2]	A
Upset	• Deadweight • Thermal • OBE [5] • Internal Upset Mechanical Loads • Upset Condition Transients [2]	B
Emergency [4]	• Deadweight • Thermal • Internal Emergency Mechanical Loads • Emergency Condition Transients [2]	C
Faulted	• Deadweight • Thermal • Safe Shutdown Earthquake (SSE) [3] • Internal Faulted Mechanical Loads • Pipe Rupture Loads [3] • Faulted Condition Transients [2]	D

NOTES

1. Loading responses are combined using the absolute sum method with the exception of those addressed in Notes 3 and 5.
2. Transients are applied one at a time unless otherwise noted in this specification.
3. For faulted condition evaluations, the effects of SSE and pipe rupture are combined using the square root of the sum of the squares (SRSS) method, per NUREG-0484, “Methodology for Combining Dynamic Responses.”

4. This loading condition is not part of the Design Basis Criteria for CR-3, but is included here for completeness relative to the ASME Code definitions only.
5. For fatigue analysis, five (5) OBE events with a minimum of ten (10) maximum stress cycles per event were assumed and stresses due to OBE were combined with the Normal (Level A) and Upset (Level B) transients which produce the maximum positive and negative stresses. For the OBE and SSE conditions, the OTSG was considered at the EPU conditions with the water inventory (mass) corresponding to a 35°F superheated condition. The 10% grouping method was used for the seismic model combination and the absolute sum of horizontal and vertical was used for seismic directional combination.

Table 2.2.2.5.2.2-2
Stress Summary for Design Conditions

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
PRIMARY SIDE COMPONENTS			
Manway Covers	Pm	18.0	26.7
	Pm + Pb	31.6	40.0
Manway Bolting	$\sigma_{average}$	22.1	27.6
Handhole Cover	Pm	14.5	26.7
	Pm + Pb	31.4	40.0
Handhole Bolting	$\sigma_{average}$	22.8	27.6
Inlet Nozzle Within Limit of reinforcement (LOR)	Pm	12.3	30.0
Inlet Nozzle Outside LOR	Pm	16.6	30.0
	Pm (PL) + Pb	16.9	45.0
Inlet Nozzle / Head Juncture	PL	19.5	45.0
Outlet Nozzles Within LOR	Pm	19.9	30.0
Outlet Nozzles Outside LOR	Pm	17.4	30.0
	Pm + Pb	27.2	45.0
Outlet Nozzle / Head Junctures	PL	25.6	45.0
Tubes	Pm	22.2	26.7
	Pm + Pb	22.2	40.0
	$P_{external}$	1.00	1.02
Tube Plug Walls	Pm	19.3	23.3
Tube Plug End Caps	Pm	3.0	23.3
	Pm + Pb	5.3	34.9
Tube to Tubesheet Seal Welds	$\tau_{average}$	10.6	14.0
Tube Plug to Tube Seal Welds	$\tau_{average}$	10.1	14.0
Primary Heads	Pm	13.6	30.0
Head to Tubesheet Junctures	Pm (PL) + Pb	28.3	45.0
Head to Tubesheet Welds	PL	29.4	45.0
Tubesheet to Thick Shell Junctures	Pm (PL) + Pb	27.7	45.0
Thick Shell to Tubesheet Welds	PL	27.8	45.0

Table 2.2.2.5.2.2-2
Stress Summary for Design Conditions (continued)

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
PRIMARY SIDE COMPONENTS (continued)			
Perforated Tubesheets	Pm	16.5	30.0
	Pm + Pb	30.1	45.0
Tubes (Tube / Shell Interaction)	$\sigma_{compressive}$	-3.3	-6.7
Head at Base Support Stool	Pm	26.7	30.0
	Pm (PL) + Pb	42.4	45.0
Lower Head at Flat Section	Pm	15.4	30.0
	Pm + Pb	33.4	45.0
Base Support Stool	Pm	25.5	30.0
	Pm + Pb	32.6	45.0
Base Support Anchor Bolts [1]	$\sigma_{tensile}$	52.3	58.0
Base Support Foundation Concrete [1]	$\sigma_{compressive}$	1.3	2.0
Tubesheet Threaded Holes for ULS [2]	τ_{shear}	28.6	32.3
Upper Head Manway Cover Support Cross-Tee Lugs	Pm	0.6	23.3
	Pm + Pb	24.8	34.9
Lug / Top Plate Dual Fillet Welds	τ_{shear}	16.0	21.0
Top Plate Bolt Holes	τ_{shear}	1.9	14.0
	$\sigma_{bearing}$	1.3	38.0
Lower Head Manway Curtain Rod	Pm	0.6	16.7
	Pm + Pb	10.6	25.1
Lower Head and Base Support Dual Lugs	Pm	1.9	23.3
	Pm + Pb	2.2	34.9
Base Support Step Plate	Pm	8.9	16.7
	Pm + Pb	23.3	25.0
Base Support Gusset Plate to Mounting Rod Full-Pen Welds	Pm	0.8	16.7
	Pm + Pb	14.3	25.0

Table 2.2.2.5.2.2-2
Stress Summary for Design Conditions (continued)

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
SECONDARY SIDE COMPONENTS			
Thin Shells	Pm	27.8	30.0
	Pm + Pb	29.0	45.0
Thick Shells	Pm	16.7	30.0
	Pm + Pb	19.0	45.0
1” Nozzles Within LOR	Pm	13.2	22.2
1” Nozzles Outside LOR	Pm	5.2	22.2
	Pm + Pb	14.8	33.3
1” Nozzle Plug Walls	Pm	10.7	17.6
1” Nozzle Plug End Caps	Pm	3.0	17.6
	Pm + Pb	10.4	26.4
Thin Shells at 1” Nozzles	Pm	27.6	30.0
(bound 1” Nozzles on Thick Shells)	Pm + Pb	27.8	45.0
1 ½” Nozzles Within LOR	Pm	15.7	22.2
1 ½” Nozzles Outside LOR	Pm	6.8	22.2
	Pm + Pb	21.2	33.3
Thick Shells at 1 ½” Nozzles	Pm	16.7	30.0
(bound 1 ½” Nozzles on Tubesheets)	Pm + Pb	16.8	45.0
Manway Openings [3]	P_{Design}	1.15	1.26
Manway Covers	Pm	12.9	26.7
	Pm + Pb	31.3	40.0
Manway Bolting	$\sigma_{average}$	25.8	27.9
Handhole Openings	PL	40.6	45.0
Handhole Covers	Pm	7.9	26.7
	Pm + Pb	16.7	40.0
Handhole Bolting	$\sigma_{average}$	17.7	27.9
Inspection Port Openings [3]	P_{Design}	1.15	1.16

Table 2.2.2.5.2.2-2
Stress Summary for Design Conditions (continued)

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
SECONDARY SIDE COMPONENTS (continued)			
Inspection Port Covers	Pm	7.7	26.7
	Pm + Pb	23.1	40.0
Inspection Port Bolting	$\sigma_{average}$	16.1	27.9
Steam Outlet Nozzles Within LOR	Pm	11.6	30.0
Steam Outlet Nozzles Outside LOR	Pm	19.5	22.2
	Pm + Pb	32.8	33.3
Steam Outlet Nozzle / Shell Junctures	PL	23.8	45.0
Main and Emergency FW Nozzle at Elbows [4]	Pm	8.7	23.7
	Pm + Pb	30.5	35.5
Main and Emergency FW Nozzles Within LOR [4]	Pm	17.9	22.2
Main and Emergency FW Nozzle at Thermal Sleeve Junctures [4]	Pm	2.1	23.7
	Pm + Pb	8.3	35.5
Main and Emergency FW Nozzle Thermal Sleeves [4]	Pm	0.4	23.3
	Pm + Pb	0.4	35.0
Main and Emergency FW Nozzle Bolted Flanges [4]	Pm	6.8	26.7
	Pm + Pb	28.3	40.1
Main and Emergency FW Nozzle Bolts [4]	$\sigma_{average}$	32.2	55.8
	$\sigma_{maximum}$	50.5	83.7
Main and Emergency FW Nozzle Seal Welds	SI	20.4	33.3
Thick Shells at Main and Emergency FW Nozzles [4]	Pm	21.5	30.0
	Pm + Pb	21.5	45.0
Main FW 3” Riser Straight Pipes	$SI_{Equation\_9}$	18.7	26.8
Main FW 3” Riser Bend Pipes	$SI_{Equation\_9}$	20.3	26.8
Main FW 3” Riser Elbows	$SI_{Equation\_9}$	21.7	26.8
Main FW Riser Stub / Header Tees	$SI_{Equation\_9}$	21.9	26.8
Main FW 14” Header Pipes	$SI_{Equation\_9}$	7.3	26.8

Table 2.2.2.5.2.2-2
Stress Summary for Design Conditions (continued)

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
SECONDARY SIDE COMPONENTS (continued)			
Main FW 14” Inlet Tees	$SI_{Equation\_9}$	13.7	26.8
Main FW 14” Inlet Elbows	$SI_{Equation\_9}$	12.6	26.8
Main FW 14” Inlet Extensions	$SI_{Equation\_9}$	10.4	25.7
Emergency FW 3” Riser Pipes	$SI_{Equation\_9}$	11.9	26.8
Emergency FW 3” Riser Elbows	$SI_{Equation\_9}$	16.2	26.8
Emergency FW Riser Stub / Header Tees	$SI_{Equation\_9}$	25.9	26.8
Emergency FW 6” Header Pipes	$SI_{Equation\_9}$	8.2	26.8
Emergency FW 6” Inlet Tee	$SI_{Equation\_9}$	16.9	26.8
Emergency FW 6” Inlet Extension	$SI_{Equation\_9}$	9.3	25.7
Trunnion Pipes	Pm	21.9	23.3
	Pm + Pb	28.8	35.0
	$\sigma_{bearing}$	13.9	40.0
Trunnion Cap Plates	Pm	10.4	23.3
	Pm + Pb	31.4	35.0
Trunnion Cap Screws	$\sigma_{average}$	67.1	75.0
	$\sigma_{maximum}$	84.6	130.0
Thick Shell at Trunnions	PL	16.4	45.0
Lower Shroud Lugs	Pm	1.4	18.5
	Pm + Pb	12.4	27.8
Thick Shell at Lower Shroud Lugs	PL	16.7	45.0
Lower Shroud Shear Lugs [5]	Structurally adequate for seismic fatigue loads		
Thick Shell at Upper Shroud Ring	PL	14.1	45.0
Upper Shroud Ring	Pm	14.5	17.1
	Pm + Pb	15.3	25.7

NOTES

1. Base support anchor bolts and foundation concrete are not ASME B&PV Code NB/NF components. They are included here for completeness and their values represent the worst case bolt removal scenarios.
2. The design load for the upper lateral support threaded holes on the upper tubesheet is the Level D maximum pipe rupture load, and thus this Level D stress is reported here for completeness.
3. Limit analysis per ASME B&PV Code NB-3228.1 is performed for these unreinforced openings.
4. Bounding results for both Main and Emergency Feedwater Nozzles and for models with and without seal weld.
5. The lower shroud shear lugs serve as an anti-vibration device for the lower shroud and are subjected to very small, if any, non-axisymmetric load on the lower shroud. Conservatively considered, it is shown that the shear lugs (and shear keys) can take cyclic OBE secondary imposed loads without fatigue failure.

Table 2.2.2.5.2.2-3
Stress Range and Fatigue u.f. Summary for Level A and B Condition

Component / Location	Classification	Value	Limit
PRIMARY SIDE COMPONENTS			
Primary Inlet Nozzle	SI_{range} (ksi) [1]	38.7	90.0
	U.F. [2]	0.01	1.0
Inlet Nozzle / Primary Head Juncture	SI_{range} (ksi)	41.5	90.0
	U.F.	0.01	1.0
Upper Primary Head at Manway Manipulator Lug	SI_{range} (ksi)	32.3	90.0
	U.F.	0.02	1.0
Upper Primary Head away from Inlet Nozzle	SI_{range} (ksi)	29.0	90.0
	U.F.	0.01	1.0
Primary Manways	SI_{range} (ksi)	50.0	90.0
	U.F.	0.06	1.0
Primary Manway Covers	SI_{range} (ksi)	19.5	80.0
	U.F.	0.01	1.0
Primary Manway Studs	$\sigma_{average}$ (ksi)	48.1	55.2
	$\sigma_{maximum}$ (ksi)	75.1	82.8
	U.F.	0.64	1.0
Primary Manway Diaphragm Seal Welds	U.F.	3.82 [3]	1.0
Primary Handhole	SI_{range} (ksi)	27.9	90.0
	U.F.	0.01	1.0
Primary Handhole Cover	SI_{range} (ksi)	25.4	80.0
	U.F.	0.01	1.0
Primary Handhole Studs	$\sigma_{average}$ (ksi)	38.2	55.2
	$\sigma_{maximum}$ (ksi)	63.8	82.8
	U.F.	0.43	1.0
Primary Handhole Diaphragm Seal Weld	U.F.	0.94	1.0
Primary Outlet Nozzle Elbows	SI_{range} (ksi)	26.7	90.0
	U.F.	0.01	1.0
Primary Outlet Nozzles	SI_{range} (ksi)	25.9	90.0
	U.F.	0.01	1.0

Table 2.2.2.5.2.2-3

Stress Range and Fatigue Usage Factor Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
PRIMARY SIDE COMPONENTS (continued)			
Outlet Nozzle / Primary Head Junctures	SI_{range} (ksi)	72.6	90.0
	U.F.	0.11	1.0
Lower Primary Head away from Outlet Nozzles	SI_{range} (ksi)	48.9	90.0
	U.F.	0.01	1.0
Upper Primary Head	SI_{range} (ksi)	55.8	90.0
	U.F.	0.02	1.0
Upper Primary Head / Tubesheet Knuckle including Vent / Level Sensing	SI_{range} (ksi)	53.0	90.0
	U.F.	0.88	1.0
Upper Lateral Support Holes at Upper Tubesheet	U.F.	0.31	1.0
Thickshell #4 / Upper Tubesheet Knuckle and Fillet Radius	SI_{range} (ksi)	50.9	90.0
	U.F.	0.24	1.0
Thickshell #4 including External Lifting Trunnions and Sample Tap	SI_{range} (ksi)	35.1	90.0
	U.F.	0.05	1.0
Thickshell / Thinshell #4 Juncture	SI_{range} (ksi)	36.8	90.0
	U.F.	0.01	1.0

Table 2.2.2.5.2.2-3
Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
PRIMARY SIDE COMPONENTS (continued)			
Thinshells #3 and #4 including Thermocouples	SI_{range} (ksi)	42.5	90.0
	U.F.	0.07	1.0
Thickshell #3 including Level Sensing, Shell Plate Drain, Thermocouple and Grounding Plate	SI_{range} (ksi)	34.3	90.0
	U.F.	0.05	1.0
Thickshell / Thinshell #3 Juncture including Internal Lower Shroud Shear Lugs	SI_{range} (ksi)	65.7	90.0
	U.F.	0.30	1.0
Thinshells #1, #2 and #3 including Temperature and Level Sensing, Sample Taps and Thermocouples	SI_{range} (ksi)	62.0	90.0
	U.F.	0.48	1.0
Thickshell / Thinshell #1 Juncture	SI_{range} (ksi)	55.3	90.0
	U.F.	0.02	1.0
Thickshell #1 including Temperature and Level Sensing, N2 Nozzle, Sample Tap and Orifice Ring	SI_{range} (ksi)	74.6	90.0
	U.F.	0.26	1.0
Thickshell #1 / Lower Tubesheet Knuckle and Fillet Radius including Drain Nozzle and Acoustic Sensors	SI_{range} (ksi)	87.9	90.0
	U.F.	0.36	1.0
Lower Primary Head / Tubesheet Knuckle	SI_{range} (ksi)	61.3	90.0
	U.F.	0.16	1.0

Table 2.2.2.5.2.2-3
Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

<table>
<tr><th colspan="3">Component / Location</th><th>Classification</th><th>Value</th><th>Limit</th></tr>
<tr><td colspan="6">PRIMARY SIDE COMPONENTS (continued)</td></tr>
<tr><td colspan="3" rowspan="2">Lower Primary Head</td><td>SI_{range} (ksi)</td><td>55.0</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.02</td><td>1.0</td></tr>
<tr><td rowspan="12">Upper and Lower Perforated Tubesheets</td><td rowspan="4">Nominal ligament / ligament efficiency of 0.232” / 0.265 (both sides)</td><td rowspan="2">Primary Side</td><td>SI_{range} (ksi)</td><td>56.6</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.05</td><td>1.0</td></tr>
<tr><td rowspan="2">Secondary Side</td><td>SI_{range} (ksi)</td><td>61.3</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.08</td><td>1.0</td></tr>
<tr><td rowspan="4">Design thin ligament / ligament efficiency of 0.226” / 0.258 primary side and 0.109” / 0.125 secondary side</td><td rowspan="2">Primary Side</td><td>SI_{thin} (ksi) [4]</td><td>27.8</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.05</td><td>1.0</td></tr>
<tr><td rowspan="2">Secondary Side</td><td>SI_{thin} (ksi)</td><td>57.4</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.54</td><td>1.0</td></tr>
<tr><td rowspan="4">Postulated thin ligament / ligament efficiency of 0.093” / 0.106 (both sides)</td><td rowspan="2">Primary Side</td><td>SI_{thin} (ksi)</td><td>67.4</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.76</td><td>1.0</td></tr>
<tr><td rowspan="2">Secondary Side</td><td>SI_{thin} (ksi)</td><td>67.4</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.95</td><td>1.0</td></tr>
<tr><td colspan="3" rowspan="2">Lower Primary Head away from Base Support</td><td>SI_{range} (ksi)</td><td>35.5</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.01</td><td>1.0</td></tr>
<tr><td colspan="3" rowspan="2">Lower Primary Head at Manway Curtain Rod Lugs and Vessel Nameplate</td><td>SI_{range} (ksi)</td><td>32.4</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.24</td><td>1.0</td></tr>
<tr><td colspan="3" rowspan="2">Lower Primary Head / Base Support Juncture</td><td>SI_{range} (ksi)</td><td>52.5</td><td>90.0</td></tr>
<tr><td>U.F.</td><td>0.21</td><td>1.0</td></tr>
</table>

Table 2.2.2.5.2.2-3

Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
PRIMARY SIDE COMPONENTS (continued)			
Lower Primary Head / Base Support Weld	SI_{range} (ksi)	65.1	90.0
	U.F.	0.25	1.0
Base Support Access Hole	SI_{range} (ksi) [5]	20.5	90.0
Base Support Flange at Bolt Holes	SI_{range} (ksi) [5]	6.6	90.0
Tubes	SI_{range} (ksi)	123.3 [6]	80.0
	U.F.	0.16 [6]	1.0
	$\sigma_{comp+bend}$ (ksi)	48.8	80.0
Tube Seal Welds	SI_{range} (ksi)	51.6	69.9
	U.F.	0.74	1.0

Table 2.2.2.5.2.2-3

Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS			
Secondary Manways	SI_{range} (ksi)	99.5 [6]	90.0
	U.F.	0.34 [6]	1.0
Secondary Manway Covers	SI_{range} (ksi)	22.7	80.0
	U.F.	0.17	1.0
Secondary Manway Studs	$\sigma_{average}$ (ksi)	35.0	55.5
	$\sigma_{maximum}$ (ksi)	50.8	74.9
	U.F.	0.41	1.0
Secondary Manway Diaphragm Seal Welds	U.F.	0.23	1.0
Secondary Handholes	SI_{range} (ksi)	94.8 [6]	90.0
	U.F.	0.06 [6]	1.0
Secondary Handhole Covers	SI_{range} (ksi)	14.1	80.0
	U.F.	0.01	1.0
Secondary Handhole Studs	$\sigma_{average}$ (ksi)	40.5	57.6
	$\sigma_{maximum}$ (ksi)	57.8	86.4
	U.F.	0.28	1.0
Secondary Handhole Diaphragm Seal Welds	U.F.	0.10	1.0
Lower Secondary Inspection Ports	SI_{range} (ksi)	104.2 [6]	90.0
	U.F.	0.35 [6]	1.0
Lower Secondary Inspection Port Covers	SI_{range} (ksi)	19.7	80.0
	U.F.	0.01	1.0
Lower Secondary Inspection Port Studs	$\sigma_{average}$ (ksi)	40.0	57.6
	$\sigma_{maximum}$ (ksi)	76.5	86.4
	U.F.	0.28	1.0
Lower Inspection Port Diaphragm Seal Welds	U.F.	0.86	1.0
Upper Secondary Inspection Ports	SI_{range} (ksi)	67.3	90.0
	U.F.	0.34	1.0
Upper Secondary Inspection Port Covers	SI_{range} (ksi)	16.6	80.0
	U.F.	0.04	1.0

Table 2.2.2.5.2.2-3

Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	**Classification**	**Value**	**Limit**
SECONDARY SIDE COMPONENTS (continued)			
Upper Secondary Inspection Port Studs	$\sigma_{average}$ (ksi)	40.0	55.5
	$\sigma_{maximum}$ (ksi)	49.4	74.9
	U.F.	0.11	1.0
Upper Inspection Port Diaphragm Seal Welds	U.F.	0.50	1.0
Steam Outlet Nozzles	SI_{range} (ksi)	48.8	90.0
	U.F.	0.01	1.0
Steam Outlet Nozzle Safe-End Build-Ups	SI_{range} (ksi)	56.6	60.9
	U.F.	0.01	1.0
Steam Outlet Nozzle / Thickshell Junctures	SI_{range} (ksi)	58.3	90.0
	U.F.	0.52	1.0
Thickshells away from Steam Outlet Nozzles	SI_{range} (ksi)	52.2	90.0
	U.F.	0.02	1.0
Lower Shroud Lugs	SI_{range} (ksi)	16.8	59.8
	U.F.	0.01	1.0
Shell at Lower Shroud Lugs	SI_{range} (ksi)	80.4	90.0
	U.F.	0.84	1.0
Upper Shroud Ring at Shell Juncture	SI_{range} (ksi)	30.3	53.3
	U.F.	0.12	1.0
Thickshell at Upper Shroud Ring	SI_{range} (ksi)	25.0	90.0
	U.F.	0.01	1.0
Thinshell away from Upper Shroud Ring	SI_{range} (ksi)	30.2	90.0
	U.F.	0.01	1.0
Main Feedwater Nozzle to Elbow Welds	SI_{range} (ksi)	55.9	71.4
	U.F.	0.02	1.0
Main Feedwater Nozzles at Thermal Sleeves	SI_{range} (ksi)	110.2 [6]	71.4
	U.F.	0.42 [6]	1.0
Main Feedwater Nozzle Forgings	SI_{range} (ksi)	33.3	71.4
	U.F.	0.02	1.0

Table 2.2.2.5.2.2-3
Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS (continued)			
Main Feedwater Nozzle P1 Weld Build-Ups	SI_{range} (ksi)	17.9	67.8
	U.F.	0.01	1.0
Main Feedwater Nozzle Thermal Sleeves	SI_{range} (ksi)	76.4 [6]	69.9
	U.F.	0.29 [6]	1.0
Main Feedwater Nozzle Bolted Flanges	SI_{range} (ksi)	17.1	80.0
	U.F.	0.01	1.0
Main Feedwater Nozzle P1 Seal Welds	U.F.	0.14	1.0
Main Feedwater Nozzle Studs	$\sigma_{average}$ (ksi)	28.8	56.8
	$\sigma_{maximum}$ (ksi)	43.2	76.7
	U.F.	0.25	1.0
Thickshell away from Main Feedwater Nozzles	SI_{range} (ksi)	43.1	90.0
	U.F.	0.09	1.0
Emergency Feedwater Nozzle to Elbow Welds	SI_{range} (ksi)	56.7	71.4
	U.F.	0.04	1.0
Emergency Feedwater Nozzles at Thermal Sleeves	SI_{range} (ksi)	80.5 [6]	71.4
	U.F.	0.28 [6]	1.0
Emergency Feedwater Nozzle Forgings	SI_{range} (ksi)	28.7	71.4
	U.F.	0.05	1.0
Emergency Feedwater Nozzle P1 Weld Build-Ups	SI_{range} (ksi)	15.3	67.8
	U.F.	0.07	1.0
Emergency Feedwater Nozzle Thermal Sleeves	SI_{range} (ksi)	61.9	69.9
	U.F.	0.28	1.0
Emergency Feedwater Nozzle Bolted Flanges	SI_{range} (ksi)	15.7	80.0
	U.F.	0.01	1.0
Emergency Feedwater Nozzle P1 Seal Welds	U.F.	0.07	1.0
Emergency Feedwater Nozzle Studs	$\sigma_{average}$ (ksi)	26.5	56.8
	$\sigma_{maximum}$ (ksi)	39.0	76.7
	U.F.	0.17	1.0

Table 2.2.2.5.2.2-3
Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS (continued)			
Thickshell away from Emergency Feedwater Nozzles	SI_{range} (ksi)	43.5	90.0
	U.F.	0.08	1.0
Main Feedwater Inlet Extension Pipes	SI_{e_rg} (ksi) [7]	94.2 [6]	54.3
	U.F.	0.06 [6]	1.0
Main Feedwater Inlet Elbows	SI_{e_rg} (ksi)	87.2 [6]	53.7
	U.F.	0.50 [6]	1.0
Main Feedwater Inlet Header Tees	SI_{e_rg} (ksi)	77.8 [6]	53.7
	U.F.	0.04 [6]	1.0
Main Feedwater Inlet Header Tee Crotch Regions	U.F. [8]	0.04	1.0
Main Feedwater Inlet Tee Header Pipes	SI_{e_rg} (ksi)	78.7 [6]	53.7
	U.F.	0.15 [6]	1.0
Main Feedwater Riser Pipes	SI_{e_rg} (ksi)	58.5 [6]	53.7
	U.F.	0.01 [6]	1.0
Main Feedwater Riser Branch Stubs	SI_{e_rg} (ksi)	63.4	71.5
	U.F.	0.01	1.0
Main Feedwater Header Pipe / Riser Stub Welds	SI_{e_rg} (ksi)	65.0 [6]	53.7
	U.F.	0.02 [6]	1.0
Main Feedwater Header Pipes	SI_{e_rg} (ksi)	80.6 [6]	53.7
	U.F.	0.14 [6]	1.0
Main Feedwater Nozzle to Riser Elbow Welds	SI_{e_rg} (ksi)	53.1	53.7
	U.F.	0.01	1.0
Main Feedwater Riser Elbows	SI_{e_rg} (ksi)	59.6 [6]	53.7
	U.F.	0.01 [6]	1.0
Main Feedwater Elbow to Riser Welds	SI_{e_rg} (ksi)	45.3	53.7
	U.F.	0.01	1.0
Main Feedwater Riser Pipe Sockolet Welds	SI_{e_rg} (ksi)	43.1	53.7
	U.F.	0.23	1.0

Table 2.2.2.5.2.2-3
Stress Range and Fatigue UF Summary for Level A and B Condition (continued)

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS (continued)			
Emergency Feedwater Inlet Extension Pipe	SI_{e_rg} (ksi)	55.4 [6]	51.7
	U.F.	0.11 [6]	1.0
Emergency Feedwater Inlet Tee	SI_{e_rg} (ksi)	68.8 [6]	53.7
	U.F.	0.41 [6]	1.0
Emergency Feedwater Inlet Tee Crotch Region	U.F. [8]	0.23	1.0
Emergency Feedwater Inlet Tee Header Pipe	SI_{e_rg} (ksi)	51.4	53.7
	U.F.	0.08	1.0
Emergency Feedwater Riser Pipes	SI_{e_rg} (ksi)	47.9	53.7
	U.F.	0.03	1.0
Emergency Feedwater Riser Branch Stubs	SI_{e_rg} (ksi)	53.4	71.2
	U.F.	0.12	1.0
Emergency Feedwater Header Pipe / Riser Stub Welds	SI_{e_rg} (ksi)	47.4	53.7
	U.F.	0.17	1.0
Emergency Feedwater Header Pipes	SI_{e_rg} (ksi)	78.1 [6]	53.7
	U.F.	0.14 [6]	1.0
Emergency Feedwater Nozzle to Riser Elbow Welds	SI_{e_rg} (ksi)	45.8	53.7
	U.F.	0.01	1.0
Emergency Feedwater Riser Elbows	SI_{e_rg} (ksi)	67.5 [6]	53.7
	U.F.	0.01 [6]	1.0
Emergency Feedwater Elbow to Riser Welds	SI_{e_rg} (ksi)	50.9	53.7
	U.F.	0.01	1.0
Emergency Feedwater Riser Pipes	SI_{e_rg} (ksi)	48.0	53.7
	U.F.	0.25	1.0

NOTES

1. SI_{range} is the range of (Pm or PL + Pb + Q) per ASME B&PV Code Figure NB-3222-1.
2. U.F. is the cumulative fatigue usage factor. A conservative upper bound value of 0.01 is reported for all insignificant fatigue usage factor values.
3. The primary manway seal weld lasts for 40.0 / 3.82 = 10.4 years of continuous service. Currently no primary man way seal weld is in place. “10.4 years” is judged to be acceptable since the primary manway seal weld, if used, cannot remain in place for more than 6 years as Technical

Specifications require that no generator can operate for more than 6 years or three refueling outages (whichever is less) without being inspected (i.e., the primary manway needs to be opened for maintenance service).

4. SI_{thin} is the perforated tubesheet thin ligament average stress intensity per ASME Appendix A-8143.1.
5. The base support is ASME Subsection NF component which doesn't require fatigue analysis.
6. The stress intensity entry is the result of a simplified linear analysis. When the limit is exceeded, a simplified elastic-plastic analysis is performed to the requirements of ASME B&PV Code Section III, NB-3228.5 applying the elastic-plastic factor (Ke) to the fatigue usage factor. In all cases these analyses met the acceptance criteria of ASME B&PV Code NB-3228.5. Results from simplified elastic-plastic analysis are presented in Table 2.2.2.5.2.2-4.
7. SI_{e_rg} is the range of (PL + Pb + Pe + Q) and Pe is the secondary expansion stress per ASME B&PV Code Figure NB-3222-1 for piping components.
8. For the crotch region in a piping tee, the stress range analysis is not required per ASME B&PV Code Table NB-3217-2.

TABLE 2.2.2.5.2.2-4
NB322-8-5 Simplified Elastic-Plastic Analysis

Component / Location	Classification	Value	Limit
PRIMARY SIDE COMPONENTS			
Tubes	NB322-8-5 (a)	78.9 ksi	80.0 ksi
	NB322-8-5 (b)	2.785 max	N/A
	NB322-8-5 (c)	0.16	1.0
	NB322-8-5 (d)	38.4 ksi	80.0 ksi
	NB322-8-5 (e)	644^0F	800^0F
	NB322-8-5 (f)	0.471	0.8
SECONDARY SIDE COMPONENTS			
Secondary Manways	NB322-8-5 (a)	62.4 ksi	90.0 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.34	1.0
	NB322-8-5 (d)	99.5 ksi	114.7 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.72	0.8
Secondary Handholes	NB322-8-5 (a)	42.6 ksi	90.0 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.06	1.0
	NB322-8-5 (d)	94.8 ksi	114.5 ksi
	NB322-8-5 (e)	560^0F	700^0F
	NB322-8-5 (f)	0.72	0.8

Component / Location	**Classification**	**Value**	**Limit**
SECONDARY SIDE COMPONENTS			
Emergency Feedwater Nozzle at Thermal Sleeves	NB322-8-5 (a)	61.0 ksi	90.0 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.35	1.0
	NB322-8-5 (d)	104.2 ksi	114.5 ksi
	NB322-8-5 (e)	560^0F	700^0F
	NB322-8-5 (f)	0.72	0.8
Main Feedwater Nozzle at Thermal Sleeves	NB322-8-5 (a)	47.2 ksi	71.4 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.28	1.0
	NB322-8-5 (d)	Note 2	N/A
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.63	0.8
Main Feedwater Inlet Extension Pipes	NB322-8-5 (a)	42.4 ksi	54.3 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.06	1.0
	NB322-8-5 (d)	Note 2	N/A
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.58	0.8

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS			
Main Feedwater Inlet Elbows	NB322-8-5 (a)	41.0 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.5	1.0
	NB322-8-5 (d)	Note 2	N/A
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.58	0.8
Main Feedwater Inlet Header Tees	NB322-8-5 (a)	36.0 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.04	1.0
	NB322-8-5 (d)	Note 2	N/A
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.58	0.8
Main Feedwater Inlet Tee Header Pipes	NB322-8-5 (a)	21.0 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.15	1.0
	NB322-8-5 (d)	Note 2	N/A
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.58	0.8

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS			
Main Feedwater Riser Pipes	NB322-8-5 (a)	26.8 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.01	1.0
	NB322-8-5 (d)	54.7 ksi	69.0 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.5	0.8
Main Feedwater Header Pipe/Rise Stub Welds	NB322-8-5 (a)	19.9 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.02	1.0
	NB322-8-5 (d)	54.7 ksi	69.0 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.5	0.8
Main Feedwater Header Pipe	NB322-8-5 (a)	26.8 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.14	1.0
	NB322-8-5 (d)	54.7 ksi	69.0 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.5	0.8

Component / Location	**Classification**	**Value**	**Limit**
SECONDARY SIDE COMPONENTS			
Main Feedwater Riser Elbows	NB322-8-5 (a)	45.9 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.01	1.0
	NB322-8-5 (d)	Note 2	N/A
	NB322-8-5 (e)	600^0F	700^0F
	NB322-8-5 (f)	0.5	0.8
Emergency Feedwater Inlet Extension Pipes	NB322-8-5 (a)	23.8 ksi	51.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.11	1.0
	NB322-8-5 (d)	34.6 ksi	77.0 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.6	0.8
Emergency Feedwater Inlet Tee	NB322-8-5 (a)	37.1 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.41	1.0
	NB322-8-5 (d)	34.6 ksi	77.0 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.5	0.8

Component / Location	Classification	Value	Limit
SECONDARY SIDE COMPONENTS			
Emergency Feedwater Header Pipes	NB322-8-5 (a)	45.7 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.14	1.0
	NB322-8-5 (d)	35.0 ksi	77.0 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.5	0.8
Emergency Feedwater Riser Elbows	NB322-8-5 (a)	52.9 ksi	53.7 ksi
	NB322-8-5 (b)	Note 1	N/A
	NB322-8-5 (c)	0.01	1.0
	NB322-8-5 (d)	55.5 ksi	63.5 ksi
	NB322-8-5 (e)	640^0F	700^0F
	NB322-8-5 (f)	0.5	0.8

Notes:

1. The Ke factors are internally calculated and applied by ATAPP fatigue analysis computer program. ATAPP 1-0 01 Transient Analysis Post-Processor calculates stress range from ANSYS output and, performs fatigue analysis.
2. There is no internal pressure or negligible internal pressure when the maximum stress intensities occur. Therefore, thermal ratcheting will not happen and the requirement of NB-3222.5 will be certainly satisfied.

NB-3228.5 Simplified Elastic–Plastic Analysis. The $3S_m$ limit on the range of primary plus secondary stress intensity (NB-3222.2) may be exceeded provided that the requirements of (a) through (f) below are met.

a. The range of primary plus secondary membrane plus bending stress intensity, excluding thermal bending stresses, shall be $\leq 3S_m$.
b. The value of S_a used for entering the design fatigue curve is multiplied by the factor K_e, where:

$$K_e = 1.0, \text{ for } S_n \leq 3S_m$$

$$= 1.0 + [(1 - n) / n(m - 1)](S_n / 3S_m - 1), \text{ for } 3S_m < S_n < 3mS_m$$

$$= 1 / n, \text{ for } S_n \geq 3mS_m$$

S_n = range of primary plus secondary stress intensity, psi

The values of the material parameters m and n for the various classes of permitted materials are as given in Table NB-3228.5(b)-1.

c. The rest of the fatigue evaluation stays the same as required in NB-3222.4, except that the procedure of NB-3227.6 need not be used.
d. The component meets the thermal ratcheting requirement of NB-3222.5.
e. The temperature does not exceed those listed in Table NB-3228.5(b)-1 for the various classes of materials.
f. The material shall have a specified minimum yield strength to specified minimum tensile strength ratio of less than 0.80.

Table 2.2.2.5.2.2-5

Stress Summary for Level C Condition

Component / Location [1]	Stress Classification	Stress (ksi)	Limit (ksi)
PRIMARY SIDE COMPONENTS			
Primary Manway Studs	$\sigma_{average}$	41.6	56.6
	$\sigma_{maximum}$	57.6	84.9
Primary Handhole Studs	$\sigma_{average}$	37.4	56.6
	$\sigma_{maximum}$	58.6	84.9
Tubes	$\sigma_{comp+bend}$	54.1	80.0
SECONDARY SIDE COMPONENTS			
Secondary Manway Studs	$\sigma_{average}$	22.4	56.8
	$\sigma_{maximum}$	32.8	85.2
Secondary Handhole Studs	$\sigma_{average}$	45.9	56.8
	$\sigma_{maximum}$	68.0	85.2
Secondary Inspection Port Studs	$\sigma_{average}$	39.4	57.4
	$\sigma_{maximum}$	79.9	86.1
Main Feedwater Nozzle Studs	$\sigma_{average}$	31.1	56.8
	$\sigma_{maximum}$	71.7	85.2
Emergency Feedwater Nozzle Studs	$\sigma_{average}$	35.2	56.8
	$\sigma_{maximum}$	71.5	85.2

NOTE

1. For components not listed in this table, the Level C condition is bounded by the Design condition. The Design condition results are listed in Table 2.2.2.5.2.1-1.

Table 2.2.2.5.2.2-6
Stress Summary for Level D Condition

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
PRIMARY SIDE COMPONENTS			
Primary Manway Studs	$\sigma_{average}$	44.3	84.9
	$\sigma_{maximum}$	66.7	125.0
Primary Handhole Studs	$\sigma_{average}$	41.4	84.9
	$\sigma_{maximum}$	57.2	125.0
Tubesheet Perforated Regions	Pm	34.9	63.0
	Pm (PL) + Pb	75.4	94.5
Tubesheet Solid Rims	Pm	11.4	63.0
	Pm (PL) + Pb	30.7	94.5
Primary Heads	Pm	11.8	63.0
	Pm (PL) + Pb	74.5	94.5
Secondary Shells	Pm	11.3	63.0
	Pm (PL) + Pb	24.3	94.5
	$\sigma_{compressive}$	10.5	18.6
Tubes	Pm	17.9	56.0
	Pm (PL) + Pb	34.4	84.0
Tube Seal Welds	$\sigma_{maximum}$	41.2	50.7
Head at Base Support Stool	Pm	36.2	63.0
	Pm (PL) + Pb	42.8	94.5
Lower Head at Flat Section	Pm	21.0	63.0
	Pm + Pb	40.0	94.5
Base Support Stool	Pm	56.0	63.0

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
	Pm + Pb	60.8	94.5

Table 2.2.2.5.2.2-6
Stress Summary for Level D Condition (continued)

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
SECONDARY SIDE COMPONENTS			
Secondary Manway Studs	$\sigma_{average}$	40.5	85.3
	$\sigma_{maximum}$	57.8	125.0
Secondary Handhole Studs	$\sigma_{average}$	44.3	85.3
	$\sigma_{maximum}$	57.8	125.0
Secondary Inspection Port Studs	$\sigma_{average}$	38.4	86.2
	$\sigma_{maximum}$	46.9	125.0
Shroud Lugs	Pm	1.9	49.0
	Pm + Pb	19.1	73.5
Thickshell at Shroud Lugs	PL	13.4	94.5
Shroud Ring	Pm	11.6	45.5
	Pm (PL) + Pb	12.2	68.3
Thickshell at Shroud Ring	PL	11.2	94.5
Main Feedwater Nozzle Studs	$\sigma_{average}$	31.2	85.3
	$\sigma_{maximum}$	47.3	125.0
Main Feedwater 3” Riser Straight Pipes	$SI_{Equation\_9}$ [2]	32.3	53.7
Main Feedwater 3” Riser Bend Pipes	$SI_{Equation\_9}$ [2]	36.6	53.7
Main Feedwater 3” Riser Elbows	$SI_{Equation\_9}$ [2]	37.4	53.7
Main Feedwater Riser Tees	$SI_{Equation\_9}$ [2]	35.9	53.7
Main Feedwater 14” Header Pipes	$SI_{Equation\_9}$ [2]	7.6	53.7
Main Feedwater 14” Inlet Tees	$SI_{Equation\_9}$ [2]	18.6	53.7
Main Feedwater 14” Inlet Elbows	$SI_{Equation\_9}$ [2]	16.4	53.7

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
Main Feedwater 14" Inlet Extensions	$SI_{Equation\_9}$ [2]	13.0	57.4

Table 2.2.2.5.2.2-6
Stress Summary for Level D Condition (continued)

Component / Location	Stress Classification	Stress (ksi)	Limit (ksi)
SECONDARY SIDE COMPONENTS (continued)			
Emergency Feedwater Nozzle Studs	$\sigma_{average}$	28.4	83.0
	$\sigma_{maximum}$	44.2	125.0
Emergency Feedwater 3" Riser Pipes	$SI_{Equation\_9}$ [2]	18.3	53.7
Emergency Feedwater 3" Riser Elbows	$SI_{Equation\_9}$ [2]	25.5	53.7
Emergency Feedwater Riser Tees	$SI_{Equation\_9}$ [2]	42.5	53.7
Emergency Feedwater 6" Header Pipes	$SI_{Equation\_9}$ [2]	9.6	53.7
Emergency Feedwater 6" Inlet Tee	$SI_{Equation\_9}$ [2]	25.0	53.7
Emergency Feedwater 6" Inlet Extension	$SI_{Equation\_9}$ [2]	11.8	57.4

NOTES

1. For components not listed in this table, the Level D condition is bounded by the Design condition. The Design condition results are listed in Table 2.2.2.5.2.1-1.
2. Stress intensity per Equation (9) of ASME B&PV Code NB-3652 and NB-3656.

Table 2.2.2.5.2.2-7
Internals Stress Summary for EPU Design Condition

Component / Location	Classification		Value	Limit
Shroud Skirt [1]	σ_{normal}	(ksi)	0.7	21.0
	τ_{shear}	(ksi)	0.1	21.0
Shroud Lugs	σ_{normal}	(ksi)	6.6	21.0
	τ_{shear}	(ksi)	0.5	21.0
Shroud Pins [2]	$\sigma_{average}$	(ksi)	1.0	62.5
	τ_{shear}	(ksi)	0.4	11.7
Shroud Keys	τ_{shear_weld}	(ksi)	8.1	21.0
	τ_{shear_base}	(ksi)	5.7	11.4
Tube Support Plates	Pm	(ksi)	15.6	23.3
	Pm + Pb	(ksi)	28.4	34.9
	$\sigma_{bearing}$	(ksi)	22.3	38.0
	τ_{shear}	(ksi)	4.6	14.0
Tie Rods [3]	$\sigma_{average}$	(ksi)	21.7	65.0
	$\sigma_{maximum}$	(ksi)	66.9	100.0
	$\sigma_{bearing}$	(ksi)	22.3	100.0
	τ_{shear}	(ksi)	7.8	26.0
Anti-Rotation Blocks	τ_{shear_weld}	(ksi)	14.9	17.4
	τ_{shear_base}	(ksi)	10.6	14.4
Filler Bars	τ_{shear_weld}	(ksi)	5.8	17.4
	τ_{shear_base}	(ksi)	4.1	14.4

NOTES

1. The stresses on the shroud skirt bound those of the shroud cans and cone.

2. The shroud pin available thread length is 2.98” and is larger the 2.37” required to prevent stripping of internal threads.
3. The tie rod available thread engagement length is 0.99” and is larger than the 0.46” required to prevent stripping of internal thread.

Table 2.2.2.5.2.2-8
Internals Stress and Fatigue Summary for EPU Level A and B Operating Condition

Component / Location	Classification	Value	Limit
Tube Support Plates [1]	Range of SI (ksi)	48.6	57.1
	Fatigue U.F.	0.08	1.0
Tie Rod Threads	$\sigma_{maximum}$ (ksi)	27.2	108.3
	$F_{compressive}$ (lbf)	2966	3827 [2]
	Fatigue U.F.	0.29	1.0

NOTES

1. Tube support plate maximum rotation during service conditions is 0.82 degree which is less than the available broached hole to tube clearance of 0.88 degree and thus the tube is not locked in the tube support plate during service conditions.
2. 3827 lbf is 2/3 of the tie rod elastic buckling load.

Table 2.2.2.5.2.2-9
Internals Stress Summary for EPU Level D Faulted Condition

Component / Location	Classification		Value	Limit
Shroud Skirt [1]	σ_{normal}	(ksi)	1.0	42.0
	τ_{shear}	(ksi)	0.2	42.0
Shroud Lugs	σ_{normal}	(ksi)	9.2	42.0
	τ_{shear}	(ksi)	0.8	42.0
Shroud Pins	$\sigma_{average}$	(ksi)	2.0	83.9
	τ_{shear}	(ksi)	0.7	29.4
Shroud Keys	Bounded by Design condition as reported in Table 2.2.2.5.2.2-5			
Tube Support Plates	Pm	(ksi)	17.9	34.9
	Pm + Pb	(ksi)	30.7	52.3
	$\sigma_{bearing}$	(ksi)	19.1	147.0
	τ_{shear}	(ksi)	4.0	29.4
Tie Rods	$\sigma_{average}$	(ksi)	33.9	84.1
	$\sigma_{maximum}$	(ksi)	76.3	84.3
	$\sigma_{bearing}$	(ksi)	19.1	252.4
	τ_{shear}	(ksi)	6.7	50.5
Anti-Rotation Blocks	Bounded by Design condition as reported in Table 2.2.2.5.2.2-5			
Filler Bars				

NOTE

1. The stresses on the shroud skirt bound those of the shroud cans and cone.

Table 2.2.2.5.2.2-10

Tube RG 1.121 / EPRI Flaw Size Results - Allowable Flaw Size for Tubes with Volumetric Defects

Type of Flaws	**a/t**	**Tube Burst Pressure / Axial Force**	**EPRI Structural Integrity Performance Criteria Limits**		
			3.0 NOPD	**1.4 LAPD**	**1.2 PL + 1.0 ASL**
Uniform 360° Thinning over a given TSP Length of 1.125”	40%	P_{burst}=5562 psi F_{ax}=2981 lbf	4050 psi	4620 psi	2887 lbf
Tapered Fret with a length of 1.125” and 135° circumferential extent	70% #	P_{burst}=4663 psi F_{ax}=3618 lbf	4050 psi	4620 psi	2887 lbf

a crack depth

t nominal tube wall thickness

Maximum value at one end of tapered fret

Table 2.2.2.5.2.3-1
Input Parameters for EPU Conditions

	Start-Up Conditions	**End of Life Conditions**
RCS Loop Coolant Flow – lb/hr	65.8×10^6	65.8×10^6
Fouling Factor, hr-ft^2-°F/Btu	0.00002	0.00005
Steam Nozzle Pressure - psia	964	964
RCS Average Temperature - °F	582	582
Feedwater Temperature - °F	461	461

The following thermal-hydraulic acceptance criteria were adopted for the EPU conditions:

- Superheat ≥ 35°F
- Secondary Side Inventory ≤ 68,059 lbm

Table 2.2.2.5.2.3-2
Crystal River Unit 3 OTSG Thermal-Hydraulic Results for Bounding Cases, Thermal Design Flow

Thermal-hydraulic Attribute		Start-Up for original Conditions	End of Life for original Conditions	Start-Up for EPU Conditions	End of Life for EPU Conditions
NSSS Power [MWt] (includes pump power)		2584	2584	3030	3030
Plugging [%]		0	20	0	5.2
Temperatures [°F]	Feedwater	460	460	461	461
	Primary Inlet	604	604	611	611
	Avg. Primary	579	579	582	582
	Primary Outlet	554	554	553	553
	Superheat	60.62	47.58	46.96	35.01
Total Pressure [psia]	Steam Outlet Nozzle	925	925	964	964
Flow Rates [10^6 lbm/hr] per steam generator	Steam	5.425	5.500	6.467	6.558
	Primary Fluid	65.7	65.7	65.8	65.8
Total Inventory – [lbm]	Secondary	48,473	≤ 56,400	63,562	≤ 68,059

2.2.2.6 Reactor Coolant Pumps and Supports

2.2.2.6.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation.

2.2.2.6.2 Technical Evaluation

Introduction

The reactor coolant pumps (RCPs) are described in the CR-3 FSAR, Sections 4.1.3.3 and 4.2.2.5. The RCP supports are described in CR-3 FSAR Section 4.2.6.4, 4.2.6.5 and 4.2.6.6. Each reactor coolant loop contains two single stage, single suction, constant speed, vertical centrifugal reactor coolant pumps which return coolant from the steam generators to the reactor vessel. Each RCP employs a shaft sealing system consisting of three mechanical seal assemblies arranged in a removable cartridge and a top vapor barrier standpipe to prevent reactor coolant leakage to the atmosphere. The functions of the RCPs are:

- To maintain an adequate cooling flow rate by circulating a large volume of primary coolant water at high temperature and pressure through the Reactor Coolant System (RCS).
- To provide adequate flow coastdown to prevent core damage in the event of a simultaneous loss of power to all pumps.
- To provide a portion of the reactor coolant pressure boundary (the pressure boundary parts of the RCP).

The RCP casing, internals, and motor weight are connected to the 28 inch reactor coolant lines and supported by constant load hangers, link bars and seismic snubbers. The design of the RCP Casings is in accordance with the 1968 Edition (no Addenda) of the ASME B&PV Code, Section III for Class A Vessels. The structural portions of the constant load hangers for the RCPs are constructed in accordance with USAS B31.7 “USA Standard Code for Pressure Piping, Nuclear Power Piping,” and/or MSS-SP-58, as applicable.

Description of Analyses and Evaluation

The RCPs are installed in the RCS cold leg, between the steam generator outlet and the reactor vessel inlet. Therefore, the RCS operating parameters which could potentially impact the structural integrity of the RCPs are T_{COLD} and RCS operating pressure. As shown in Table 1.1-1, T_{COLD} is decreasing by less than 1°F for EPU. Additionally, as shown in Table 1.1-1, the RCS operating pressure will not change for uprate. Further, the NSSS Design Transient parameters were considered in the EPU evaluations. The temperature changes per time are unchanged (i.e. the overall shape of the transient curves remains the same for the EPU). Only the beginning and end temperatures are changing which has a negligible effect on thermal expansion. Thus, there are no changes to the NSSS transients that would affect the reactor coolant pumps or their supports. Also, no changes have been made to the allowable number of design cycles defined for each transient. Thus, the RCS operating parameters which could potentially impact the RCPs are essentially unchanging and, therefore, the RCPs and RCP supports are not impacted by the EPU.

As discussed in Section 2.2.2, Pressure-Retaining Components and Component Supports, the current design basis structural analysis was performed for the RCS loop due to the replacement steam generators (OTSGs), but also considered the proposed EPU conditions. This analysis employed the same methodology as used in the pre-OTSG (pre-EPU) structural analyses. Loads for the RCP nozzles and supports were calculated for the EPU considering deadweight, thermal, seismic and pipe rupture effects. A set of allowable loads was determined for the RCP nozzles and supports based on the loads previously analyzed and qualified prior to the steam generator replacement. The RCP nozzles and supports were then qualified for the EPU using the load comparison method. That is, the EPU loads for each nozzle/support were compared to the corresponding allowable loads. Where the EPU analysis loads were higher than the allowable loads, stress and fatigue analyses were performed in accordance with the ASME Code.

RCP Supports

The RCP supports have been evaluated to be acceptable for the EPU loads. This is done by comparing the EPU loads to the allowable loads (the loads qualified prior to SG replacement). The worst loading combination from all of the snubbers was calculated and compared to the allowable load for that snubber. Similarly, the link bars were evaluated by comparing the EPU loads to the allowable loads.

RCP Suction Nozzles

A similar loading comparison was performed for the suction nozzles and indicated that some of the EPU loads were greater than the allowables. Therefore, a stress and fatigue evaluation was performed.

The suction nozzle stresses were calculated at the end of the nozzle for the loading combinations listed in FSAR Table 4-24. Using Subsection NB-3227.5 of Section III, Division I of the 1992 ASME Boiler and Pressure Vessel Code (with 1993 Addenda), the stress calculations were done using inside the limits of reinforcement methodology; therefore thermal expansion was added to each of the loading combinations and bending was included in the calculation of primary general membrane stress intensity (P_m).

<u>Primary + Secondary Stress / Fatigue</u>

For inside the limits of reinforcement, the following primary + secondary stress intensity criteria must be satisfied.

$P_L+P_b+Q \leq 3.0*S_m$ (ASME Code, Section III, NB-3227.5)

where P_L = primary local membrane stress intensity
P_b = primary bending stress intensity
Q = secondary stress intensity
S_m = allowable membrane stress intensity

RCP Discharge Nozzles

Similar to the suction nozzles, a stress and fatigue evaluation was performed for the RCP discharge nozzles. The discharge nozzle stresses were calculated at the end of the nozzle similar to the suction nozzle stress calculations. The discharge nozzle dimensions and material properties are the same as the suction nozzles at the end of the nozzle, therefore the stress allowables and stress calculations were the same as for the suction nozzles (only the loads change).

Primary + Secondary Stress / Fatigue

Similar to the suction nozzles, inside the limits of reinforcement, the following primary + secondary stress intensity criteria must be satisfied.

$P_L+P_b+Q \leq 3.0*S_m$ (ASME Code, Section III, NB-3227.5)

Results

RCP Supports Results

Tables 2.2.2.6-1 and 2.2.2.6-2 present the comparison of the RCP snubbers and link bar supports axial loads under EPU conditions against the allowable loads.

As seen in these tables, the calculated axial loads on the RCP support snubbers and link bars are less than the allowable loads and require no further evaluation.

RCP Suction Nozzles Results

Tables 2.2.2.6-3 through 2.2.2.6-6 show the ratio of the primary general membrane stress intensity (P_m) under the EPU conditions to the allowable stress limit for each RCP Suction Nozzle Per the current licensing basis, the Deadweight (DW) and Thermal (THRM) loads were combined algebraically in all loading combination Cases. The DW+THRM loads were added absolutely to the Operating Basis Earthquake (OBE), Safe Shutdown Earthquake (SSE) and High-Energy Line Break Accident (HELBA) loads in all loading combination Cases. The SSE and HELBA loads were combined by square root sum of the squares for loading combination Case IV.

Table 2.2.2.6-7 provides the ratio of the primary+secondary stress intensity ($P_L + P_b + Q$) to the allowable stress limit for each RCP Suction Nozzle.

From these tables, the calculated EPU stresses for all four RCP suction nozzles are less than the ASME allowable stress limits for primary and primary+secondary stress. Furthermore, the calculation of the EPU secondary stress (Q) above is equivalent to the maximum EPU stress intensity seen by the nozzle, which is less than the limiting stress intensity. Therefore, the cumulative usage factor remains unchanged from the previous analysis, and the RCP suction nozzles are qualified for the EPU loads.

RCP Discharge Nozzles Results

Tables 2.2.2.6-8 through 2.2.2.6-11 show the ratio of P_m under the EPU conditions to the allowable stress limit for each RCP Suction Nozzle Per the current licensing basis, the DW and THRM loads were combined algebraically in all loading combination Cases. The DW+THRM loads were added absolutely to the OBE, SSE and HELBA loads in all loading combination Cases and the SSE and HELBA loads were combined by square root sum of the squares for loading combination Case IV.

Table 2.2.2.6-12 provides the ratio of the EPU primary+secondary stress intensity ($P_L + P_b + Q$) to the allowable stress limit for each RCP Discharge Nozzle.

From these tables, the calculated EPU stresses for all four RCP discharge nozzles are less than the ASME allowable stress limits for primary and primary+secondary stress. Furthermore, the calculation of the EPU secondary stress (Q) above is equivalent to the maximum EPU stress intensity seen by the nozzle, which is less than the limiting stress intensity. Therefore, the cumulative usage factor remains unchanged from the previous analysis, and the RCP discharge nozzles are qualified for the EPU loads.

The nozzles have been shown acceptable and they are the limiting section of the RCP casing. Further, since the EPU support loads are less than the allowable loads, it can be concluded that the overall load on the RCPs at the EPU conditions remain lower than allowable limits and, thus, the loads on miscellaneous parts of the RCPs remain lower than allowable limits. Therefore, the pump cover, bolts and other miscellaneous structural parts of the RCPs will maintain their structural integrity for the EPU.

2.2.2.6.3 Conclusion

See Section 2.2.2 for Conclusion.

2.2.2.6.4 References

None

Table 2.2.2.6-1: RCP Snubber Supports - Axial Load Comparison to Allowables

Support	Allowable Axial Load (kips)	Worst Case Axial Load (kips)	Ratio
RCH-614 (A2)	1200	362.23	0.30
RCH-620 (D3)	1200	359.97	0.30
RCH-618 (C4)	1600	373.64	0.23
RCH-619 (C8)	1600	368.55	0.23

Table 2.2.2.6-2: RCP Link Bar Supports - Axial Load Comparison to Allowables

Support	Allowable Axial Load (kips)	Worst Case Axial Load (kips)	Ratio
RCH-622 (A7)	1200	496.88	0.414
RCH-623 (B5)	2000	493.55	0.247
RCH-624 (B7)	1200	498.44	0.415
RCH-625 (D5)	1600	488.61	0.305

Table 2.2.2.6-3: Northeast RCP Suction Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	16726	676	16.78	18.7	0.897
Case II (DW+THRM)+SSE	17178	726	17.24	22.44	0.768
Case III (DW+THRM)+HELBA	21486	663	21.53	23.18	0.929
Case IV (DW+THRM)+(SSE+HELBA)	22355	736	22.40	46.67	0.480

Notes: (1) σ_N = bending stress in nozzle (includes pressure)
(2) τ = shear stress in nozzle
(3) RCP Nozzles labeled Northwest, Northeast, Southwest or Southeast with respect to Plant North with the pressurizer connected to the Northwest loop.

Table 2.2.2.6-4: Northwest RCP Suction Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	16486	152	16.49	18.7	0.882
Case II (DW+THRM)+SSE	17008	210	17.01	22.44	0.758
Case III (DW+THRM)+HELBA	22441	129	22.44	23.18	0.968
Case IV (DW+THRM)+(SSE+HELBA)	22547	218	22.55	46.67	0.483

Table 2.2.2.6-5: Southwest RCP Suction Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	16570	698	16.63	18.7	0.889
Case II (DW+THRM)+SSE	17040	748	17.11	22.44	0.762
Case III (DW+THRM)+HELBA	18682	685	18.73	23.18	0.808
Case IV (DW+THRM)+(SSE+HELBA)	19542	756	19.60	46.67	0.420

Table 2.2.2.6-6: Southeast RCP Suction Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	17385	104	17.39	18.7	0.930
Case II (DW+THRM)+SSE	17971	188	17.97	22.44	0.801
Case III (DW+THRM)+HELBA	22554	54	22.55	23.18	0.973
Case IV (DW+THRM)+(SSE+HELBA)	23305	194	23.31	46.67	0.499

Table 2.2.2.6-7: RCP Suction Nozzle $P_L + P_b$ Stress to Allowable Stress Ratio

RCP Nozzle	Stress Location	σ_N (psi)	τ (psi)	$P_L + P_b$ (ksi)	Q (ksi)	$P_L + P_b + Q$ (ksi)	Allowable (ksi)	Ratio
Northeast	Inside Radius	15034	610	15.08	19.57	34.65	57.96	0.598
	Outside Radius	17191	743	17.25	19.57	36.82	57.96	0.635
Northwest	Inside Radius	14849	137	14.85	19.57	34.42	57.96	0.594
	Outside Radius	16896	167	16.90	19.57	36.46	57.96	0.629
Southwest	Inside Radius	14895	629	14.95	19.57	34.51	57.96	0.595
	Outside Radius	17016	767	17.08	19.57	36.65	57.96	0.632
Southeast	Inside Radius	15626	94	15.63	19.57	35.19	57.96	0.607
	Outside Radius	17917	115	17.92	19.57	37.48	57.96	0.647

Table 2.2.2.6-8: Northeast RCP Discharge Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	13678	991	13.82	18.7	0.739
Case II (DW+THRM)+SSE	14127	1100	14.30	22.44	0.637
Case III (DW+THRM)+HELBA	19594	1022	19.70	23.18	0.849
Case IV (DW+THRM)+(SSE+HELBA)	20473	1153	20.60	46.67	0.441

Table 2.2.2.6-9: Northwest RCP Discharge Nozzle Pm Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	15194	1211	15.39	18.7	0.823
Case II (DW+THRM)+SSE	15629	1295	15.84	22.44	0.706
Case III (DW+THRM)+HELBA	21763	1282	21.91	23.18	0.945
Case IV (DW+THRM)+(SSE+HELBA)	21840	1359	22.01	46.67	0.472

Table 2.2.2.6-10: Southwest RCP Discharge Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	13844	931	13.97	18.7	0.747
Case II (DW+THRM)+SSE	14391	1035	14.54	22.44	0.648
Case III (DW+THRM)+HELBA	15851	1133	16.01	23.18	0.690
Case IV (DW+THRM)+(SSE+HELBA)	16812	1206	16.98	46.67	0.364

Table 2.2.2.6-11: Southeast RCP Discharge Nozzle P_m Stress to Allowable Stress Ratio

Load Combination	σ_N (psi)	τ (psi)	P_m (ksi)	Allowable (ksi)	Ratio
Case I (DW+THRM)+OBE	14872	1121	15.04	18.7	0.804
Case II (DW+THRM)+SSE	15444	1216	15.63	22.44	0.697
Case III (DW+THRM)+HELBA	20541	1172	20.67	23.18	0.891
Case IV (DW+THRM)+(SSE+HELBA)	21326	1272	21.48	46.67	0.460

Table 2.2.2.6-12: RCP Discharge Nozzle $P_L + P_b$ Stress to Allowable Stress Ratio

RCP Nozzle	Stress Location	σ_N (psi)	τ (psi)	$P_L + P_b$ (ksi)	Q (ksi)	$P_L + P_b + Q$ (ksi)	Allowable (ksi)	Ratio
Northeast	Inside Radius	12272	893	12.40	19.57	31.97	57.96	0.552
	Outside Radius	13854	1089	14.02	19.57	33.59	57.96	0.580
Northwest	Inside Radius	13613	1092	13.79	19.57	33.35	57.96	0.575
	Outside Radius	15546	1331	15.77	19.57	35.34	57.96	0.610
Southwest	Inside Radius	12416	839	12.53	19.57	32.09	57.96	0.554
	Outside Radius	14041	1023	14.19	19.57	33.76	57.96	0.582
Southeast	Inside Radius	13356	1011	13.51	19.57	33.07	57.96	0.571
	Outside Radius	15158	1233	15.36	19.57	34.92	57.96	0.603

2.2.2.7 Pressurizer and Supports

2.2.2.7.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation.

2.2.2.7.2 Technical Evaluation

Introduction

The CR-3 pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the Reactor Coolant System (RCS) piping at the reactor outlet (T_{HOT}). The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section to maintain RCS pressure within desired limits. The electrically heated pressurizer establishes and maintains the RCS pressure within prescribed limits and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The Pressurizer and Supports are addressed in the FSAR Sections 4.2.2.3, 4.2.4, 4.3.6, 4.3.7, 4.3.8, 4.3.9, 4.6, and 4.6.1.

The design of the pressurizer is in accordance with the 1965 Edition, with Addenda through Summer 1967, of the ASME B&PV Code, Section III for Class A vessels. The pressurizer surge line nozzle was reconciled to the ASME B&PV Section III, Subsection NB 1986 Code, and includes the fatigue evaluation and the effects of thermal stratification as stipulated in NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, December 20, 1988. Thus, the acceptance criteria for the pressurizer surge line nozzle thermal stratification analysis are per the ASME B&PV Code and are as specified in the current design basis in BAW-2127 (Reference 1). The acceptance criteria for the surge line piping discussed in Section 2.2.2.1, NSSS Piping, Components and Supports.

Description of Analyses and Evaluations

The major inputs used in the evaluation of the Pressurizer and Supports are provided in Section 1.0, Introduction to the Crystal River 3 Extended Power Uprate Technical Report, (for operating parameters) and Section 2.2.2.8, NSSS Design Transients (for transients). These sections provided the operating and transient conditions that were considered in the EPU evaluation of the pressurizer and supports.

Seismic analyses and non-pressure boundary component evaluations were considered to be unaffected by the EPU as the conditions in the original design specification remain bounding.

As stated in Section 1, the RCS operating pressure is not changing for the EPU (and, therefore, T_{SAT} is not changing). The cold leg temperature (T_{COLD}) will decrease by less than 1°F due to the EPU as shown in Table 1.1-1. Since the operating pressure and temperature of the pressurizer are not impacted by the EPU, the pressurizer vessel will not experience any increase in thermal loads during normal operation. Further, this small change in T_{COLD} will have very little effect on the temperature differential between the cold leg piping and the spray line piping. Therefore, the effects of thermal stratification on the pressurizer spray line nozzle resulting from this slight change in T_{COLD} are not negatively impacted by the EPU.

Thermal stratification in the surge nozzle occurs mainly during plant heatup and cooldown transients and is driven by the temperature difference between the hot leg and the pressurizer. Since the hot leg temperature is increasing (by 6.6°F max) due to the EPU as shown in Table 1.1-1, the temperature differential between T_{HOT} and T_{SAT} will decrease (since T_{SAT} is not changing). Therefore, the EPU has no adverse impact on either the thermal stratification or the fatigue analysis of the pressurizer surge nozzle.

Changes in the NSSS Design Transient parameters have also been considered for the pressurizer shell and supports. The rate of change of the RCS temperatures for the various transients are unchanging for the EPU (i.e., the overall shape of the transient curves remains the same for the EPU). Also, as stated in Section 2.2.2.8, NSSS Design Transients, no changes have been made to the allowable number of design cycles defined for each transient. Thus, there are no changes to the NSSS transients that would negatively affect the structural integrity of the pressurizer or supports.

The effects of flow induced vibration were considered for the Pressurizer and supports. Since the pressurizer is isolated from flow of the reactor coolant by the surge line, there is no concern for flow induced vibration (FIV) for the pressurizer or its supports.

Results

The changes to the NSSS operating parameters and design transients have a negligible impact on the pressurizer and supports. Also, the existing design basis seismic analyses are unaffected by the proposed EPU. As such, the pressurizer and supports will experience no increased stress or fatigue usage as a result of the EPU and stress and fatigue usage values will remain below ASME Code allowables. Thus, the structural integrity of the pressurizer and pressurizer supports remain acceptable under the proposed EPU conditions.

2.2.2.7.3 Conclusion

See Section 2.2.2 for Conclusion.

2.2.2.7.4 References

1. BAW-2127, Final Submittal in Response to NRC Bulletin 88-11 “Pressurizer Surge Line Thermal Stratification and Supplement 2 (Rev. 02)”

2.2.2.8 NSSS Design Transients

2.2.2.8.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation.

2.2.2.8.2 Technical Evaluation

Introduction

As discussed in FSAR Section 4.1.2.4, all Reactor Coolant System (RCS) components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Such cyclic loading is the result of normal unit load transients (i.e., normal, upset, emergency and faulted design transients in accordance with ASME III criteria). This evaluation compares the CR-3 design parameters developed for the proposed EPU to the design parameters used in the development of the current design basis transient time histories. Where revisions were necessary, the transient time histories were revised to reflect the operating conditions for the proposed EPU.

Description of Analyses and Evaluations

The CR-3 design parameters for the proposed EPU were compared to the design parameters used in the current design basis transient time histories. Where revisions were necessary due to sufficient differences between the two sets of operating conditions, evaluations and analyses of the existing applicable nuclear steam supply system (NSSS) Design Transient time histories for CR-3 were performed and the time histories were revised, as needed, to reflect the operating conditions for the EPU.

The NSSS Design Transient time histories were based primarily on the NSSS design and operating parameters developed for the proposed EPU (see summary values in Section 1.1, Nuclear Steam Supply System Parameters). In all cases, the transient descriptions and their associated frequencies of occurrence (design cycles) remain unchanged from those in the current design basis transient list and are shown in Table 2.2.2.8-1.

The existing design transient breakpoints of 8% and 15% power for transients 1A, 1B, 2, 3, 4, and 7 are based on a pre-EPU rated core power level of 2609 MWt. The related transient descriptions are shown below. For the proposed EPU (3014 MWt rated core power) the absolute power level of these breakpoints are maintained and are based on 2609 MWt. This results in a post-EPU breakpoint % power designation of 7% and 13% power. The actual operating breakpoints associated with these transients will be greater than the design breakpoints discussed above.

	Pre-EPU Description	Post-EPU Description
1A	Heatup from 70°F to 8% Full Power (Normal)	Heatup from 70°F to 7% Full Power (Normal)
1B	Cooldown from 8% Full Power (Normal)	Cooldown from 7% Full Power (Normal)
2	Power change 0% to 15% and 15% to 0% Power (Normal)	Power change 0% to 13% and 13% to 0% Power (Normal)
3	Power Loading 8% to 100% Power (Normal)	Power Loading 7% to 100% Power (Normal)
4	Power Unloading 100% to 8% Power (Normal)	Power Unloading 100% to 7% Power (Normal)
7	Step Load Reduction, 100% to 8% (Upset)	Step Load Reduction, 100% to 7% (Upset)

Time histories, such as reactor coolant temperatures, OTSG pressure, feedwater flow, etc., associated with several transients presented in Table 2.2.2.8-1 were modified to reflect the EPU conditions.

Results

The revised design transients were used in the NSSS component structural evaluations at EPU conditions. The results of these evaluations are provided for each NSSS component under individual subsections of Section 2.2, Mechanical and Civil Engineering.

Consistent with the current NSSS Design Transient time histories, the revised NSSS Design Transient time histories determined for the proposed EPU are conservative representations of transients that, when used as a basis for component fatigue analyses, provide confidence that the component remains appropriate for its application over the operating license period of CR-3.

2.2.2.8.3 Conclusion

See Section 2.2.2 for Conclusion

2.2.2.8.4 References

None

Table 2.2.2.8-1: CR-3 NSSS Design Transients

Transient	Description	Design Cycles
1A	Heatup from 70°F to 7% Full Power (Normal)	240
1B	Cooldown from 7% Full Power (Normal)	240
2	Power change 0% to 13% and 13% to 0% Power (Normal)	1,440
3	Power Loading 7% to 100% Power (Normal)	18,000
4	Power Unloading 100% to 7% Power (Normal)	18,000
5	10% Step Load Increase (Normal)	8,000
6	10% Step Load Decrease (Normal)	8,000
7	Step Load Reduction, 100% to 7% (Upset) Resulting from Turbine Trip Resulting from Electrical Load Rejection Total	 160 150 310
8	Reactor Trip (Upset) Type A (Corresponding to loss of reactor coolant flow) Type B (Corresponding to high RCS outlet temperature trip) Type C (Corresponding to high RCS pressure trip) Manual Actuation of High Pressure Injection (HPI) system after reactor trip	 40 160 88 11
9	Rapid Depressurization (Upset)	40
10	Change of Reactor Coolant Flow (Upset)	20
11	Rod Withdrawal Accident (Upset)	40
12	Hydrotests (Test) RCS Components except OTSG OTSG Primary Side OTSG Secondary Side	 20 10 10
13	Deleted	-
14	Control Rod Drop (Upset)	40
15	Loss of Station Power (Upset)	40
16	Steam Line Failure (Faulted)	1
17A	Loss of Feedwater to One OTSG (Upset)	20
17B	Stuck Open Turbine Bypass Valve (Emergency)	10
18	Loss of Feedwater heater (Upset)	40
19	Feed and Bleed Operations (Normal)	4,000
20 (1)	Miscellaneous A (Normal) Miscellaneous B Miscellaneous C	30,000 20,000 4×10^6
21	Loss of Coolant (Faulted)	1
22	Test Transients (Normal) HPI System Core Flooding Check Valve	 13 (2) 240
23	OTSG Filling, Draining, Flushing and Cleaning (Normal) OTSG Secondary Side Filling (3) Condition 1 Condition 2 OTSG Primary Side Filling (4) Condition 1 Condition 2 Flushing Chemical Cleaning	 120 120 120 120 40 4
24	Hot Functional Testing (Test)	1

Transient	Description	Design Cycles
25	Refill of Hot, Dry Depressurized OTSG (Upset)	50
26	Emergency Feedwater Actuation (Upset)	1510
27	Reactor Coolant Pump Restart with Voids in the RCS (emergency)	5

(1) **Miscellaneous A** – makeup flow is assumed to drop from normal flow to 1 gpm for 15 minutes, increase to 38 gpm for 5 minutes, and then decrease to normal flow to complete the cycle.

Miscellaneous B – the spray is assumed to be actuated for the cycle.

Miscellaneous C – makeup flow is assumed to drop from normal flow to 6 gpm for 0.5 minutes, increase to 20 gpm for 0.5 minutes, and then decrease to normal flow to complete the cycle.

(2) As of December 2007, CR-3 had logged a maximum of 13 HPI test cycles. Since then, the HPI flow test is performed during refueling outages and per CR-3 test procedures, the reactor vessel head is removed as a prerequisite for performing the test. Therefore, there are no additional transients associated with future HPI test events.

(3) **Secondary Side Filling, definition of Condition 1 and Condition 2**

		Primary Side	**Secondary Side**	**Feedwater (nozzles)**
Condition 1	Temperature	≤200°F	≥ 140°F	50 to 225°F (MFW)
	Pressure	0 – 485 psig	0 psig	
Condition 2	Temperature	≤120°F	≥60°F	50 to 225°F (MFW) 50 to 120°F (EFW)
	Pressure	0 – 485 psig	≈0 psig	

(4) **Primary Side Filling, definition of Condition 1 and Condition 2**

		Primary Fill Water	**Secondary Side**
Condition 1	Temperature	50°F	140°F
	Pressure	0 psig	0 psig
Condition 2	Temperature	140°F	50°F
	Pressure	0 psig	0 psig

2.2.3 Reactor Pressure Vessel Internals and Core Supports

2.2.3.1 Regulatory Evaluation

Reactor Pressure Vessel (RPV) Internals and Core Supports consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. CR-3 reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with loss-of-coolant accidents (LOCAs), and the identification of design transient occurrences. The CR-3 review covered the analyses of flow-induced vibration (FIV) for reactor internal components, as well as the analytical methodologies, assumptions, and computer programs used for these analyses. The RS-001 scope includes a comparison of the resulting stresses and cumulative fatigue usage factors against the corresponding Code-allowable limits.

The NRC's acceptance criteria for were Reactor Pressure Vessel Internals and Core Supports based on:

- 10 CFR 50.55a and GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed;
- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions;
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and
- GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.1, Quality Standards, and 10 CFR 50.55a insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed. [GDC-1 and10 CFR 50.55a]

- FSAR Section 1.4.2, Performance Standards, insofar as it requires that those SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions. [GDC-2]
- FSAR Section 1.4.23, Protection Against Multiple Disability for Protection Systems, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. [GDC-4]
- FSAR Section 1.4.6, Reactor Core Design, insofar as it required that the reactor core be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. [GDC-10]

Additionally, FSAR Section 4.4.1 provides criteria for quality standards insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed in conformance with 10 CFR 50.55a [10 CFR 50.55a] [GDC-1}

2.2.3.2 Technical Evaluation

Introduction

The RPV Internals and Core Supports are described in FSAR Sections 3.1.2.4.1 and 3.2.4.1. The RPV internal components include the Plenum Assembly and the Core Support Assembly. The Core Support Assembly consists of the core support shield, internals vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance holder tubes. The reactor internals are designed to withstand forces due to normal, upset, emergency, and faulted conditions.

As described below, the RPV Internals and Core Supports have been evaluated for thermal, seismic and LOCA loadings associated with the proposed EPU conditions. Additionally, the effects of gamma heating and flow-induced vibration on the internals have been evaluated for the EPU conditions.

Description of Analyses and Evaluations

Changes in the Reactor Coolant System (RCS) operating conditions for the EPU also result in changes to the boundary conditions. These changes include loads and temperatures experienced by the RPV Internals and Core Supports, which could result in changes in the stress levels in these components and changes in the relative displacement between the RPV and the RPV internals. To ensure that the RPV internals maintain their design functions following the EPU, the proposed changes in the RCS operating conditions have been evaluated to assess the impact on the structural integrity of the RPV Internals and Core Supports.

The reactor outlet temperature (T_{HOT}) is increasing 6.6°F (which is the maximum increase given in Table 1.1-1) as a result of EPU. This increase will have a negligible impact on the thermal stresses of the RPV internals. Conservatively applying this temperature increase to all the RPV internals, there will be a negligible change in stresses since the change in thermal expansion is negligible for such a small temperature increase and since the RPV internals are unconstrained and allowed to freely expand.

Further, this small temperature increase will result in a negligible change in clamping forces on the RPV internals and fuel assemblies since these components expand together.

Force differential time-histories for the RPV internals due to loss-of-coolant accident (LOCA) loads resulting from core flood line, surge line and decay heat line breaks have been analyzed for Mark-B-HTP Fuel. That analysis used the largest pressure loss along the Mark-B-HTP fuel assembly for the EPU conditions. There is no change to horizontal core plate motions due to the EPU since seismic motions are unchanging and there is no change in RCS pressure and negligible change to RCS flow as shown in Table 1.1-1 (horizontal core plate motions are driven primarily by differential pressures as a result of LOCA). Therefore, only vertical forces on the core were calculated since no horizontal load-bearing internals are affected by the EPU. The force time histories were used as input into the structural evaluation of the fuel (see Section 2.8.1 Fuel System Design).

The Core Barrel Assembly was assessed for the changes in gamma heating for the proposed EPU conditions. This analysis evaluated the effects of the EPU heating rates upon baffle to former bolts (FB bolts) and core barrel to former bolts (CB bolts). Steady state thermal and structural analyses were performed for nominal operating and design transient gamma heating conditions. Structural analyses accounted for uncertainty in preload and coefficient of friction by repeating analyses for a set of bounding conditions. An approved finite element analysis code ANSYS was used for these analyses.

The RPV internals were also evaluated for the effects of flow-induced vibration (FIV) associated with the EPU. The driving parameter of FIV is the RCS volumetric flow rate. As shown in Table 1.1-1, the RCS volumetric flow rate changes negligibly under the EPU conditions. Thus, the reactor internals and core supports will continue to be qualified for FIV following the EPU.

For discussion on irradiation embrittlement of RPV internals, see Section 2.1.4, Reactor Internal and Core Support Materials.

Results

Stress and fatigue in the RPV Internals and Core Supports due to thermal and seismic loadings is unaffected by the EPU. Further, since horizontal core plate motions and seismic motions are not impacted by the EPU, no horizontal load-bearing internals are impacted by the EPU. Regarding LOCA loadings, the vertical LOCA forces due to breaks in the core flood line, surge line and decay heat line are much less than the large break LOCA forces the RPV internals were previously qualified for in BAW-1621 (Reference 1). Due to application of Leak-Before-Break (LBB), the large break LOCA loads for which the internals were previously qualified are not impacted by the EPU (see Section 2.1.6, Leak-Before-Break, for further discussion on LBB). Thus, the vertical load-bearing bolts of the internals will continue to maintain their structural integrity for the EPU conditions when subject to LOCA loads. Therefore, stress and fatigue on the reactor internals due to thermal, seismic and LOCA loadings were not re-analyzed for the EPU and the pre-EPU design basis remains valid for the EPU conditions.

The results of the analysis of the Core Barrel Assembly for changes in gamma heating as a result of the EPU conditions confirms the same conclusion as the original design basis analysis for the pre-EPU conditions. The concern for bolts not meeting the specified design criteria (as found in the original design basis analysis) continues to be mitigated by the existing inspection frequencies and foreign material exclusion (FME) analysis contained in the EPRI reports referenced below. As inspection frequencies were established based on the original design basis calculations there is no need to increase inspection frequencies. FME has been addressed in EPRI reports MRP-156 (Reference 2) and MRP-157 (Reference 3) for loose parts such as baffle bolt cracking. Since the original design basis conclusions are still applicable there is no reason to revise earlier recommendations on FME. The analysis performed for the EPU conditions demonstrate that stresses and strains in the core barrel assembly are not more severe than the original design before the EPU. These results demonstrate that the overall structural integrity of the core barrel assembly is maintained when subject to the EPU gamma heating loads.

The effects of FIV on RPV internals were evaluated for the EPU conditions. Since RCS flow is essentially unchanging for the EPU, there is no impact to FIV on the RPV internals. Therefore, the RPV Internals and Core Supports will continue to maintain their structural integrity when subject to FIV for the EPU conditions.

2.2.3.3 Conclusion

CR-3 has reviewed the evaluations related to the structural integrity of RPV Internals and Core Supports and concludes that the evaluations have adequately addressed the effects of the proposed EPU on the RPV Internals and Core Supports. CR-3 further concludes that the evaluations demonstrate that the RPV Internals and Core Supports will continue to meet the requirements of 10 CFR 50.55a and FSAR Sections 1.4.1, 1.4.2, 1.4.6, and 1.4.23. Therefore, CR-3 finds the proposed EPU acceptable with respect to the design of the RPV Internals and Core Supports.

2.2.3.4 References

1. BAW-1621, B&W 177-FA Owners Group, “Effects of Asymmetric LOCA Loadings, Phase II Analysis”

2. EPRI Report MRP-156, “Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure,” December 2005

3. EPRI Report MRP-157, “Materials Reliability Program: Updated B&W Design Information for the Issue Management Tables,” October 2005

2.2.4 Safety Related Valves and Pumps

2.2.4.1 Regulatory Evaluation

The CR-3 review included certain Safety-Related Pumps and Valves typically designated as Class 1, 2, or 3 under Section III of the ASME B&PV Code and within the scope of Section XI of the ASME B&PV Code and the ASME Operations and Maintenance (O&M) Code, as applicable. The CR-3 review of Safety Related Pumps and Valves focused on the effects of the proposed EPU on the required functional performance of the valves and pumps.

The NRC's acceptance criteria for Safety Related Valves and Pumps are based on:

- GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed;

- GDC-37, 40, 43, and 46, insofar as they require that the ECCS, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components;

- GDC 54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and

- 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section.

Specific criteria of NRC Generic Letter (GL) 89-10, Motor-Operated Valve Testing and Surveillance, GL 96-05, Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves, and GL 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, are also included in this review.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are applicable CR-3 specific criteria:

- FSAR Section 1.4.1, Quality Standards, insofar as they require SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed [GDC-1];

- FSAR Sections 1.4.38, Reliability and Testability of Engineered Safety Features, 1.4.46, Testing of Emergency Core Cooling System Components, 1.4.47, Testing of Emergency Core Cooling

Systems, 1.4.48, Testing of Operation Sequence of Emergency Core Cooling Systems, 1.4.59, Testing of Containment Pressure-Reducing System Components, 1.4.60, Testing of Containment Spray Systems, 1.4.61, Testing of Operational Sequence of Containment Pressure-Reducing Systems, 1.4.63, Testing of Air Cleanup Systems Components, 1.4.64, Testing of Air Cleanup Systems, and 1.4.65, Testing of Operational Sequence of Air Cleanup Systems, insofar as they require that the ECCS, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components [GDC-37, 40, 43, and 46]; and

- FSAR Sections 1.4.56, Provisions for Testing of Penetrations, and 1.4.57, Provisions for Testing of Isolation Valves, insofar as they require that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits [GDC-54].

Additionally, FSAR Section 1.4.19 provides criteria for Protection Systems Reliability insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section in conformance with 10 CFR 50.55a(f). CR-3 implementation of the requirements of GL 89-10 was evaluated and accepted by the NRC as cited in NRC to FPC Response Letter 3N1195-09, dated November 13, 1995. CR-3 participation in the Joint Owners Group (JOG) program, thereby implementing the requirements of GL 96-05, was evaluated and accepted by the NRC (Reference 1 and 2). Additionally, CR-3 implementation of the requirements of GL 95-07 was evaluated and accepted by the NRC as cited in NRC to FPC Response Letter 3N1097-27, dated October 16, 1997.

2.2.4.2 Technical Evaluation

Introduction

Safety Related Pumps and Valves are maintained under strict guidelines to ensure that they will function in all operating and accident conditions. Programs are in place to maintain that all safety measures and testing criteria are being met under the EPU conditions, which have been analyzed and meet the NRC criteria for Safety Related Pumps and Valves.

Safety-related motor operated valves (MOVs), solenoid operated valves (SOVs), air operated valves (AOVs), check valves, relief valves, and pumps were evaluated.

Description of Analyses and Evaluations

The following addresses the impact of the EPU on performance requirements of CR-3 Safety-Related Pumps and Valves, including impact on the Inservice Testing Program. If not discussed below, no adverse impact was found to the pumps or valves. Safety related pumps were reviewed and evaluated for their continued adequate performance after the implementation of the EPU as part of each safety related section of this LAR. Acceptable system performance demonstrates that no modifications to any existing safety related pumps are required by the EPU.

Continued acceptable performance of Safety-Related Pumps and Valves will be assured through the Inservice Testing Program.

Emergency Feedwater (EFW)

EFW flow needs to be increased roughly in proportion to decay heat for EPU conditions. To supply this higher flow the current continuously in-service recirculation flow paths will be modified to include new safety-related automatic recirculation valves (EFV-179 and EFV-180). In order to increase the flow sufficiently, the recirculation flow will be isolated when the EFW pumps are automatically actuated and flow reaches the higher setpoint or on shutdown when flow returns to an appropriate lower setpoint. Changes are reflected in the Inservice Testing Program.

Decay Heat System (Low Pressure Injection (LPI))

Evaluations show that after modifications to the Decay Heat (DH) System, the DH system pumps continue to perform acceptably under the EPU conditions (refer to Sections 2.8.4.4 (Residual Heat Removal System), 2.8.5.6.3 (Emergency Core Cooling System and Loss of Coolant Accidents) and Appendix E (Major Plant Modifications)). As part of the modifications to the DH system, MOVs DHV 210 and 211 are removed from the system and replaced with check valves. Two new safety-related MOVs are added to the system in the Boron Precipitation line (DHV 514 and DHV 614). The new check valves and MOVs will be designed to, and installed to support the EPU conditions. Changes are reflected in the Inservice Testing Program.

Two new LPI check valves (DHV-611 and DHV-612) will be installed to prevent reverse flow into the LPI headers and provide a location for the ASME Section XI, Class 1/Class 2 boundary. These valves will be RCS pressure isolation valves and will be addressed by Improved Technical Specification 3.4.13 and associated Bases. DHV-611 also will be credited as the inboard containment isolation valve (CIV) for the BP line. Changes are reflected in the Inservice Testing Program.

Main Steam System

The EPU requires the installation of a fast cooldown system using larger capacity safety-grade atmospheric dump valves (ADVs). The replacement ADVs and associated isolation valves are sufficiently sized to satisfy all functional requirements of the ADVs at the EPU conditions and to support fast cooldown of the RCS by reducing secondary side steam pressure for certain events. The valves being replaced are Main Steam Valve (MSV) 25 and MSV 26 (ADVs), as well as MSV 27 and MSV 28 (the associated isolation valves). Additionally, safety-related control air system is provided with associated safety-related system valves, which will also be reflected in the Inservice Testing Program.

More detail about the ADV modification is found in Appendix E.

Main Feedwater System

As discussed in Section 2.5.5.4, Condensate and Feedwater, the EPU requires modifications to the Main Feedwater (MF) System. The valves in the MF System that are impacted by the EPU include Feedwater Valves (FWV) 14, 15, and 29. FWV 14 and FWV 15 will be replaced. The FWV 14 and FWV 15 replacements are designed to the EPU parameters. Refer to Appendix E for additional details. The increased discharge head does not adversely impact FWV 29 such that valve modification is required. However, the thrust margin for FWV 29 will decrease and result in the valve being re-classified for testing in accordance with the JOG program requirements. Closure time requirement for FWV 28, 29, and 30 will be decreased from 34 seconds to 31 seconds based on analysis. FWV 31, 32, 33, and 36 will also experience increases in differential pressure due to the discharge head increase. However, the functional performance of these MOVs are not impacted with regard to stroke time, pressure locking/thermal binding and weak link. Changes to MF valves are reflected in the Inservice Testing Program.

Motor-Operated Valve Program

The intent of the CR-3 MOV program identifies the maximum differential pressure (dP) for which a MOV shall operate against to perform its required safety function during all conditions in compliance with GL 89-10, GL 95-07, and GL 96-05. Once the maximum dP has been established, minimum thrust requirements at accident conditions, motor overloads and reduced voltage must be considered and factored into the MOV actuator specific thrust calculations to determine if a MOV will perform adequately. Additionally, pressure locking and thermal binding of gate valves must be included if applicable. CR-3 MOV program implementation ensures that changes to MOVs as a result of the EPU will be processed through the MOV program as applicable. The EPU will not change the CR-3 MOV program.

The impact of the EPU on AC and DC Onsite Power Systems will not affect the GL 89-10 motor-operated valves. There is no reduction in voltage for both the AC and DC electrical MOV systems. Calculations reviewing the stroke times for MOVs predict that the stroke times will increase minimally, therefore the impact to stroke time is negligible.

An evaluation of the EPU impact was conducted on the existing analyses performed to show that motor-operated valve actuators have sufficient thrust to overcome the pressure-locked bonnet condition. This evaluation concluded that the EPU does not create any new conditions which would affect susceptibility of valves to pressure locking or thermal binding.

Tables 2.2.4-1 and 2.2.4-2 show MOVs which were found to have post-EPU increases in differential pressure and were reviewed for thrust and weak link susceptibilities. As can be seen from these tables, the MOVs are acceptable as-is for post-EPU conditions relevant to MOV thrust capability and weak link integrity. CR-3 MOV Program implementation is adequate for addressing changes as a result of the EPU. Therefore the EPU has no impact on the requirements of GL 95-07.

MOVs DHV 210 and DHV 211 will be replaced with check valves, DHV 510 and DHV 610, respectively. DHV-210 and DHV-211 are currently used to throttle flow during quarterly DH/LPI pump testing and when the DH pumps are used to cool the spent fuel pool (SFP). These valves are being replaced due to their inherently high loss factors and their throttling function will be accomplished with DHV 9 and DHV 48. Details related to the DH/LPI modification are described in Appendix E.

Implementation of the EPU will not change the requirements of the MOV tracking and trending program. The tracking and trending portion of the MOV program is acceptable to address changes in MOV performance requirements for the EPU conditions. Therefore, the EPU conditions do not affect the requirements of the Program for periodic verification of safety-related motor-operated valve capabilities in accordance with GL 96-05.

Results

The valves modified to support the EPU conditions in the DH System include DHV-210 and 211 which are being replaced with check valves DHV 510 and 610. Two new MOVs DHV 514 and 614 are within the scope of the MOV program to meet the EPU conditions.

ADVs and their associated isolation valves are being replaced with safety-related valves that meet EPU parameters. The valves being replaced are MSV 25, 26, 27 and 28. Safety-related control air system valves are also being added to support the ADVs.

FWV 14 and 15 are being replaced with new MOVs that meet the EPU parameters in the FW System. FWV 29 is capable of performing its function at the EPU conditions without modification but is reclassified for testing in accordance with JOG program requirements due to a decrease in the thrust margin. Closure time requirements for FWV 28, 29, and 30 will be reduced. All other Safety-Related Pumps and Valves are not adversely impacted by EPU parameters.

MOV Program implementation is adequate for addressing changes as a result of the EPU and therefore the EPU has no impact on the CR-3 MOV Program.

2.2.4.3 Conclusion

CR-3 has reviewed the assessments related to the functional performance of Safety-Related Valves and Pumps and concludes that the effects of the proposed EPU on Safety-Related Pumps and Valves have been adequately addressed. CR-3 further concludes that the effects of the proposed EPU on motor-operated valve programs related to GL 89-10, GL 96-05, and GL 95-07 have been adequately evaluated. Based on this, CR-3 concludes that it has been demonstrated that Safety-Related Valves and Pumps will continue to meet CR-3 current licensing basis with respect to the requirements of FSAR 1.4.1, 1.4.38, 1.4.46, 1.4.47, 1.4.48, 1.4.56, 1.4.57, 1.4.59, 1.4.60, 1.4.61, 1.4.63, 1.4.64, and 1.4.65.following implementation of the proposed EPU. Therefore, the proposed EPU is acceptable with respect to Safety-Related Valves and Pumps.

2.2.4.4 References

1. FPC to NRC Letter 3F1200-03, dated December 21, 2000.
2. NRC to FPC Letter 3N0301-04, dated March 13, 2001.
3. NUREG-1482, “Guidelines for Inservice Testing at Nuclear Power Plants”, April 1995.

Table 2.2.4-1: MOV Thrust Values

MOV	Pre-EPU		Post-EPU		
	Minimum Required Thrust (lbf)	Minimum Required Thrust + ROL (lbf)	Minimum Required Thrust (lbf)	Minimum Required Thrust + ROL (lbf)	20% Margin
FWV-29	60600.65	71396.68	62792.47	73978.98	88774.78
FWV-30	60600.65	71396.68	62792.47	73978.98	88774.78
FWV-31	25001.85	29455.94	25873.22	30482.55	36579.06
FWV-32	25001.85	29455.94	25873.22	30482.55	36579.06
FWV-33	12814.73	15097.68	13205.66	15558.25	18669.90
FWV-36	12814.73	15097.68	13205.66	15558.25	18669.90

Table 2.2.4-2: Post-EPU Thrust and Weak Link Values

MOV	Minimum Required Thrust (20% margin) (lbf)	Stem Factor (ft)	Max Capability of Actuator (ft-lbf)	Max Capability of Actuator (lbf)	Torque Switch Trip Setpoint (TST) (AS LEFT TEST) (lbf)	Weak Link (lbf)
FWV-29	88774.78	0.0197	1723.75	87500.00	95827	192886
FWV-30	88774.78	0.0197	1892.76	96079.19	110943	192886
FWV-31	36579.06	0.0154	614.81	39922.73	41223	73643
FWV-32	36579.06	0.0154	578.05	37535.71	47339	73643
FWV-33	18669.90	0.0140	443.91	31707.86	21787	25400
FWV-36	18669.90	0.0140	431.73	30837.86	20218	25400

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

2.2.5.1 Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant release of radioactive materials to the environment are also covered by this section. The CR-3 review focused on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated with pipe whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by the EPU.

The NRC's acceptance criteria for Seismic and Dynamic Qualification of Mechanical and Electrical Equipment are based on:

- 10 CFR Part 50, Appendix B, which sets quality assurance requirements for safety-related equipment;
- 10 CFR Part 100, Appendix A, which sets forth the principal seismic and geological considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site;
- GDC-1, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed;
- GDC-2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions;
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;
- GDC-14, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture; and
- GDC-30, insofar as it requires that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The applicability of the above criteria is discussed below.

The CR-3 acceptance criteria are based on:

- FSAR Sections 1.4.1, Quality Standards, insofar as it requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. [GDC-1];
- FSAR Section 1.4.2, Performance Standards, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions. [GDC-2];
- FSAR Section 1.4.23, Protection Against Multiple Disability for Protection Systems, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. [GDC-4];
- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of rapidly propagating fracture. [GDC-14]; and
- FSAR Sections 1.4.16, Monitoring Reactor Coolant Pressure Boundary, and 1.4.33, Reactor Coolant Pressure Boundary Capability, insofar as they require that components that are part of the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. [GDC-30].

Additionally, FSAR Sections 1.6 Quality Program (Preoperational) and 1.7 Quality Program (Operational), provide criteria for quality assurance requirements for safety-related equipment. [10 CFR Part 50, Appendix B];

Additionally, FSAR Sections 2.1, Summary, and 2.2, Site and Adjacent Areas, and 2.5, Engineering Geology and Foundation Considerations, provide criteria for the principal seismic and geological considerations for the evaluation of the suitability of plant design bases established in consideration of the seismic and geologic characteristics of the plant site. [10 CFR Part 100, Appendix A].

2.2.5.2 Technical Evaluation

Introduction

Safety-related SSCs at CR-3 are designed for both seismic and dynamic events as described in FSAR Chapter 5. The seismic design is not impacted by the EPU since seismic requirements remain unchanged. There is no change to seismic inputs (amplified response spectra) or loads resulting from the EPU. The existing seismic design basis for piping and supports remains valid and unaffected by the EPU. Hence, piping and support seismic loadings will continue to meet the current licensing basis. Dynamic qualification can be impacted if equipment operating conditions such as pressure, temperature, and fluid flow change as a result of the EPU. Additionally, ability of the equipment to withstand the effects of pipe-whip, and jet impingement may also be affected as a result of the EPU impact on systems in physical proximity of essential safety-related equipment.

Description of Analyses and Evaluations

Dynamic effects of internally and externally generated missiles under the EPU have been evaluated and are addressed in Section 2.5.1.2, Missile Protection. Dynamic effects of pipe-whip and jet impingement under the EPU conditions have been evaluated and are addressed in Sections 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, and 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems. Based on these evaluations, the EPU will have no adverse impact on essential equipment as a result of pipe whip, jet impingement, internal and external missiles.

Other evaluations related to the dynamic and environmental effects of the EPU are addressed in Sections 2.2.2.1, NSSS Piping, Components and Supports, 2.2.2.2, BOP Piping, Components and Supports, and 2.3.1, Environmental Qualification of Electrical Equipment.

The evaluation of mechanical and electrical equipment for the proposed EPU concluded that safety-related equipment will continue to be protected from seismic and dynamic events, and will continue to meet the CR-3 current licensing basis.

2.2.5.3 Conclusion

CR-3 reviewed the evaluations of the effects of the proposed EPU on the Seismic and Dynamic Qualification of Mechanical and Electrical Equipment and concludes that the review has 1) adequately addressed the effects of the proposed EPU on equipment and 2) demonstrated that the equipment will continue to meet the requirements of FSAR Sections 1.4.1, 1.4.2, 1.4.9, 1.4.16, 1.4.23, 1.4.33, 10 CFR Part 100, Appendix A, and 10 CFR Part 50, Appendix B, following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Seismic and Dynamic Qualification of the Mechanical and Electrical Equipment.

2.2.5.4 References

None.

2.2.6 Incore Instrumentation Guide Tubes

2.2.6.1 Regulatory Evaluation

See Section 2.2.2 for Regulatory Evaluation.

2.2.6.2 Technical Evaluation

Introduction

The Incore Instrumentation Guide Tubes penetrate the bottom head of the reactor vessel (RV) at the incore instrument nozzles, which are installed with partial penetration welds. The guide tubing is routed from the RV bottom head to a shielded area in the reactor building where the incore detector assemblies can be inserted or withdrawn through the guide tubes while the reactor is depressurized. As such, the pressure barrier between the RCS and containment atmosphere with respect to incore instrumentation consists of the guide tubes, the incore instrumentation nozzles (addressed in Section 2.2.2.3, Reactor Vessel and Supports), and the incore closure assembly at the incore instrument tank. The non-pressure boundary guide tubes internal to the reactor vessel are addressed in Section 2.2.3, Reactor Vessel Internals and Core Supports.

Description of Analyses and Evaluations

This section addresses the structural integrity of the pressure-retaining incore guide tubes and incore closure assembly for the changes in RCS conditions (pressure and temperature) due to the EPU. For discussion of the effects of primary water stress corrosion cracking (PWSCC) on Alloy 600/182/82 nickel base alloys of reactor coolant pressure boundary (RCPB) materials, see Section 2.1.5, Reactor Coolant Pressure Boundary Materials.

Since the RCS component weights and seismic loadings are unchanged for the EPU, the stresses of the Incore Instrumentation Guide Tubing and incore closure assembly due to deadweight, operating basis earthquake (OBE), and safe shutdown earthquake (SSE) loadings remain unchanged for the EPU. See 2.2.5, Seismic and Dynamic Qualification of Mechanical and Electrical Equipment, for additional seismic discussion.

Based on the remote location of the incore closure assembly relative to the RV, the change in RCS conditions due to the EPU has a negligible impact on its structural integrity.

However, two aspects of incore guide tubing structural integrity could be potentially affected by the EPU changes. They are:

1) Change in RCS pressure during normal operating conditions or transients

2) Change in RCS core inlet temperature (T_{COLD}) during normal operating conditions or transients

The EPU design conditions were evaluated for impact on the above two items which could potentially affect the Incore Instrumentation Guide Tubing. As stated in Section 1.0, there is no increase in normal operating pressure due to the EPU. Also, the changes to NSSS Design Transients (discussed in Section 2.2.2.8) have been reviewed for the EPU conditions and it has been confirmed that there is no increase in

RCS pressure during transients compared to the pre-EPU design transients. Thus, regarding RCS pressure, there is no impact on the incore guide tubing as a result of the EPU.

Regarding temperature changes, it is shown in Table 1.1-1 that T_{COLD} is decreasing by less than 1°F for the EPU. Since T_{COLD} is decreasing under the EPU conditions, there is no increase in thermal expansion loading of the incore guide tubing during normal operation when compared to the pre-EPU conditions. Thus, the guide tubing is not impacted by any normal operating temperature changes as part of the EPU. The changes in NSSS Design Transients (discussed in Section 2.2.2.8) also must be evaluated for the EPU since incore guide tubing stress/fatigue can be affected by changes to any of the following characteristics of the design basis transients: (1) rate of core inlet temperature change vs. time, (2) peak temperature or (3) allowable number of design cycles. Review of the NSSS design transients confirmed the rate of change of the RCS core inlet temperature for the various transients are unchanging for the EPU (i.e., the overall shape of the transient curves remains the same for the EPU). Also, peak temperatures are changing negligibly such that there is no increase in thermal expansion loading or Code stress allowables. Also, no changes have been made to the allowable number of design cycles defined for each transient as discussed in 2.2.2.8. Thus there is no impact on the Incore Instrumentation Guide Tubing as a result of the EPU with respect to RCS temperature changes during normal or transient conditions.

Results

Based on no changes to seismic loading and negligibly changing RCS pressure and temperatures during normal operation or transient conditions, the stress margins and fatigue usage calculated as part of the existing design basis for the Incore Instrumentation Guide Tubing and incore closure assembly are unchanging as a result of the EPU.

2.2.6.3 Conclusion

See Section 2.2.2 for Conclusion

2.2.6.4 References

None.

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

2.3.1.1 Regulatory Evaluation

Environmental Qualification (EQ) of Electrical Equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from DBAs. The CR-3 review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The CR-3 review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU.

The NRC's acceptance criteria for Environmental Qualification of Electrical Equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety which is located in a harsh environment.

CR-3 Current Licensing Basis

CR-3 originally qualified electric equipment important to safety in accordance with "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines) based on its date of first commercial operation (March, 1976), and continues to use these criteria for qualifying electric equipment important to safety, with the exception of replacement equipment as allowed by 10 CFR 50.49(k) and Regulatory Guide 1.89. Thus, at CR-3, electric equipment important to safety is environmentally qualified in accordance with the DOR Guidelines per IEEE-323-1971 as originally licensed. New or replacement equipment installed after February 23, 1983, has been environmentally qualified in accordance with the provisions of Regulatory Guide 1.89, Revision 1 and IEEE 323-1974.

2.3.1.2 Technical Evaluation

Introduction

The following are documents or program elements necessary to demonstrate compliance with 10 CFR 50.49:

- The EQ Equipment Master List (EQML) is maintained electronically in the CR-3 Equipment Database (EDB).
- The EQ Plant Profile Document (EQPPD) defines the normal and post-accident environmental parameters (e.g., temperature, radiation, flooding) for each plant area (or zone).
- Vendor Qualification Packages (VQPs) provide the evaluation and analysis which demonstrates that the "as tested" qualified configuration bounds the CR-3 "installed" configuration with appropriate margin.

As described in the CR-3 EQPPD, the environmental threshold categories are defined as follows:

- MILD – an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation (including anticipated operational occurrences (AOOs)) as a result of a Design Basis Accident (DBA).
- HARSH – an area which is postulated to become significantly more severe than that experienced during normal plant operation (including AOOs) as a result of a DBA.
- HARSH – RAD ONLY – an area which will become significantly more severe (radiation only) due to increased radiation levels ≥ 1.0 E+5 rads.
- HARSH RAD ONLY – FOR ELECTRONICS ONLY – an area which will become significantly more severe (radiation only) as a result of increased radiation levels ≥ 1.0 E+4 rads.

The environmental parameters evaluated for EPU conditions include temperature, pressure, radiation dose, submergence, chemical spray effects, and humidity, for both normal operation and post-accident conditions.

Description of Analyses and Evaluations

As part of the CR-3 Design Control process, EQ impacts from EPU conditions (both normal operation and post-accident) are evaluated to support component selection and design, as well as to identify any appropriate maintenance activities (replacement, periodic refurbishment, etc.). Even though the EPU design change packages are not yet issued for construction, EPU impacts on parameters important to EQ are known. The CR-3 EQ/Design staff evaluated the information currently available to identify and report herein any EQ impacts based upon EPU impacts on parameters important to EQ.

An independent assessment of EPU impacts on EQ was obtained using a third party with extensive EPU and EQ experience. The EPU environmental results were compared to current EQ qualification levels, as defined in the existing EQ qualification packages. The evaluation identified impacts to existing EQ component qualification and to components being added to the EQ Program as a result of these EPU environmental condition changes.

EPU Impact on EQ Program

EQ programs are based on the impact of plant conditions (both normal operation and post-accident) on components which are required to support the mitigation and/or monitoring of the appropriate DBA. EQ qualification of components currently qualified, and for those which are required to become EQ qualified as a result of projected EPU conditions or plant modifications, are based on the limiting environmental parameters for normal operation and post-accident conditions as summarized below.

The increase in reactor power at EPU conditions does not result in a significant increase in temperature, pressure, or humidity, which are generally a function of the energy in the failed piping systems (Reactor Coolant System, Main Steam or Feedwater). The energy released is presented in 2.6.3.1, Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents, and resultant impacts on temperature, pressure, or humidity for the containment are presented in Section 2.6.1, Primary Containment Functional Design. For outside containment, the high energy line break (HELB) results are presented in Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.

The most significantly impacted plant environmental parameter is an increase in the radiation dose levels based upon updated source terms. Various equipment (e.g., primarily piping, tanks, and filters) contains potentially contaminated fluids (both normal and post accident). Radiation dose levels associated with this equipment are impacted by increased source term. The source term increases are a result of two factors: (1) the increase in core power and (2) an increase in the number of isotopes identified due to utilizing more current source term methods. Inside containment, the doses from liquid and airborne activity includes beta contribution.

The radiation dose in a particular zone is calculated by integrating the dose contributions from all potential sources (e.g., pipe, tank, adjacent building, etc.) and accounting for shielding and distance. The zone dose is the highest calculated dose anywhere in that particular zone. Point-specific doses at specific component locations are calculated, as needed, to demonstrate a reduced dose over that of a more conservative zone dose for the purposes of demonstrating qualification of individual components at that precise location. CR-3 utilizes a three dimensional computer plant model to support both the zone and point-specific calculations.

- The normal operating dose was conservatively calculated over 60 years to account for the potential duration of extended operation as discussed in Section 2.14, The Effects of EPU on the Renewed Licensing and License Renewal Programs. The normal operating dose is based upon a conservative methodology. The projected operating dose was calculated by taking the current operating doses and adjusting them by the increase from accident dose levels.

- The 60 year normal operating dose over the life of the component is combined with post-accident dose that exists for the post-accident mission time. The combined dose (referred to as Total Integrated Dose (TID) is used as the qualification design basis of the electrical equipment important to safety. A thorough review of both normal and post-accident conditions in each zone was performed. The revised TID in each zone was compared to the existing qualification value and used as the key input to the EPU EQ impact evaluation.

- Point-specific component calculations have been performed to support the qualification of individual components adversely impacted by the increase in maximum zone radiation dose.

Summary EPU Impact (by Building)

Reactor Building (RB)

The projected post-EPU accident radiation dose increase is 16% inside containment.

Loss of Coolant Accident (LOCA) mass and energy releases and resulting temperature conditions remain within the current EQ bounding profile as indicated in Figure 2.3.1-1. This break is considered to be the most limiting HELB condition inside containment. The original CR-3 Safety Evaluation Report (SER) for Environmental Qualification verified that the Large-Break Loss-Of-Coolant-Accident (LBLOCA) environmental conditions are more severe than the Main Steam Line Break (MSLB) environmental conditions based on the availability of redundant building spray trains not subject to disabling single component failures. This position is consistent with NUREG-0458, “Short Term Safety Assessment on the Environmental Qualification of Safety-Related Electrical Equipment of Systematic Evaluation Program (SEP) Operating Reactors.

The RB postulated maximum flood level is not impacted by the proposed EPU conditions.

There is no increase in boric acid concentrations beyond the current EQ upper range of 3000 ppm and any changes to humidity levels as a result of EPU do not adversely impact the assumed current values of 20 – 100%.

Intermediate Building (IB)

The projected post-EPU accident radiation average dose increase is 36% inside the Intermediate Building.

HELB mass and energy releases and resulting temperature conditions remain bounded by the current EQ bounding profile as indicated in Figure 2.3.1-2.

The IB postulated maximum flood level is not impacted by the proposed EPU conditions. There are no significant sources of additional fluid inventory in the IB as discussed in Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.

There are no significant changes in humidity levels from the assumed current values of 20 – 100%.

Auxiliary Building

The projected post-EPU accident radiation average dose increase is 39%. There are no changes to temperature and pressure environmental conditions in the Auxiliary Building at EPU conditions as discussed in Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.

Control Complex and Turbine Building

The impacts to the environmental conditions in the Control Complex and/or Turbine Building as a result of EPU are not significant enough to reclassify these areas from ‘mild’ to ‘harsh.’ Therefore, the EQ status of electrical equipment important to safety in these areas is unchanged.

Detailed EPU Impacts by Zone

The following plant areas (i.e.,CR-3 EQ zones - refer to Figures 2.3.1-3 through 2.3.1-6) require EQ reclassification due to projected post-EPU environmental conditions.

- The Intermediate Building, Zone 17, is reclassified from “ HARSH RAD ONLY – FOR ELECTRONICS ONLY” environment to “ HARSH – RAD ONLY.”
- The Auxiliary Building, Zones 23 and 47, are reclassified from “MILD” to “HARSH RAD ONLY – FOR ELECTRONICS ONLY” area.
- The Auxiliary Building Zones 18, 60, and 62 are reclassified from “HARSH RAD ONLY – FOR ELECTRONICS ONLY” to “HARSH – RAD ONLY” area.

- The Auxiliary Building Zone 78 is reclassified from “MILD” to “HARSH RAD ONLY – FOR ELECTRONICS ONLY.”

Summary of EPU Impact on Component Basis

Based on the environmental conditions for EPU, all equipment currently in the EQ Program will remain qualified.

Proposed plant changes as a result of EPU (refer to Appendix E) which add or modify equipment subject to EQ requirements (based on location and functions) have been reviewed for inclusion in the EQ Program. Based upon a review of these proposed EPU plant modifications, the following new EQ components will be added to the EQ Program. These new components will be designed, procured, and installed in compliance with appropriate EQ requirements. The details of assuring compliance will be addressed as an integral and routine part of the CR-3 engineering change process.

- The Atmospheric Dump Valve (ADV)/Fast Cooldown System (FCS) design change will add new transmitters and transducer I/P to the EQ program.
- The Low Pressure Injection Cross Tie and Hot Leg Injection design change will add new EQ transmitters and motor operator valves.
- The Feedwater Booster Pump Modification design change replaces safety related EQ motor operated valves with new EQ motor operator valves.
- The Emergency Feedwater Recirculation Instrumentation design change adds new safety-related EQ differential pressure indicating switches.
- The Inadequate Core Cooling Monitor system design change will add new transmitters and classification change to EQ core exit thermocouple.

2.3.1.3 Conclusion

CR-3 has evaluated the effects of the proposed EPU on the EQ of Electrical Equipment and concludes that the effects of the proposed EPU on the environmental conditions for and the qualification of electrical equipment have been adequately addressed. CR-3 further concludes that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Environmental Qualification of Electrical Equipment.

2.3.1.4 References

None

Figure 2.3.1-1
Post-EPU LOCA vs. EQ Bouding Profile Comparison
(Inside Containment)

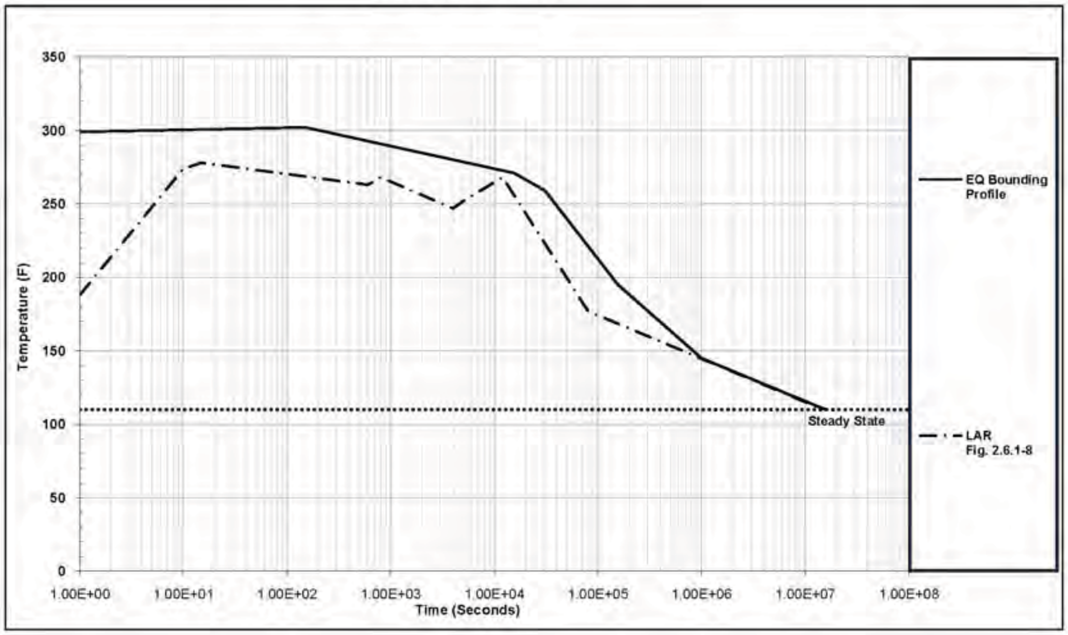
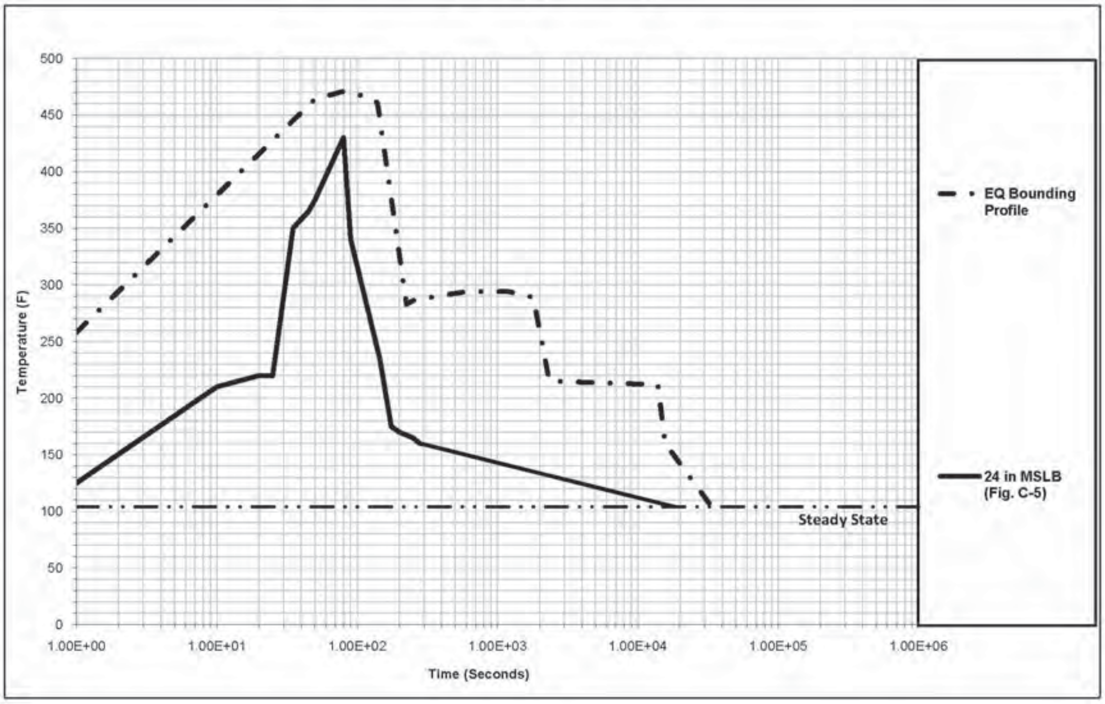


Figure 2.3.1-2
Post-EPU Bounding HELB vs. EQ Bounding Profile Comparison
(Outside Containment)



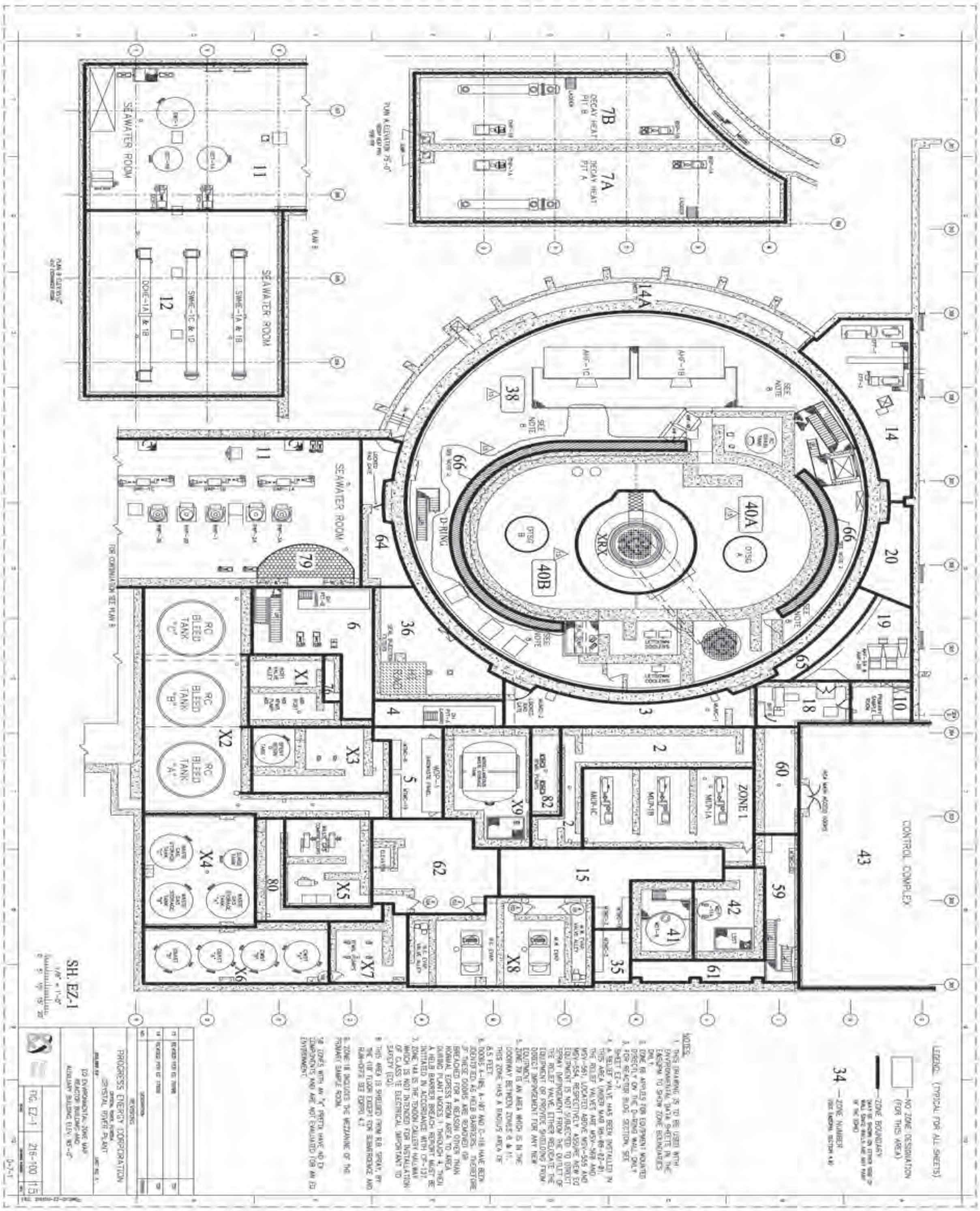


Figure 2.3.1-3

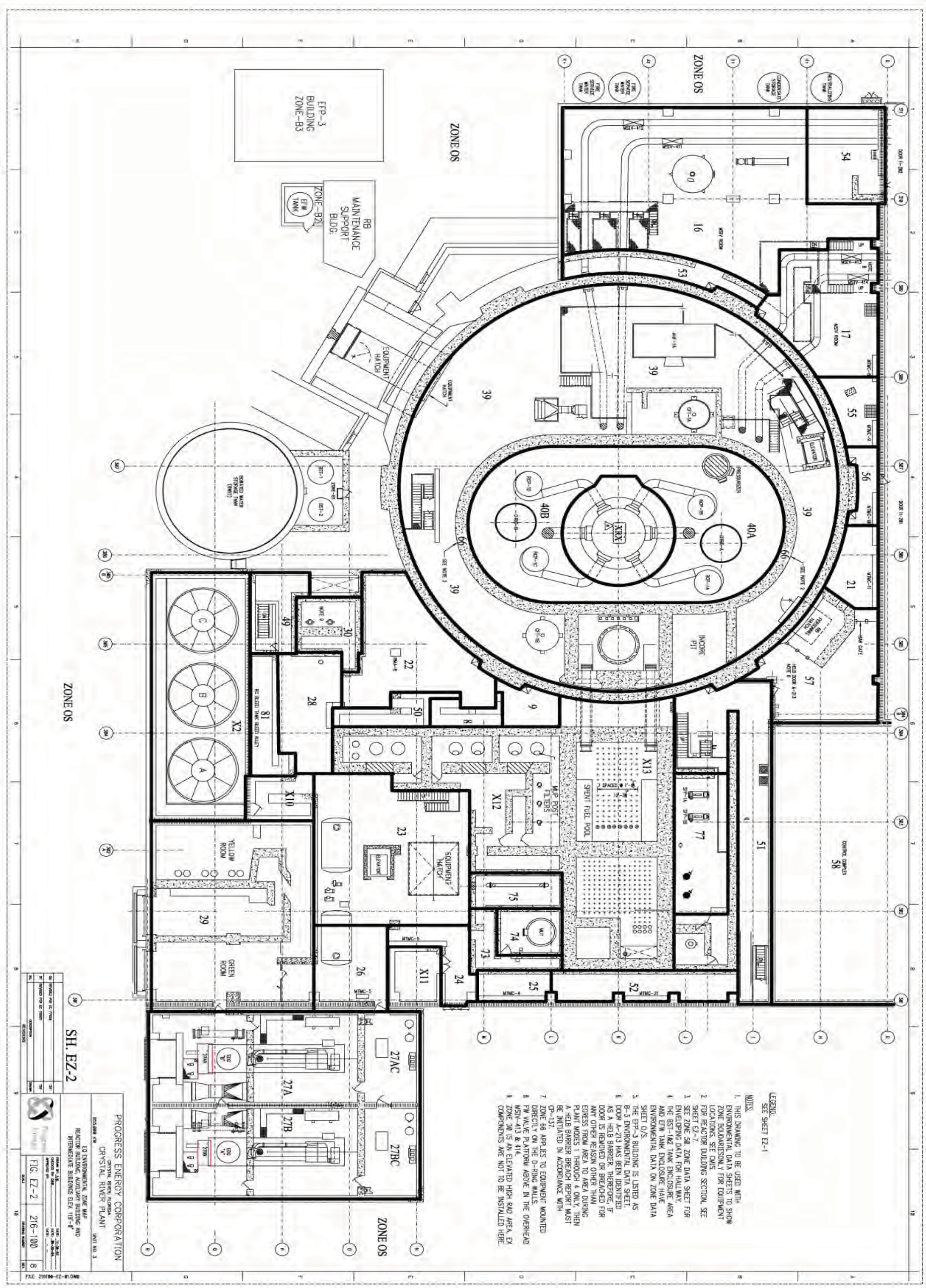


Figure 2.3.1-4

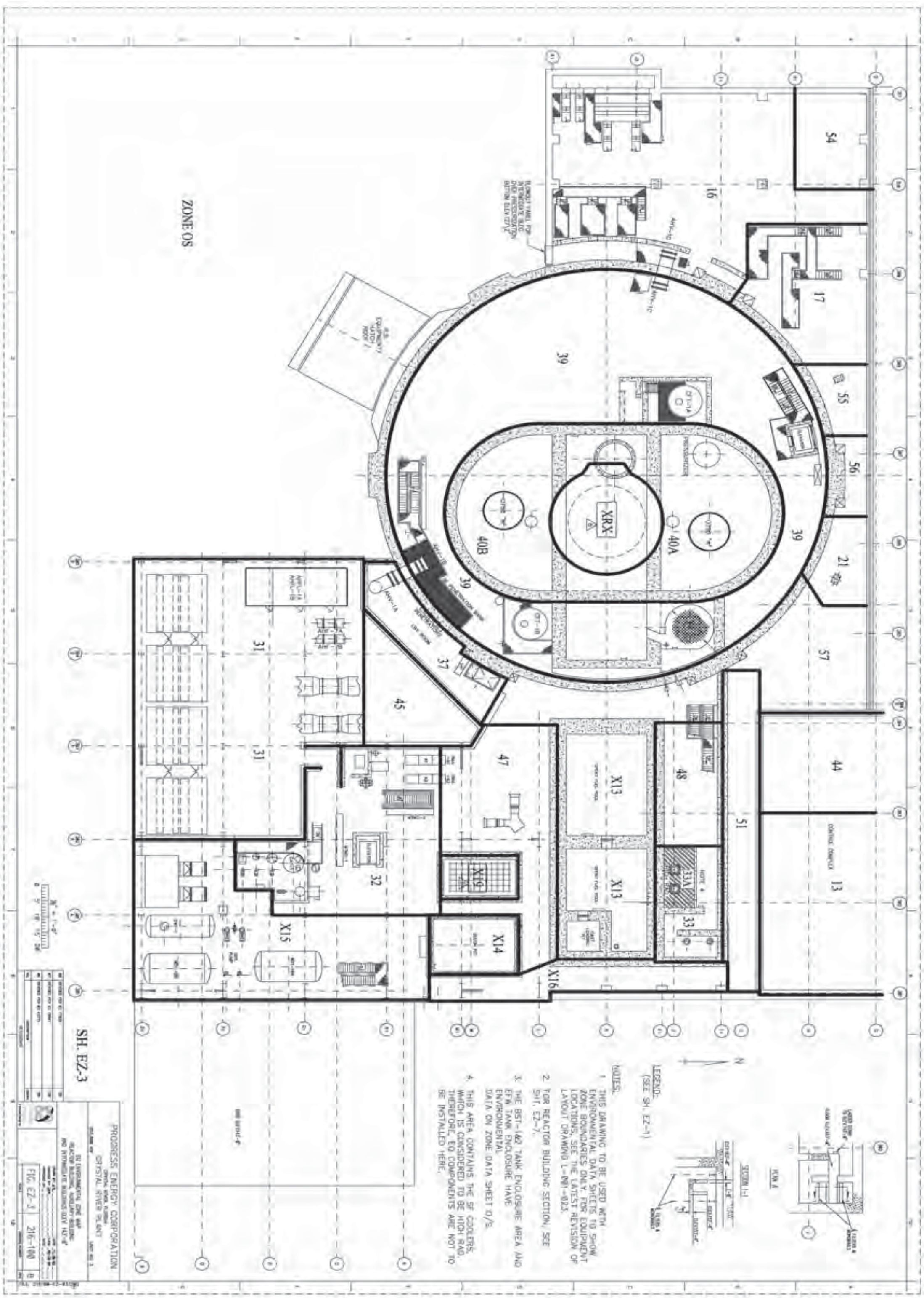

ZONE OS
SH. EZ-3
PROGRESS ENERGY CORPORATION
XRX
X13
X14
X15
X16
X19

Figure 2.3.1-5

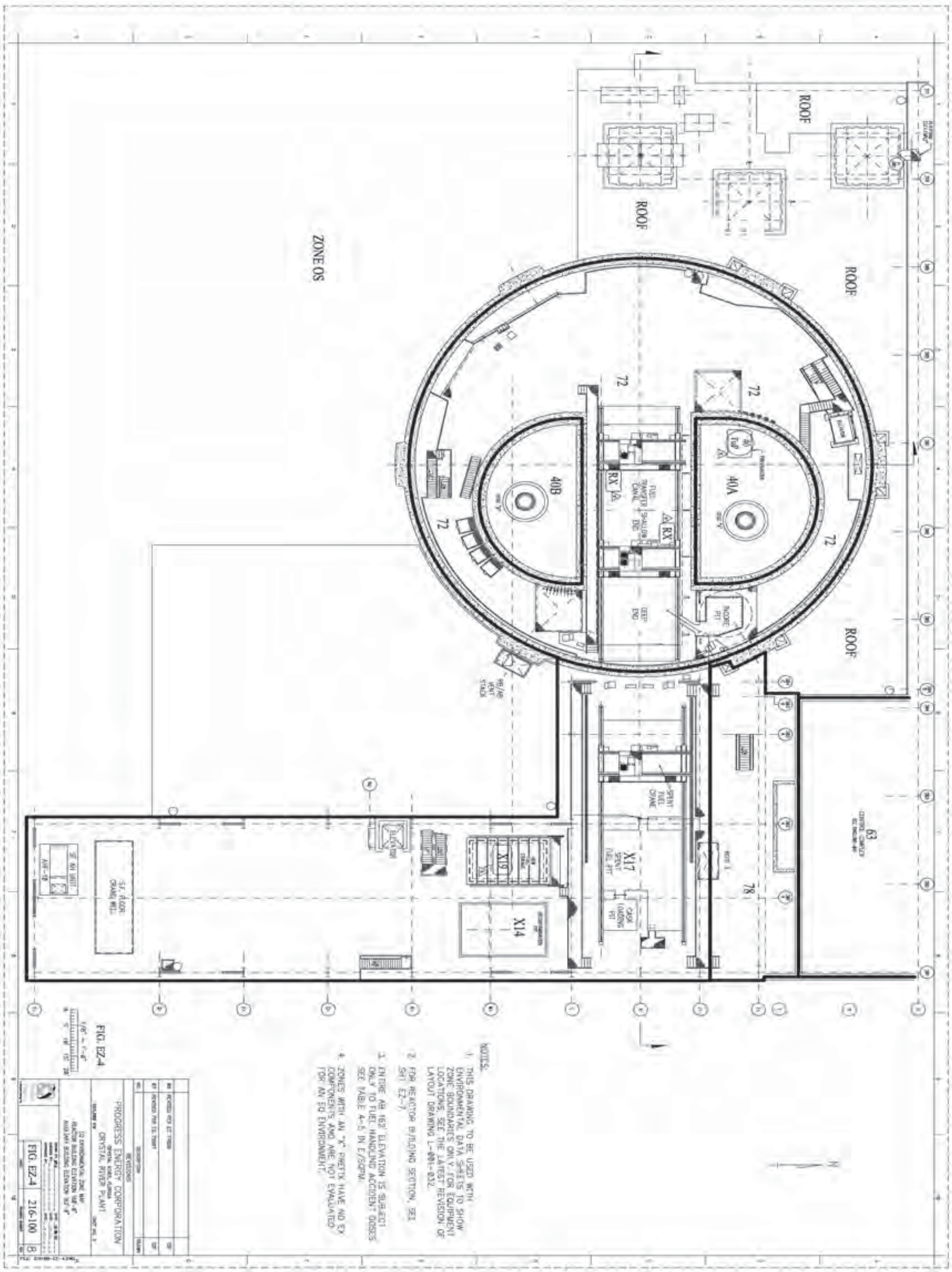


Figure 2.3.1-6

2.3.2 Offsite Power System

2.3.2.1 Regulatory Evaluation

The Offsite Power System includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The CR-3 review covered the descriptive information, analyses, and referenced documents for the offsite power system, and the stability studies for the electrical transmission grid. The CR-3 review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU.

The NRC's acceptance criteria for the Offsite Power Systems are based on:

- GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50 Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Sections 1.4.24, Emergency Power for Protection Systems, and 1.4.39, Emergency Power for Engineered Safety Features, insofar as it requires the system to have the capability and capacity to perform its intended functions during anticipated operational occurrences and accident conditions. [GDC-17]

2.3.2.2 Technical Evaluation

Introduction

The CR-3 transmission system is discussed in FSAR Section 8.2.1. The 230 kV switchyard provides the preferred source of shutdown power for CR-3 and for operation of emergency systems and engineered safeguards (ES) features during all modes of operation. The breaker-and-a-half switching arrangement in the 230 kV substation includes two full capacity main buses, where each bus is individually capable of supplying all the load required for the 230 kV system. The 230 kV switchyard is shared with other units at the Crystal River Energy Complex (Units 1, 2, and 4). There are five offsite transmission lines supplying the 230 kV switchyard, traveling through three independent rights-of-way, terminating in three separate substations.

The 500 kV lines connect the output of the CR-3 generator step-up transformers (GSU) to the switchyard. The 500 kV switchyard is shared with Crystal River Unit 5, a 720 MWe generator. Additionally, two full capacity offsite lines terminate at the 500 kV switchyard. The GSU transformers and the 500 kV lines are designed to deliver power from the output of the main generator to the external transmission lines located in the 500 kV switchyard under normal plant operating conditions.

The GSU transformers were replaced November 2007. These replacement GSU transformers and bushings are rated for 1200 MVA to continuously carry the maximum expected main generator output current at the minimum expected main generator operating voltage of 22 kV. There is no circuit breaker between the main generator and the generator step-up transformer. The 500 kV line from the step-up transformer joins the 500 kV ring bus between the two generator breakers. The main generator output breakers are rated for 3000 amps with a short circuit current rating of 37,000 amps. The current through the breaker pre-EPU is $\leq$ 1875 amps, and the expected current post-EPU is expected to be $\leq$ 2100 amps, therefore there continues to be significant margin to breakers rating during EPU conditions.

The current CR-3 gross generation capacity is approximately 924 MWe, with an expected EPU power increase to approximately 1080 MWe. The main generator is nameplate rated for 1200 MVA at 0.93 power factor lagging, providing sufficient capability for supporting operation at EPU conditions and sufficient reactive capability. Several modifications were installed in the plant during the R16 outage to support this rating. Additional modifications will be installed in the R17 outage to support the uprate, including the low and high pressure turbine replacement. Reference Appendix E – Major Modifications for further discussion of modifications.

Description of Analyses and Evaluations

Operation at EPU conditions does not impact the physical and electrical separation of the offsite circuits since there are no modifications to the offsite circuits required to support EPU.

The Grid Stability Study (refer to Appendix F, Grid Stability) was performed to evaluate the impact of the EPU on the reliability of the 230 kV and 500 kV systems in accordance with North American Electric Reliability Corporation (NERC) and regional electric power system guidelines, procedures and practices. The cases were built using the Florida Reliability Coordinating Council (FRCC) 2007 Databank cases, Revision 4.1, with modifications. The Base case represented the CR-3 generation prior to any of the uprate steps, and the transfer cases represented post-EPU. For the grid stability analysis, a 2012 summer dynamic case developed by the FRCC Stability Working Group (SWG) was used as a starting point. A steady state analysis, extracted from the stability summary report, examined post-transient power flow for the bulk Florida transmission system to determine if the loss of line and units subsequent to breaker failure will cause voltage and/or overload of system components.

Contingencies were evaluated with load flow analysis for each seasonal condition. This analysis involved an extensive examination of contingencies of local and cross-state transmission facilities located around the CR-3 plant area. Several analyses were performed in the study, including load flow (i.e., analyzing adequacy of thermal and voltage conditions), stability, and short-circuit cases. Extreme contingencies (i.e., loss of entire generation units) were also evaluated and found to be acceptable. CR-3 concludes that there is essentially no voltage variation between the base case and the post-EPU case. CR-3 finds that the results of the grid stability studies indicate that the grid remains stable for the EPU conditions. Therefore, the power uprate will not adversely impact the availability of the offsite power source for CR-3. Additional details of this analysis are provided in Appendix F. These studies were reviewed by the FRCC and the Florida/Southern Interface Planning Committee.

CR-3 has a formal Interface Agreement to provide grid support up to +/- 300 MVAR (leading/lagging) at the grid connections. This agreement has responsibilities defined for all Progress Energy parties involved in the generation and transmission of electrical power. Additionally, activities involved in the operation of the grid are defined including emergency operations and defense in depth requirements. Specific requirements for communication and frequency of communication between the control room and the transmission organization are identified. The intent of the Interface Agreement is to ensure that the voltage necessary to assure safe operation of the nuclear plant remains available at the switchyard for all expected or identified conditions. With the increased output of CR-3, the unit can provide up to +/- 523 MVAR (leading/lagging) at the grid connections, therefore providing capability to support the interface agreement.

Results

There is no significant change to the loading on the 230 kV switchyard post-EPU. Studies have been performed that demonstrate the 500 kV switchyard and transmission network have adequate capacity to accept the additional power from the uprate.

The studies also demonstrated acceptable steady state voltage at the 230 kV CR-3 bus. This means that for all scenarios reviewed, voltage in the switchyard remains above the minimum required voltage specified in the CR-3 design basis calculations, and there is significant margin to undervoltage conditions on the safety-related switch gear. However, the study indicates that the loss of the central Florida-Crystal River 500 kV line overloads the Brookridge 500/230 kV transformer to 101.1% and 101.8% in Winter 2011 and 2012, respectively (post-EPU). Should this overload condition occur in real time, Progress Energy Florida dispatchers have an established operation mitigation to reduce generation at Crystal River Unit 5 and increase generation at another Progress Energy site. There were no other impacts attributable to the CR-3 uprate on either Progress Energy Florida's system or on neighboring utilities' systems. The simulations and studies are based on the full 180 MWe uprate, which is considered the worst case situation for transient stability.

2.3.2.3 Conclusion

The CR-3 evaluation concludes that the Offsite Power System will continue to meet the CR-3 current licensing basis with respect to the requirements of FSAR Sections 1.4.24 and 1.4.39 following implementation of the proposed EPU. There is adequate physical and electrical separation and the Offsite Power System has the capacity and capability to supply power to all safety loads and other required equipment. CR-3 further concludes that the impact of the proposed EPU on grid stability is insignificant. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Offsite Power System.

2.3.2.4 References

None

2.3.3 AC Onsite Power System

2.3.3.1 Regulatory Evaluation

The Alternating Current (AC) Onsite Power System includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The CR-3 review covered the descriptive information, analyses, and referenced documents for the AC Onsite Power System.

The NRC's acceptance criteria for the AC Onsite Power System are based on:

- GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Sections 1.4.24, Emergency Power for Protection Systems, and 1.4.39, Emergency Power for Engineered Safety Features, insofar as it requires the system to have the capability and capacity to perform its intended functions during anticipated operational occurrences and accident conditions. [GDC-17]

2.3.3.2 Technical Evaluation

Introduction

As discussed in the FSAR Section 8.0, the electrical systems provide required power sources and equipment to ensure continued operation of essential reactor and station auxiliary equipment under all conditions. The onsite Class 1E AC Distribution System is divided into redundant trains so that the loss of any one train does not prevent the minimum safety functions from being performed. Engineered Safeguards (ES) Buses 3A and 3B of the 4160 Volt Auxiliary System are normally fed from the offsite power transformer through a direct cable connection or the backup ES transformer through the Backup ES Transformer Auxiliary Bus 3 and a cable/bus duct connection. The ES Buses comprise the two redundant Class 1E Electrical Systems. Each ES 4160V bus has connections to three offsite power sources and a single Emergency Diesel Generator (EDG). Each emergency generator is rated for 4083 kVA at 0.857 power factor (3500 kW).

CR-3 is provided with one full capacity unit auxiliary transformer and one full capacity start-up transformer. The unit auxiliary transformer is connected to the generator and can supply power to the unit auxiliary buses. The start-up transformer is connected to the 230 kV substation and can serve as the normal source of non-1E power during all modes. Each of the aforementioned transformers have two isolated secondary windings, one at 6900 volts and one at 4160 volts. Each transformer is capable of supplying the normal full non-1E load requirements of CR-3.

The function of the main generator is to convert the rotational (mechanical) energy supplied by the main turbine into electrical energy. The generator design requirements for EPU is hydrogen intercooled, 3 phase; 60Hz, 1200 MVA; 0.93 power factor, 22,000 volts and will be maintained at 75 psig of hydrogen. CR-3 presently generates approximately 914 MWe at one hundred percent core thermal power, 2609 MWt. Expected generation will be approximately 1080 MWe gross. Section 2.5.1.2.2, Turbine Generator, discusses the Main Generator. The 22 kV isolated phase bus duct houses the buses from the main generator to its various loads. These include the generator bus, main power (step-up) transformer bus, the unit auxiliary transformer bus and the potential transformer bus. The main transformers step-up the voltage to 500 kV to be delivered to the 500 kV substation. The transformers are single phase, two winding, 60 Hz, class forced oil and air (FOA). They are sized based on the capacity output of the main generator at 400 MVA at 65°C average rise per phase; for a three phase total of 1200 MVA. Thus, the main transformer is adequate for the generation increase associated with the EPU.

The impact of the EPU on the loading of ES Buses, and therefore on the loading of the EDGs, was reviewed in addition to the other components of the Class 1E and Non-1E Electrical Distribution system.

<u>Description of Analyses and Evaluations</u>

The CR-3 Electrical Systems and components were evaluated to ensure they are capable of performing their intended functions at EPU conditions. The evaluation is based on the system’s required design functions and attributes, and upon a comparison between the existing equipment ratings and the anticipated operating requirements at a rated thermal power (RTP) of 3014 MWt. The impact of the electrical load increase was evaluated using load flow, short circuit and protection-coordination studies. The results show that there is sufficient capacity and capability to accommodate the plant changes required to support EPU.

The load impacts on ES Buses (and therefore EDGs) from the EPU-related modifications (reference Appendix E for modification details) are summarized below:

- Modifying the low pressure injection (LPI) System to provide an additional crosstie inside containment which will assure the delivery of low pressure flow in the event of any kind of LOCA, especially a Core Flood Line Break accident. Also, in order to assure sufficient flow is available for core cooling, the maximum pump flow will be increased slightly which is expected to add approximately 6 kW to each ES Bus. For a Large Break LOCA coincident with a loss of offsite power, the minimum amount of EDG margin available is approximately 27 kW.

- Installing a hot leg injection (HLI) System to be used for boron precipitation control. Two motor operated valves will be removed from their current location in the LPI System and two smaller motor operated valves will be installed inside containment. Therefore, there is no increase in loading on the ES buses.

- The Atmospheric Dump Valves are air-operated valves and there is no expected load impact to the ES Bus.

- The position of the throttle valves on the HPI pumps will change to allow more flow to the reactor core during a SBLOCA. This will allow adequate HPI flow to the reactor during LOCA scenarios where HPI is required. The increase in flow will have no impact on EDG margin since the current analyses for HPI flow is bounding (run-out conditions).
- The new design for the Inadequate Core Cooling Mitigation System (ICCMS) is described in Section 2.4.2.3. The increased loads on the vital bus inverters and ES bus are bounded by the existing analyses.

These modifications have been reviewed for, and detailed implementation designs are being developed to address, the safety-related loads that could potentially impact diesel generator (DG) loading. The proposed EPU modifications will not result in any configuration changes that would adversely impact the maximum DG loading currently assumed in the licensing basis. Therefore, the DGs remain capable of performing the intended design function at EPU conditions.

The Non-1E Electrical System upgrades include

- The main generator was modified to be capable of producing 1080 MWe.
- The isolated-phase bus duct cooling system was modified to be capable of carrying the additional current. The pre-EPU and post-EPU isolated-phase bus duct design ratings are summarized in Table 2.3.3-1.
- Transformers were replaced due to material concerns and were sized to support EPU loads as identified in Table 2.3.3-2
- The balance of plant pumps modifications or replacements support the capability of supplying the additional flow required at EPU conditions. The impacts of these modifications are identified in Table 2.3.3-2. These modifications include the Condensate pump replacement (2000 HP to 2500 HP), the Secondary Cooling Closed Cycle pump replacement (350 HP to 600 HP), the Feedwater Booster Pump replacement (2500 HP to 3500 HP), and the Isolated-Phase Bus Duct cooler fans (40 HP to 100 HP), for an approximate load increase of 2750 kW

The majority of increased loading to the station distribution systems occur on non-1E distribution systems. For the Non-1E Distribution System, an ETAP model of the plant was used to determine the impact of the EPU on major electrical distribution system components. ETAP is a widely used industry electrical distribution system transient analyzer program. Table 2.3.3-2 provides a summary of the non-1E loading changes and EPU ratings for major non-1E electrical distribution components. In general, the pre-EPU loadings reflect the non-1E loadings prior to EPU modifications outlined in Appendix E (which include some modifications currently installed).

<u>Results</u>

The worst case transformer loading and short circuit studies performed for this evaluation indicate that the 1E and non-1E equipment voltages and fault duties are not adversely affected by EPU conditions or plant modifications required to support EPU, when powered from the offsite or onsite power sources. Also the loading requirements of switchgear buses and circuit breakers, bus ducts, 480V transformers and motor control centers are bounded by equipment ratings. Therefore, the Onsite AC Power System will continue to meet the CR-3 current licensing basis and perform their intended functions during all plant operating and accident conditions for EPU operation.

All lower voltage buses, switchgear and motor control centers were demonstrated to have sufficient voltages at the lowest operating voltage on the grid to assure operability of the connected equipment. Additionally, the EDGs are unaffected by changes required for uprated conditions and retain the capability to supply the ES buses as needed.

As such, the CR-3 Electrical Distribution System has the capacity and capability of performing its function during all operational and emergency conditions.

2.3.3.3 Conclusion

The CR-3 evaluation concludes that the AC Onsite Power System adequately accounted for the effects of the proposed EPU on the system's functional design. CR-3 further concludes that the AC Onsite Power System will continue to meet the CR-3 current licensing basis with respect to the requirements of FSAR Sections 1.4.24 and 1.4.39 following implementation of the proposed EPU: Therefore, CR-3 finds the proposed EPU acceptable with respect to the AC Onsite Power System.

2.3.3.4 References

None

Table 2.3.3-1 Isolated Phase Bus Duct Ratings

Bus Section	Pre-EPU Design Rating (amps continuous)	Post EPU Design Rating (amps continuous)
Generator Bus	27,500	33,100
Main Power Step-up Transformer Bus	16,000	17,700
Unit Auxiliary Transformer Bus	2,000	2000
Potential Transformer Bus	1,200	1,200

Table 2.3.3-2 non-Class 1E Component Summary

Component	Pre-EPU Loading	Post-EPU Loading	Post-EPU Component Rating
Generator	921 MVA / 25,355 A	1157 MVA / 30,670 A	1200 MVA
Main Power Transformer	874 MVA / 24,122 A	1100 MVA / 29,286 A	1200 MVA
Iso-Phase Bus Duct	920 MVA / 25,354 A	1155 MVA / 30,670 A	33,100 A
Unit Auxiliary Transformer	49.044 MVA / 1315 A	53.001 MVA / 1407 A	61.6 MVA FOA @ 65°C
Start-Up Transformer	50.465 MVA / 122.4 A	54.622 MVA / 132.5 A	61.6 MVA FOA @ 65°C
4160 non-segregated bus duct	21.56 MVA / 3161 A	12.834 MVA / 1908 A 12.17 MVA / 1809 A	2000 A 2000 A
4160 V Reactor Auxiliary Bus	1.062 MVA / 153.4 A	1.062 MVA / 151.6 A	1200 A
4160 V Unit Bus 3A/3B	11.042 MVA / 1619 A	12.834 MVA / 1908 A	2000 A
6900 V Reactor Auxiliary Bus 3A/3B	13.255 MVA / 1148 A	13.258 MVA / 1132 A	2000 A
480 V Reactor Auxiliary Bus 3A/3B	0.836 MVA / 1080 A	0.924 MVA / 1162 A	3000 A
480 V Turbine Auxiliary Bus 3A/3B	1.576 MVA / 2079 A	1.559 MVA / 2087 A	3000 A
480 V Plant Auxiliary Bus (Heating Aux Bus)	0.518 MVA / 665.2 A	0.508 MVA / 661.8 A	3000 A
480 V Intake Bus A/B	0.138 MVA / 177.5 A	0.141 MVA / 176 A	1600 A

2.3.4 DC Onsite Power System

2.3.4.1 Regulatory Evaluation

The Direct Current (DC) Onsite Power System, hereafter referred to as the Class 1E DC Power Systems, includes the DC power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The CR-3 review covered the information, analyses, and referenced documents for the DC Onsite Power System.

The NRC's acceptance criteria for the DC Onsite Power System are based on:

- GDC-17, insofar as it requires the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences (AOOs) and accident conditions

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria:

- FSAR Section 1.4.24 and FSAR Section 1.4.39, insofar as they require the system to have the capacity and capability to perform its intended functions during AOOs and accident conditions. [GDC-17]

2.3.4.2 Technical Evaluation

Introduction

The CR-3 250/125 volt DC (DP) System is discussed in FSAR Section 8.2.2.6. The Class 1E portion includes two separate and electrically independent 125/250 volt DC power distribution systems. Each 125/250 volt DC system includes a battery, two battery chargers plus a spare, and distribution panels. The loads on the Class 1E 125/250 volt DC systems include the alternating current (AC) Vital Bus inverters as well as various motor operated valves and control devices. In addition, there is a 125 volt Class 1E DC System dedicated to the support of the diesel driven emergency feedwater pump, EFP-3, including a battery, battery charger, and distribution panel.

Stand alone DC power supplies will be independent and separate from the CR-3 1E batteries. Normal power will be supplied from a non-safety-related, non 1E battery charger. These will be used to support the Atmospheric Dump Valves and Fast Cooldown System. (reference Appendix E – Major Modifications for details).

Description of Analyses and Evaluations

Operation at EPU conditions does not impact the physical and electrical separation of the Class 1E DP System, since there are no significant modifications to the distribution circuits required to support EPU.

The load impacts on the Class 1E DP System from the EPU-related modifications (reference Appendix E for modification details) are summarized below:

- Modifying the low pressure injection (LPI) system to provide an additional crosstie inside containment which will assure the delivery of low pressure flow in the event of any kind of LOCA, especially a Core Flood Line Break accident. Two AC power, motor operated valves will be removed from the system and no additional motor operated valves will be installed. Also, in order to assure sufficient flow is available for core cooling, the maximum pump flow will be increased slightly. The net result of these changes is expected to be a slight increase on the ES AC Buses. These changes will not affect the DC Power System.

- Installing a hot leg injection (HLI) system to be used for boron precipitation control. The motor operated valves will be AC powered and the control power for the valves will also be AC. These changes will not impact the DC power system.

- The Atmospheric Dump Valves will have their own separate DC power system, and will therefore not impact the existing Class 1E DP Systems.

- Emergency feedwater flow is increased by the addition of a recirculation flow control system. The DC power impacts from this modification remain within the loading capabilities of the station batteries.

- Instrumentation changes resulting from the above modifications are expected to result in a very minor increase on the Vital Bus inverters and remain within the loading capabilities of the Vital Bus inverters and station batteries.

Results

The EPU project modifications will not result in a significant load change on the Class 1E DP Systems. Based on the results from a plant evaluation all of the identified modifications that add loads to the Vital AC System are well within the working margin established for the individual inverters. The Class 1E DC loads remain within the loads assumed in the battery sizing calculations.

2.3.4.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the Class 1E DP Systems and concludes that effects of the proposed EPU on the system's functional design have been adequately addressed. It is further concluded that the Class 1E DC Power System will continue to meet the requirements of FSAR Sections 1.4.24 and 1.4.39 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all safety loads and other required equipment. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Class 1E DC Power Systems.

2.3.4.4 References

None

2.3.5 Station Blackout

2.3.5.1 Regulatory Evaluation

Station Blackout (SBO) refers to a complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves the loss of offsite power (LOOP) concurrent with a turbine trip and failure of the onsite emergency AC Power System. SBO does not include the loss of available AC power to buses fed by station batteries through inverters or the loss of power from “alternate AC sources” (AACs). The CR-3 review focused on the impact of the proposed EPU on the plant’s ability to cope with and recover from an SBO event for the period of time established in the CR-3 licensing basis.

The NRC’s acceptance criteria for Station Blackout are based on:

- 10 CFR 50.63, “Loss of All Alternating Current Power”

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

Additionally, FSAR Section 14.1.2.9.4.2 provides criteria for the required station blackout coping duration. [10 CFR 50.63]

2.3.5.2 Technical Evaluation

Introduction

The SBO accident, addressed in FSAR Section 14.1.2.9.5.2 as a complete loss of all unit AC power, is a hypothetical case where all unit power is lost except the unit Class 1E Direct Current (DC) System batteries. Following the loss of all unit AC power, the control rods are inserted into the core via gravity and the turbine valves close. The loss of station power also results in a reactor coolant pump (RCP) trip, a main feedwater pump trip, and a condensate booster pump trip. The resulting heat removal mismatch is the driving force behind the dynamics of the SBO transient.

Description of Analyses and Evaluations

The coping duration for CR-3 requires sustaining an SBO event for 4 hours. EPU conditions were considered and determined not to impact SBO actions or coping duration as follows:

- EPU will not impact grid stability (Section 2.3.2, “Offsite Power System”)
- EPU will not increase loss of offsite power due to extremely severe weather
- EPU will not alter the independence of Offsite Power Systems

- EPU will not alter the number of or reliability of Emergency Diesel Generators

Condensate Inventory for Decay Heat Removal

At the EPU conditions, 90,737 gallons of water are needed for natural circulation cooldown to provide HOT STANDBY cooling for approximately 4 hours (SBO coping time) assuming conservative decay heat conditions. The decay heat is based on inputs that include initial core power of 3026.1 MWt and 16.4 MWt from RCP heat. The decay heat is based on 1.0 times the American Nuclear Society (ANS) 1971 decay heat standard for fission plus heavy actinides. CR-3 Improved Technical Specifications (ITS) requires at least 150,000 gallons to be in the Emergency Feedwater Tank while at power so sufficient inventory is available without having to perform manual operator actions. Additional inventory may be directly supplied as follows:

- Condensate Storage Tank with 120,000 gallons
- Fire Service Water Storage Tanks with 600,000 gallons

The SBO event analyses for CR-3 demonstrated that the core remains adequately covered. Assumed reactor coolant system (RCS) leakage is the 11 gpm ITS limited leakage (1 gpm unidentified and 10 gpm identified leakage) and 25 gpm seal leakage from each RCP, for a total of 111 gpm. Letdown flow of 80 gpm is assumed to be manually isolated after 10 minutes. These parameters are not affected by EPU. The evaluation for EPU conditions demonstrated the RCS coolant level remains above the top of the fuel therefore the SBO coping duration remains valid.

Steam, removing the decay heat, will be released through the Atmospheric Dump Valves (ADVs). These valves will be increased in capacity (refer to Section 2.5.5.3, “Steam Dump System,” and Appendix E, “Major Plant Modifications”).

Class 1E Battery Capacity

DC electrical power subsystems, Trains A and B, provide the motive and control power for their associated Class 1E pumps and valves. The batteries are rated at 1708 amp-hours. The capacity is based on the discharge of 116 cells (2 banks of 58 lead acid cells) from the fully charged condition down to 1.81 V per cell (105.0 V per battery) at maximum discharge. The larger safety-related ADVs will have their own source of DC power, separate from the safety-related 120 V vital power buses that will provide sufficient power for 4 hours at EPU conditions; the time required to cope with a SBO event.

Modifications outlined in Appendix E, are expected to result in a minor increase on the Vital Bus inverters and remain within the loading capabilities of the Vital Bus inverters and station batteries as addressed in Section 2.3.4, DC Onsite Power System. The impacts of the EPU modifications do not preclude the safety functions from being performed as they are needed.

Compressed Air

The components requiring the Instrument Air System during the SBO event are the ADVs. The ADVs are being replaced to enhance their functionality and designed to be classified safety-related. The ADVs will be larger valves procured as safety-related and will have safety-related back-up air. The design will allow elimination of the manual operator action to align backup air to the ADVs (reference Section 2.11, "Human Performance"). This action will be accomplished with automatic check valves. The ADVs will be capable of operating immediately, as opposed to the delay required for manual operator action. The safety-related air supply is sized to provide sufficient pressurized gas to operate the ADVs for 4 hours at EPU conditions; the time required to cope with a SBO event. Refer to Section 2.5.5.3 and Appendix E for additional discussion of the ADV modifications.

Effects of Loss of Ventilation

The maximum temperatures at CR-3 during a SBO event remain under the maximum allowable temperatures for the applicable areas at EPU conditions. The temperatures have been evaluated for the added EPU heat loads and found acceptable. The dominant areas of concern include the turbine driven emergency feedwater pump area and the ADV area. As such, no additional manual operator actions are required to protect plant equipment necessary to cope with the SBO event for the turbine driven emergency feedwater pump area. For the ADV area, the new ADV equipment is qualified for the temperatures anticipated to occur when ventilation is lost during the SBO event.

Containment Isolation

The SBO evaluation of the containment isolation capability for a SBO condition determined that all containment isolation valves would be closed by the loss of power, were already in a closed position and would remain closed, or were included into the appropriate Emergency Operating Procedure to be closed. As discussed in Appendix E – Enclosure 1, one new inboard containment isolation valve will be installed for EPU, but it will be a check valve and as such, the containment isolation coping analysis is not impacted by EPU.

Effects of Loss of Subcooling Margin

If the loss of AC power is prolonged, RCS subcooling will be lost and voids will begin to appear and increase in size in the upper RCS elevations. Upon recognition of a loss of subcooling margin, operator actions are required to (1) select the OTSG level control setpoint to the ISCM level and establish maximum EFW flow to the OTSGs, and (2) commence a secondary plant depressurization using both ADV's. For EPU, the Fast Cooldown System (FCS) is required to be bypassed in order to quickly depressurize the OTSGs below the 350 psig FCS control setpoint. Bypassing the FCS system during an SBO with a loss of adequate SCM will allow the ADVs to be controlled as reflected in current and EPU procedures. Refer to Section 2.11 for discussion of the procedure impacts and changes to operator actions.

Results

The systems and components necessary to cope with a SBO event will continue to perform their required safety function post-EPU. The decay heat removal function requires additional inventory, but the inventory is available in normal tank volumes. Battery capacity is sufficient as the minor load changes remain within the battery loading capabilities. Safety-related backup air will be adequate to supply the new, larger ADVs for 4 hours. Ventilation loss in the turbine driven emergency feedwater pump areas will continue to require manual operator actions to preclude specific equipment from overheating. The Containment Isolation capability is unaffected by the EPU. The revised operator manual action to bypass FCS to depressurize the OTSGs upon recognition of loss of subcooling is not considered to result in a significant impact.

2.3.5.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the ability to cope with and recover from a SBO event for the period of time established in the CR-3 licensing basis. CR-3 concludes that the effects of the proposed EPU on SBO have been adequately evaluated and it has been demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to SBO.

2.3.5.4 References

None

2.4 Instrumentation and Controls

2.4.0.1 Regulatory Evaluation

The functions of the Instrumentation and Control Systems are provided: (1) to control plant processes having a significant impact on plant safety, (2) to initiate the Reactor Protection System (RPS), including control rods, (3) to initiate the Engineered Safety Features (ESF) Systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The CR-3 review of the RPS, Engineered Safeguards Actuation System (ESAS), Remote Shutdown System (RSS), Inadequate Core Cooling Mitigation System (ICCMS), Integrated Control System (ICS), and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The CR-3 review was also conducted to ensure that failures of the systems do not affect safety functions.

The NRC's acceptance criteria for Instrumentation and Controls are based on:

- GDC-1;
- GDC-4;
- GDC-13;
- GDC-19;
- GDC-20;
- GDC-21,
- GDC-22,
- GDC-23;
- GDC-24,
- 10 CFR 50.55a(a)(1), and
- 10 CFR 50.55a(h),

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Sections 1.4.1, Quality Standards, and 1.4.5, Records Requirements [GDC-1];
- FSAR Sections 1.4.23, Protection Against Multiple Disability for Protection Systems, and 1.4.40, Missile Protection [GDC-4];
- FSAR Sections 1.4.12, Instrumentation and Control Systems, 1.4.13, Fission Process Monitors and Controls, 1.4.16, Monitoring Reactor Coolant Pressure, and 1.4.17, Monitoring Radioactivity Release [GDC-13];
- FSAR Section 1.4.11, Control Room [GDC-19];
- FSAR Sections 1.4.14, Core Protection Systems, and 1.4.15, Engineered Safety Features Protection Systems [GDC-20];
- FSAR Sections 1.4.19, Protection Systems Reliability, 1.4.20, Protection Systems Redundancy and Independence, 1.4.21, Single Failure Definition, and 1.4.25, Demonstration of Functional Operability of Protection Systems [GDC-21];
- FSAR Sections 1.4.20, Protection Systems Redundancy and Independence, and 1.4.23, Protection Against Multiple Disability For Protection Systems [GDC-22];
- FSAR Section 1.4.26, Protection Systems Fail-Safe Design [GDC-23];
- FSAR Section 1.4.22, Separation of Protection and Control Instrumentation Systems [GDC-24];

Additionally, FSAR Section, 1.6 Quality Program (Preoperational), and FSAR Section 1.7, Quality Program (Operational), provide the criteria to meet 10CFR50.55a(a)(1). FSAR Section 7.1.1, Design Bases (Protection Systems), provides the criteria to meet [10CFR50.55a(h)].

Anticipated Transient Without Scram (ATWS) Mitigating System Actuating Circuitry (AMSAC), and Diverse Scram System (DSS) are addressed in Section 2.8.5.7, Anticipated Transients Without Scrams. The design changes for the ADVs and the new Fast Cooldown System (FCS) are addressed in Appendix E, “Major Plant Modifications”.

2.4.0.2 Technical Evaluation

This Section is an introduction for the subsequent sections (2.4.1 through 2.4.4). The Technical Evaluation portion will be presented in each of the individual sections as appropriate.

2.4.0.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the functional design of the RPS, ESAS, RSS, ICCMS, ICS, Control Rod Drive Control System, and ATWS instrumentation. CR-3 concludes that the evaluation has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant’s design basis. Based on the above discussion, CR-3 concludes that the systems will continue to meet the requirements of 10CFR50.55a(a)(1), 10 CFR 50.55a(h), and FSAR Sections 1.4.1, 1.4.5, 1.4.11, 1.4.12, 1.4.13, 1.4.14, 1.4.15, 1.4.16, 1.4.17, 1.4.19, 1.4.20, 1.4.21, 1.4.22, 1.4.23, 1.4.25, 1.4.26, and 1.4.40. Therefore, CR-3 finds the proposed EPU acceptable with respect to instrumentation and control.

2.4.1 Reactor Protection System (RPS)

2.4.1.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.1.2 Technical Evaluation

Introduction

The RPS monitors parameters related to safe operation and trips the reactor to protect the reactor core. It also protects against Reactor Coolant System (RCS) damage caused by high system pressure by limiting energy input to the system through reactor trip action.

The proper functioning of the RPS prevents violation of the reactor core Safety Limits (SLs) required in Improved Technical Specifications (ITS) Section 2.1.1.

Description of Analyses and Evaluations

The RPS and instrumentation were evaluated to ensure that they are capable of performing their safety related design functions with the EPU. The evaluation included a review of RPS existing instrument ranges, reactor trip settings, and plant computer process parameters considering the operating conditions that will change with the EPU. A review of safety analysis assumptions for automatic actuations confirms the existing ITS Allowable Values continue to provide the necessary core protection.

No changes to the RPS design or allowable values in the ITS are required for the EPU. The existing instrument ranges for RPS are not affected by the EPU.

- Reactor Trip on Overpower

  The High Power trip function initiates a reactor trip when the neutron power reaches a pre-defined setpoint corresponding to the design overpower limit (ITS Bases Section 3.3.1). The trip settings for the high trip function are currently limited to a ITS Allowable Value (AV) of 104.9% or 103.3% pre-EPU Rated Thermal Power (RTP) to protect against excessive power levels and reduce reactor power to prevent violation of the RCS pressure SL. The AV of 104.9% RTP is based on a secondary heat balance with required high accuracy instrumentation while the 103.3% RTP is based on Feedwater Flow nozzle input into the secondary heat balance. Transient and accident analyses were performed for the EPU modeling using 112% (RTP) as the Safety Analysis Limit, and a high flux trip setpoint of 104.9% RTP. Results of each accident and transient were determined to be within applicable limits. No change to the High Power trip AV is required and it is acceptable for the High Power trip setting to be referenced from the EPU conditions. The High Power trip function will continue to provide protection against excessive power levels by reactor power reduction and thereby minimize the likelihood for violations of the RCS pressure Safety Limit during the EPU conditions. The Nuclear Overpower – High Setpoint AV 104.9% RTP is no longer based on the assumption that the required high accuracy secondary heat balance instrumentation (LEFM) is functional because sufficient margin exists between the RPS Setpoint and Analytical Limits. However, the high accuracy secondary heat balance instrumentation continues to be necessary to operate at RTP as limited by ITS 3.3.1, Required Action J.1.

- Reactor Trip on Overpower/Reactor Coolant Flow/Axial Imbalance

  This function is affected by the cycle-specific core changes. The only calibration change for the Nuclear Overpower based on RCS Flow and Axial Power Imbalance will be the cycle specific trip setpoint as noted in the COLR. Therefore, the trip setpoint envelope is verified or re-calculated every fuel cycle. The existing instrument will support any required RPS setpoint changes. No changes to the ITS are required, since this AV is referenced to the COLR.

- Reactor Trip on High RC Outlet Temperature

  The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the departure from nucleate boiling ration (DNBR) SL. The AV is ≤618°F for the RCS High Outlet Temperature trip and is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. A reactor coolant temperature analysis limit of 620°F was used in the evaluation of the Rod Ejection Accident (FSAR Section 14.2.2.4) and is bounding for the AV of 618°F. The transient analysis concluded that the EPU was acceptable with respect to the control rod ejection accident. (Refer to Section 2.8.5, Accident and Transient Analyses.) As such, the High RC Outlet Temperature AV of 618°F and related trip setpoints remain acceptable for the EPU.

- Reactor Trip on Variable Low Pressure

  The Variable Low RCS pressure trip setpoint envelope defines allowable RCS pressure and temperature combinations that will protect the core DNB pressure-temperature limits. The variable low RCS pressure trip setpoint envelope is verified or re-calculated for each fuel cycle as noted in the COLR. The existing instrument will support any COLR required RPS changes.

- Reactor Trip on Low RCS Pressure

  The RCS Low Pressure trip provides protection for the DNBR SL. The results of transient and accident analyses confirmed that the ITS AV of ≥ 1900 psig is acceptable. Refer to Section 2.8.5, Accident and Transient Analyses for results of specific transients analyzed. No changes are required for the RCS Low Pressure trip AV or trip setpoints.

- Reactor Trip on High RCS Pressure

  The trip settings for the RCS high pressure trip function are currently limited to an ITS AV of 2355 psig to protect the RC System. Transient and accident analyses performed for the EPU confirmed that the existing AV of ≤ 2355 psig remains acceptable. There are no changes required for the RPS Reactor Trip on High RCS Pressure AV or trip settings.

- Reactor Trip on Reactor Coolant Pump (RCP) Power

  The ITS AV for RPS Reactor Coolant Pump Power Monitor (RCPPM) is more than one pump drawing ≤ 1152 or > 14,400 kW, selected to prevent normal power operation unless at least three RCPs are operating. The overpower setpoint is selected low enough to detect locked rotor conditions (although credit is not taken for this capability) but high enough to avoid a spurious trip. This criteria remains unchanged with the EPU, as such, the AV remains acceptable.

- Reactor Trip on High Reactor Building (RB) Pressure

  The AV for the RB High Pressure trip is ≤ 4 psig chosen at the lowest value consistent with avoiding spurious trips during normal operation. The RB High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB, and also provides a backup to RPS trip strings exposed to an RB HELB environment. This trip is not credited in any of the analyzed FSAR Chapter 14 accidents. The effects of this trip are considered where an early trip may make the results worse, such as the post-trip reactivity response for a steam line break. Transient analysis assumes that the high pressure trip and delay occur at the start of the break. Therefore, the AV for the RB High Pressure trip remains acceptable for the EPU.

- Reactor Trip on Loss of Both Main Feedwater Pumps

  The Loss of Main Feedwater Pumps (Control Oil Pressure) trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a loss of feedwater (LOFW) event. The AV is ≥ 55 psig at a power level of ≥ 20% RTP. The trip setting for the feedwater pump control oil pressure bistable is established in reference to an AV of ≥ 55 psig and selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. The AV for loss of both main feedwater pumps (Control Oil Pressure) will not change with the EPU.

- Reactor Trip on Main Turbine Trip

  The Main Turbine Trip Function provides an early reactor trip at higher power levels in anticipation of the loss of heat sink associated with a turbine trip. For the Main Turbine Trip (Control Oil Pressure) bistable, the AV of 45 psig was selected to provide a trip whenever turbine control oil pressure drops below the normal operating range. With the EPU, there are no changes to the normal operating range for control oil pressure. Thus, the AV of 45 psig and the 45% RTP interlock are unchanged for the EPU.

- Reactor Trip on Overpower or High RC Pressure During Shutdown Bypass Mode of Operation

  Most RPS trips can be bypassed during shutdown, which is shutdown bypass operation. If in shutdown bypass operation, and the reactor trip breakers are closed, two reactor trips, RCS High Pressure trip and Nuclear Overpower-Low Setpoint are active. The RCS High Pressure trip setpoint of ≤ 1820 psig and the Nuclear Overpower-Low Setpoint of < 5% RTP are unchanged for the EPU. The design function of the RPS shutdown bypass instrumentation will not be modified for the EPU.

<u>Results</u>

No hardware or design changes are required for the above items.

2.4.1.3 Conclusion

See Section 2.4 for Conclusion.

2.4.1.4 References

None

2.4.2 Engineered Safety Features Systems

2.4.2.1 Engineered Safeguards Actuation System (ESAS)

2.4.2.1.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.2.1.2 Technical Evaluation

Introduction

The ESAS monitors variables to detect the loss of Reactor Coolant System (RCS) boundary integrity. Upon detection of "out of limit" conditions of these variables, the ESAS initiates operation of the High Pressure Injection (HPI), Low Pressure Injection (LPI), Reactor Building Isolation and Cooling (RBIC), and Reactor Building Spray (BS) Systems. Additionally, it starts the emergency diesel generators A and B.

The ESAS initiates the functions to fulfill the following:

- protect the fuel cladding
- ensure Reactor Building integrity
- limit the maximum value of energy released by an accident
- remove fission products from the Reactor Building atmosphere in the event of a Loss of Coolant Accident (LOCA)
- prevent overloading the emergency diesel generators in the event of a Loss of Offsite Power (LOOP) coincident with an accident

Description of Analyses and Evaluations

The ESAS was evaluated to ensure that it is capable of performing the required accident mitigation functions with the EPU. This evaluation included a review of ESAS existing instrument ranges, and trip settings considering the operating conditions that will change with the EPU. Following the EPU, the RCS average coolant temperature (T_{AVG}) will increase by approximately 3°F to 582°F; however, Tcold (Cold leg coolant temperature) will decrease by less than 1°F. The RCS pressure is unaffected by the EPU. The planned EPU at CR-3 necessitated the reanalysis of the LOCA using large and small break LOCA input models that represent the CR-3 unit with Mark-B-HTP fuel.

- Steam Line Failure Accident

  The Main Steam Line Break (MSLB) analyses demonstrated that core integrity is maintained for effective core cooling during a steam line break incident. No required changes were identified for the ESAS due to Main Steam Line Break (MSLB) event. (Refer to Section 2.8.5, Accident and Transient Analyses.)

- Steam Generator Tube Rupture (SGTR)

  Steam Generator Tube Rupture Accident analysis used a Low RCS pressure trip of 1714 psia. The ESAS RCS low pressure Improved Technical Specifications (ITS) Allowable Value (AV) is ≥ 1625 psig (≥ 1640 psia). The AV of 1640 psia was conservatively assumed to be 1687 psia in the steady-state analysis for the Steam Generator Tube Rupture event. This value was then corrected for the transient analysis to 1714 psia. No required changes were identified for the ESAS due to Steam Generator Tube Rupture event. (Refer to Section 2.8.5, Accident and Transient Analyses.)

- Loss of Coolant Accident (LOCA)

  LOCA calculations and analysis are summarized in the CR-3 LOCA summary report which demonstrates compliance with the acceptance criteria set forth by 10 CFR 50.46. There are no changes that affect the ESAS instrumentation required for detection and actuation of equipment for mitigation of LOCA. (Refer to Section 2.8.5, Accident and Transient Analyses.)

- Makeup System Letdown Line Failure Accident

  The ESAS system will terminate this accident condition at 478 seconds upon closure of the letdown isolation valves due to HPI diverse containment isolation after the RCS low pressure setpoint is reached. Transient analysis for the letdown line failure at the EPU conditions demonstrated compliance with the FSAR acceptance criteria for this event. Thus, no changes to the Emergency Safety Features (ESF) System are required due to this event. (Refer to Section 2.8.5, Accident and Transient Analyses.)

- Maximum Hypothetical Accident

  For LOCA analysis, an AV of 1625 psig was used for the low RCS pressure ESAS setpoint, and an AV of 500 psig was used for the low-low-RCS pressure ESAS setpoint. LOCA analysis results are in compliance with 10 CFR 50.46 Emergency Core Cooling System (ECCS) Criteria. (Refer to Section 2.8.5 Accident and Transient Analyses.) No changes to the ESF system are required due to this event.

Results

Based on the analysis, the control actions of the ESAS will support the EPU.

2.4.2.1.3 Conclusion

See Section 2.4 for Conclusion.

2.4.2.1.4 References

None

2.4.2.2 Emergency Feedwater Initiation and Control (EFIC)

2.4.2.2.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.2.2.2 Technical Evaluation

Introduction

The EFIC System is designed to provide emergency feedwater (EFW) initiation and control, main steam and feedwater isolation, EFW termination on a steam generator overfill condition, Feed Only Good Generator control logic, and atmospheric dump valve control. The EFIC Systems are discussed in FSAR Section 7.2.4. Improved Technical Specifications (ITS) Table 3.3-11.1 identifies the EFIC instrumentation.

The EFIC system must automatically or manually provide water to the steam generators at a rate sufficient to remove decay heat for loss of main feedwater (LOFW) events. The intent of the LOFW and feedwater line break (FWLB) analyses is to demonstrate compliance with the acceptance criteria listed in the FSAR Section 14.2.2.9.2.

Description of Analyses and Evaluations

The following details the analysis for the EFIC trip functions and EFW System flow parameters.

- Low Once Through Steam Generator (OTSG) Pressure

  The Low OTSG Pressure ITS Allowable Value (AV) is ≥ 600 psig to ensure EFIC actuation on low OTSG pressure for isolation of Main Feedwater and Main Steam(see ITS Table 3.3.11-1). Transient analysis considered an OTSG low pressure of 572.76 psig, which includes EFIC signal process delay and valve closure, a conservative value with regard to the ITS value of ≥ 600 psig. As such, the current AV and setpoint for EFW actuation on low OTSG pressure will remain the same for the EPU operations.

- Low OTSG Liquid Level

  EFIC provides EFW initiation on low OTSG level and provides subsequent OTSG level control. The AV for EFW initiation on low OTSG level is ≥ 0 inches. These EFIC instruments continue to provide EFW initiation and OTSG level control on low OTSG level without changing the setting. This trip is a primary indication of steam line or feedwater line breaks. This setting is assumed in the EPU Main Steam Line Break analysis (See Section 2.8.5, Accident and Transient Analyses)

- Loss of all Four RCPs

  The ITS require that there are four EFIC channels per reactor coolant pump (RCP) at ≥ 10% RTP to receive RCP status signals from the Reactor Protection System (RPS) (see ITS Table 3.3.11-1). The RPS senses the loss of power to all four RCPs and provides a loss of RCP indication for each pump to each EFIC channel. EFIC actuates EFW when a minimum of two EFIC channels recognize loss of all RCPs. There are no changes to ITS EFIC requirements for the RCP status function as a result of implementation of the EPU. This instrumentation continues to function, as designed, to actuate EFW upon recognition of loss of four RCPs.

- Loss of both Main Feedwater Pumps

  The ITS require four EFIC channels to receive RPS signals for loss of main feedwater pumps. Loss of both main feedwater pumps initiates EFW. There are no design changes to this EFIC function and this instrumentation continues to initiate EFW with loss of both main feedwater pumps (see ITS Table 3.3.11-1).

- EFW Flow

  The EFW System is being modified as discussed in Appendix E Major Modifications. The results from the transient events utilize the modified EFW System (Refer to Section 2.8.5, Accident and Transient Analyses). Based on the results of the transient analyses that utilize the EFIC settings, the control actions of the EC System will support operation at the uprate power level.

- Control of Atmospheric Dump Valves (ADVs)

  The ADVs are controlled through the EFIC Cabinets with input from pressure transmitters in the main steam lines. The ADVs are either controlled in auto with a setpoint of 1025 psig by control modules in the EFIC Cabinets or manually from hand/auto control stations in the main control room. The control circuitry for the pressure control developed demand signal to the ADVs is being modified to add a new safety related Fast Cooldown System (FCS) function for mitigating specific small break loss of coolant accident (SBLOCA) and high pressure injection (HPI) pump/flow failure scenarios (refer to Appendix E, Major Plant Modifications), which is separate from and independent of EFIC. The FCS will be used to allow operation of the ADVs to support SBLOCA while retaining the availability of the EFIC or manual generated ADV demand signal for all other accident/event/cooldown functions.

- EFIC System Initiation of EFW with AMSAC

  The Anticipated Transients Without Scram (ATWS) Mitigation System and Actuation Circuit (AMSAC) will cause EFIC initiation of EFW with reactor power greater than or equal to 50% and Main Feedwater Flow less than 17% nominal total flow. The logic for the AMSAC is unchanged for the EPU. The 17% full-scale Main Feedwater flow setpoint has been shown to be valid for the EPU through plant-specific analyses with acceptable results. (Refer to Section 2.8.5.7, Anticipated Transients Without Scram.)

Results

The EPU requires an increase in minimum required EFW flow and a decrease in maximum EFW actuation delay time. The analysis for the LOFW transient event concluded that there remains adequate available emergency feedwater flow with the EPU. The AV and setpoint for EFW actuation on Low OTSG pressure will remain the same for EPU. The change to EFIC for the fast cooldown function is acceptable for mitigation of the applicable transients and acceptable for the EPU operation. The review also ensured that failures of the EFIC system as modified does not affect safety functions. Based on the analysis, the control actions of the EFIC System will support the EPU.

2.4.2.2.3 Conclusion

See Section 2.4 for Conclusion.

2.4.2.2.4 References

None

2.4.2.3 Inadequate Core Cooling Mitigation System (ICCMS)

2.4.2.3.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.2.3.2 Technical Evaluation

Introduction

The ICCMS monitors specific variables to detect a loss of subcooling margin. Upon detection of “out of limit” conditions of these variables coincident with a reactor trip, the ICCMS will initiate a Reactor Coolant Pump (RCP) trip within one minute and will additionally raise the steam generator secondary side water level to the inadequate core cooling level. On a sustained loss of subcooling margin together with confirmed inadequate HPI flow, the ICCMS will actuate the Fast Cooldown System (FCS). (Refer to Appendix E for a discussion of the FCS and the ICCMS) The ICCMS is an analog system and is similar in design to the ESAS.

The ICCMS initiates the functions above to fulfill the following:

- Protect the fuel cladding
- Limit the amount of energy released in an accident

The ICCMS will also provide post accident monitoring functions of:

- Subcooling Margin
- HPI flow margin
- Degrees of superheat

Description of Analyses and Evaluations

The ICCMS is being designed and evaluated to ensure that it is capable of performing the required accident mitigation functions at EPU conditions. This evaluation included a review of ICCMS instrument ranges, and trip settings to assure the ICCMS will be capable of performing its function. Additionally, the ICCMS is designed to appropriately address human factor considerations. An alternate capability of initiating these accident mitigation functions, utilizing the appropriate ICCMS displays and controls with manual operator action, will provide defense in depth capability to assure the required functions are achieved when necessary (reference Section 2.11, Human Factors).

The three channels comprising ICCMS each receive input from safety related incore thermocouples and RCS pressure instruments to generate core subcooling values. These values will be compared with an adequate subcooling margin curve that, when combined with a coincident sustained reactor trip signal, will generate a LOSM channel initiation. Sustained two-out-of- three channel initiation logic will be used to initiate an actuation train that results in an RCP trip within one minute and also results in raising the EFIC steam generator secondary side water level control setting to the Inadequate Subcooling Margin (ISCM) setpoint.

Additionally, each ICCMS channel receives HPI flow input. The total HPI flow is compared to a generated curve of HPI flow versus RCS Pressure to determine inadequate HPI flow. Should the LOSM initiation signal in conjunction with inadequate HPI flow be sustained, a separate two-out-of- three channel initiation logic will be used to initiate the FCS.

These modifications are described in detail in Appendix E. The ICCMS automatic actuation features are credited in the mitigation of certain loss of coolant accidents (LOCA) as summarized in Section 2.8.5.6.3. The following summarizes the analyses related to the ICCMS system credited functions:

- For a specific range of small break LOCAs, HPI flow (assuming worse case single failure) by itself may not be sufficient to protect the core. The ICCMS was developed to automatically actuate the FCS to support the safety analysis.
- A specific range of small break LOCA analyses requires the trip of the RCP within one minute following the loss of adequate subcooling margin in order to minimize the rate of inventory loss which would decrease the time to the core becoming uncovered. This is an existing operator manual action. Automating this function will provide enhanced assurance of the pump trip within the required time frame.
- Small break LOCA analysis requires the raising of the steam generator secondary side water level to the inadequate core cooling level within 10 minutes following the loss of adequate subcooling margin. This level is necessary in the boiler condenser phase of accident mitigation. This is an existing operator manual action. Automating this function will provide enhanced assurance of the steam generator secondary side level change within the required time frame.

Results

Based on the analysis, the ICCMS monitoring and actuation capabilities will support the EPU.

2.4.2.3.3 Conclusion

See Section 2.4 for Conclusion.

2.4.2.3.4 References
None

2.4.3 Remote Shutdown System (RSS)

2.4.3.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.3.2 Technical Evaluation

Introduction

The purpose of the RSS is to provide the capability of bringing the plant to a safe cold shutdown condition from a remote location if the Main Control Room (MCR) must be evacuated due to uninhabitable conditions (fire, toxic gas, etc.). Control and indication for components needed for safe shutdown are provided either on the Remote Shutdown Panel (RSP) located outside of the MCR or at local control stations. FSAR Table 7-10 lists key equipment for remote shutdown and the location of this equipment in the plant.

Description of Analyses and Evaluations

Plant controls and information displays provided to the operator on the RSP will remain unchanged and the panel will continue to provide the essential controls and instrumentation required to safely shutdown the plant. The following modifications to the Atmospheric Dump Valves (ADVs) and their control systems are being installed in support of the EPU and will impact ADV control from the RSP:

- Atmospheric Dump Valve Replacement – to support Appendix R cooldown requirements for EPU the ADV capacity is being increased by the replacement of the ADVs with new and larger capacity valves. This will result in a significant increase in steaming capability from the ADV control stations on the RSP, which will continue to support achieving cold shutdown within 72 hours.

- ADV Override system – Larger ADVs increase the severity of a fire induced spurious opening of an ADV. To prevent or terminate this, a manual ADV override system is being installed. The override will be actuated by operator action from the control room for control complex fires, and during a control room evacuation. The override will trip two remote lockout devices which block all control signals to the ADV positioners. To restore ADV control capability to the RSP control stations the operators will be required to (1) manually transfer plant control to the RSP using existing transfer switches (an existing operator action) and (2) reset the two remote ADV override lockout devices. Restoration of ADV control capability involves two simple operator control manipulations that are not time critical for either maintaining hot shutdown or achieving cold shutdown within 72 hours. Therefore, this modification does not adversely impact the performance or timing of operator actions performed from the RSP. Refer to Section 2.11 for discussion of the procedure impacts and changes to operator actions.

Results

The Remote Shutdown System, with the changes to include the manual ADV override switches and ADV override lockout devices, continues to function to provide the required remote shutdown capability under the EPU conditions.

2.4.3.3 Conclusion

See Section 2.4 for Conclusion.

2.4.3.4 References

None

2.4.4 Control Systems

2.4.4.1 Control Rod Drive Control System (CRDCS)

2.4.4.1.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.4.1.2 Technical Evaluation

Introduction

The CRDCS drives the control rods into and out of the core to control power level in response to Integrate Control System (ICS) demands (based on reactivity effects due to Doppler, xenon, and moderator coefficient changes) and in response to operator control. The CRDCS also provides rapid rod insertion in response to ICS commands. The CRDCS reactivity rate-of-change limits are set by a combination of movement speed, group size, and group arrangement. The CRDCS is described in FSAR Section 7.2.2.

The CRDCS safety considerations are (1) the control rods are inserted into the core upon receipt of Reactor Protection System (RPS) trip signals, (2) the trip command has priority over all other commands and (3) no single failure shall inhibit the protective action of the CRDCS.

The acceptance criteria for satisfactory performance of the existing Control Rod Drive Mechanisms are described in FSAR Section 3.2.4.3.

Description of Analyses and Evaluations

For the EPU, the effects to the Control Rod Drive (CRD) system for the EPU conditions are primarily mechanical with increased thermal stresses and increased heat loads due to the higher reactor vessel head temperatures and hydraulic, cyclic and seismic forces associated with transient and accident conditions. See Section 2.8.4.1, Functional Design of Control Rod Drive System.

For the EPU, there are no design changes for the CRDCS. There are no changes for the RPS trip signal(s) that initiate rod insertions, no additional startup or operational considerations, and no changes to continuous rod position indication. No other plant operating parameters were identified that could have an impact on the CRDCS.

The design function of the CRDCS interface with the Anticipated Transient Without Scram Mitigating System Actuating Circuitry (AMSAC) System will not be modified with the EPU.

Results

The design function of the CRDCS remains functional and supports the power uprate.

2.4.4.2 Integrated Control System (ICS)

2.4.4.2.1 Regulatory Evaluation

See Section 2.4 for Regulatory Evaluation

2.4.4.2.2 Technical Evaluation

Introduction

The ICS is a non-safety system that automatically controls the station in response to commands preset by the operator. The ICS provides control rod motion, feedwater control and turbine control under all operating conditions. The ICS includes four independent subsystems, which are the Unit Load Demand (ULD) control, the integrated master control, the steam generator control, and the reactor control. The automated unit load demand (AULD) automates the establishment of the unit load demand based upon rated thermal power output, considered part of the ICS. The ICS is described in FSAR Section 7.2.3.

The ICS is designed to maintain a constant average reactor coolant temperature (i.e., 582°F at the EPU conditions) from approximately 25% to 100% of RTP. The ICS performs the following functions:

- Reactivity control by regulation of rod insertion and withdrawal,
- Core heat removal by control of the main turbine bypass valves, main feedwater block valves, startup feedwater control valves, low load feedwater valves, main feedwater pump turbines control through the pump speed controllers, and control of main turbine through the electro hydraulic control system,
- Unit control with minimum operator participation by fully integrating and coordinating the turbine, turbine bypass, reactor, and steam generator feed systems above approximately 25% RTP,
- Balance between reactor power, steam generator feedwater flow, and turbine-generator electric load,
- Automatic runback and/or limiting action on reduced capability of the reactor, feedwater, or turbine-generator systems, and
- Interlocks on reaching various system operating limits.

Description of Analyses and Evaluations

The ICS was evaluated to ensure that it is capable of performing the design functions at the EPU conditions. The existing design parameters of the systems/components listed below were compared with the EPU conditions.

- Megawatt meter range

- Module functional ranges
- IC subsystems: ULD control, the integrated master control, the steam generator control, and the reactor control.

Results

The impacts and necessary design changes to the ICS have been evaluated for the EPU conditions with the following required impacts:

- Transducers for the megawatt meter will be replaced and rescaled to 0-1200 MW to match the indicated megawatt meter range for the EPU maximum MWe.
- IC System adjustments will be made such that IC System control is acceptable for the EPU operation. The IC System changes will include the conversion constant between MWe and MWt in the AULD because efficiency will be improved by some of the EPU changes.
- IC System adjustment of operation restriction limits for initiation of load limiting and runback functions will be made such that control is acceptable for the EPU operation.
- ICS scaling and function curves will be revised as necessary for the EPU including new baseline characterization/function generator curves for Steam Generator/Reactor Demand to Feedwater Demand, Steam Generator/Reactor Demand to Reactor Demand, Feedwater Demand to Feedwater Temperature, and Feedwater Loop Demands to Feedwater Pump Demands.
- Settings for ICS modules will be adjusted for the EPU conditions to reflect applicable maximum MWe or FW flow.

ICS will be modified to support an increase in the electrical generation of the station allowed by the 17R phase of the EPU (See Appendix E. Major Plant Modifications). Several modules within the ICS, which process the unit load demand, are referenced to a nominal electrical output value. In order for these effects to be suitable, certain modules must be referenced to the new electrical output value, as well as full load main feedwater values.

Additional ICS modification(s) will be made consistent with other EPU modification final design details, modeling and/or testing results that impact ICS functions. As an example: changes to runback targets or rates associated with the main feedwater pump, feedwater booster pump, and RCP trip runbacks discussed in Section 2.12.1, Approach to EPU Power Level and Test Plan, will be implemented

- The loss of RCP runback target will be reduced from 75% to approximately 70%.
- The Main Feedwater Pump trip runback target is being lowered from 50% to approximately 40%.
- The Feedwater Booster Pump trip runback target is being lowered from 50% to approximately 40%

Additional changes associated with EPU include:

- The Main Steam Header Pressure post-trip bias will be reduced from 125 psig to 95 psig.

- The asymmetric rod runback is being removed from the ICS.

These changes will not impact the overall operation of the ICS. The review also ensured that failures of the Integrated Control System as modified does not affect safety functions. Therefore, this system will support the uprate to the new power level.

2.4.4.2.3 Conclusion

See Section 2.4 for Conclusion.

2.4.4.2.4 References

None.

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

2.5.1.1.1.1 Regulatory Evaluation

The CR-3 review in the area of flood protection is to ensure that safety-related SSCs are protected from flooding from internal sources, such as those caused by failures of tanks and vessels. The review focused on increases of fluid volumes in tanks and vessels assumed in flooding analyses to assess the impact of any additional fluid on the flooding protection that is provided.

The NRC's acceptance criteria for Flood Protection are based on:

- GDC-2

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria for Flood Protection:

- FSAR Section 1.4.2, Performance Standards. [GDC-2]

2.5.1.1.1.2 Technical Evaluation

Introduction

This section addresses the impact that the EPU will have on the consequences of Flood Protection due to failure of tanks and vessels. Flooding is also addressed in the following sections:

- Flooding due to high energy line breaks in the Intermediate Building and Turbine Building is addressed in Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.
- Submergence inside containment is addressed in Section 2.3.1, Environmental Qualification of Electrical Equipment.
- Flooding is also addressed in Section 2.5.1.1.2, Equipment and Floor Drains.
- Protection of the Control Complex Building, Turbine Building, and Auxiliary Building from flooding due to a break and/or leakage in the circulating water system is addressed in Section 2.5.1.1.3, Circulating Water System.

Description of Analyses and Evaluations

No changes associated with sources of flooding (e.g., sizing of tanks and vessels) are being made as a result of the EPU in the Auxiliary Building where safety-related equipment would be impacted. The flood levels will remain below the level at which safety-related equipment will be affected. The EPU has no impact on internal flooding in the Auxiliary Building rooms.

No changes associated with sources of flooding (e.g., sizing of tanks and vessels) are being made as a result of the EPU in the Turbine Building where safety-related equipment would be impacted. The flood levels will remain below the level at which safety-related equipment will be affected. The EPU has no impact on internal flooding in the Turbine Building.

All Control Complex equipment is located at sufficient building elevations to preclude damage caused by flooding. No changes associated with sources of flooding (e.g., sizing of tanks and vessels) are being made as a result of the EPU in the Control Complex.

The evaluation of the impact of the EPU on protection from internal flooding outside of containment was based on:

- Flooding Due to Failure of Tanks

  For the EPU, there are no new tanks and no changes in the size of existing tanks located in the Auxiliary Building or in the amount of fluid in these tanks that could lead to flooding due to failure of the tanks. The EPU does not affect the volume of water in the condensate storage tank located outside of the Turbine Building. Therefore, the EPU is bounded by the results of existing analyses of flooding due to failure of tanks and process equipment in the Turbine Building and the Auxiliary Building.

- Flooding Due to Failure of Components Other than Tanks

  CR-3 is not increasing any liquid-volume inventories within systems outside containment, including the main condenser; therefore, the EPU does not affect flooding due to failure of such systems. Furthermore, the EPU does not affect the fire suppression systems in the Auxiliary Building, the Turbine Building, and the Control Complex Building, and therefore does not affect the actuation of the fire protection sprinklers and potential to flood safety-related equipment in these buildings.

Results

Internal flooding that would adversely affect safety-related equipment in the Turbine Building, Auxiliary Building, or the Control Complex Building is not a concern because CR-3 is not increasing any liquid-volume inventories and does not add any new tanks or increase the capacity of existing tanks. The EPU also does not affect the previously analyzed fire suppression system with respect to internal flooding. Therefore, the previously analyzed and accepted internal flooding considerations remain valid for operation at the EPU conditions.

2.5.1.1.1.3 Conclusion

CR-3 has reviewed the proposed changes in fluid volumes in tanks and vessels for the proposed EPU. The CR-3 review concludes that SSCs important to safety will continue to be protected from flooding and will continue to meet the requirements of FSAR Section 1.4.2 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Flood Protection.

2.5.1.1.1.4 References

None

2.5.1.1.2 Equipment and Floor Drains

2.5.1.1.2.1 Regulatory Evaluation

The Equipment and Floor Drainage System (EFDS) ensures that waste liquids, valve and pump leakoffs, and tank drains are directed to the proper area for processing or disposal. The EFDS is designed to handle the volume of leakage expected, prevent a backflow of water that might result from maximum flood levels to areas of the plant containing safety-related equipment, and protect against the potential for inadvertent transfer of contaminated fluids to an uncontaminated drainage system. The CR-3 review of the EFDS included the collection and disposal of liquid effluents outside containment. The review focused on any changes in fluid volumes or pump capacities that are necessary for the proposed EPU and are not consistent with previous assumptions with respect to floor drainage considerations.

The NRC's acceptance criteria for the Equipment and Floor Drains System are based on the following criteria:

- GDC-2 and GDC-4 insofar as they require the EFDS to be designed to withstand the effects of earthquakes and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures).

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.2, Performance Standards; and FSAR Section 1.4.23, Protection Against Multiple Disability for Protection Systems, insofar as they require the EFDS to be designed to withstand the effects of earthquakes, and to be compatible with the environmental conditions (flooding) associated with normal operation, maintenance, testing, and postulated accidents (pipe failures and tank ruptures). [GDC-2 and GDC-4]

2.5.1.1.2.2 Technical Evaluation

Introduction

The EFDS, including routing and control of leakage, and prevention of backflow of water / contaminated fluids to areas of the plant containing safety-related equipment, is addressed in FSAR Section 11.2.1.

Description of Analyses and Evaluations

A review of the EFDS was performed to determine if the EPU and associated modifications would impact the operation and capacity of the EFDS to perform their design functions. The review determined that fluid sources and quantities remain unchanged and therefore, leak-offs and drain flow capacities remain unchanged.

Liquids leaking from process systems and liquids from maintenance activities enter the EFDS during all plant operating modes. The EPU and associated modifications do not affect size or volume of fluid in tanks in plant areas where flooding from these tanks could affect safety-related components. Systems with increased flow rates due to the EPU do not affect the sources of liquid that enter the equipment and floor drains, i.e., valve and pump leakoffs, and tank drains. Therefore, the EPU does not result in additional leakage from these sources which would affect the EFDS.

The function of the installed backflow prevention devices to prevent flooding of safety-related areas, via backflow through floor drains, is not affected by the EPU. No new areas requiring backflow prevention are required at the EPU conditions.

Results

The EPU does not impact the volume of liquids or introduce additional sources that enter the EFDS from operations and maintenance activities (system volume required to be drained for outage purposes will not increase as a result of the EPU). The EPU does not affect the sources and quantities of liquids entering drains in plant areas where flooding from these tanks could affect safety-related components.

The EFDS pump capacity remains adequate to handle the volume of leakage expected without modification.

2.5.1.1.2.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the EFDS and concludes that the assessment has adequately accounted for the plant changes resulting in increased water volumes and larger capacity pumps or piping systems. CR-3 concludes that the EFDS has sufficient capacity to: (1) handle the expected leakage resulting from plant changes, (2) prevent the backflow of water to areas with safety-related equipment, and (3) ensures that contaminated fluids are not transferred to non-contaminated drainage systems. Based on this, CR-3 concludes that the EFDS will continue to meet CR-3's current licensing basis with respect to the requirements of FSAR Sections 1.4.2 and 1.4.23 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the equipment and floor drainage system.

2.5.1.1.2.4 References

None

2.5.1.1.3 Circulating Water System

2.5.1.1.3.1 Regulatory Evaluation

The Circulating Water System (CWS) provides a continuous supply of cooling water to the Main Condenser to remove the heat rejected by the turbine cycle and auxiliary systems. The CR-3 review of the CWS focused on changes in flooding analyses that are necessary due to increases in fluid volumes or installation of larger capacity pumps or piping needed to accommodate the proposed EPU.

The NRC's acceptance criteria for the Circulating Water System are based on:

- GDC-4, for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of SSCs.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

Additionally, FSAR Section 9.5.2.3.2 provides criteria for the effects of flooding of safety-related areas due to leakage from the CWS and the effects of malfunction or failure of a component or piping of the CWS on the functional performance capabilities of safety-related SSCs. [GDC-4]

2.5.1.1.3.2 Technical Evaluation

Introduction

Protection of safety-related equipment from flooding due to a break or leakage in the CWS is discussed in FSAR Section 9.5.2. This protection consists of (1) tripping the CWS pumps by level switches installed in either or both condensate pump pits and/or a high-level sump detection in the Turbine Building and (2) limiting the flood height from water that may escape from the CWS into the Turbine Building, and then into the adjacent Control Complex Building and Auxiliary Building, to less than 7.0 inches with all doorways open. This is below the level, which safety-related equipment is located.

As discussed in FSAR Section 9.5.2.3.2, a failure of one of the four CWS conduits, particularly a rubber expansion joint rupture, was determined to be a design basis flooding concern. The level-switch circuitry, with main control board (MCB) annunciation, for tripping the CWS pumps is installed in both condensate pump pits and in the Turbine Building sump to assist in early detection of rising water in the Turbine Building.

Furthermore, CR-3 installed encapsulation sleeves around the eight 90 inch-diameter CWS piping expansion joints located in the Turbine Building. These sleeves will act to restrict flow, so that the operator is afforded 30 minutes to determine the source of the leak and to de-energize the appropriate CWS pump. Per FSAR 9.5.2.3.2, flooding in this 30 minute period will be limited to less than 7.0 inches, which is below the level at which safety-related equipment in the adjacent Auxiliary Building is located.

Damage by internal flooding of the Control Complex Building due to potential flooding in the Turbine Building is not considered a credible accident.

Description of Analyses and Evaluations

The impact of the EPU on analyses and design features related to internal flooding due to leakage or a break in the CWS was evaluated. The CWS flow rates and pump capacities do not change at the EPU conditions. There are no modifications to the CWS resulting from the EPU. Therefore, for the Turbine Building, Auxiliary Building, and the Control Complex Building, the analyses and design features related to internal flooding due to leakage or a break in the CWS for existing plant conditions are unaffected by the EPU; protection of safety-related equipment continues to be provided.

Results

The CWS flow rates and pump capacities do not change at the EPU conditions. There are no modifications to the CWS resulting from the EPU. The analyses and design features related to internal flooding due to leakage or a break in the CWS for existing plant conditions are unaffected by the EPU. Therefore, the CWS will be able to provide a continuous supply of cooling water to the Main Condenser to remove the heat rejected by the turbine cycle and auxiliary systems at the EPU conditions without impacting the functional performance capabilities of safety-related SSCs.

2.5.1.1.3.3 Conclusion

CR-3 has reviewed the protection of safety-related equipment from flooding due to a break or leakage in the CWS. CR-3 concludes that there are no impacts on flooding analyses since there are no modifications to the CWS resulting from the proposed EPU and CR-3 will continue to meet the requirements of FSAR Section 9.5.2.3.2. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Circulating Water System.

2.5.1.1.3.4 References

None

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

2.5.1.2.1 Regulatory Evaluation

The CR-3 review concerns missiles generated from in-plant component over-speed failures and high pressure system ruptures. CR-3's review of potential missile sources covered pressurized components and systems, and high-speed rotating machinery. The review was conducted to ensure that SSCs are adequately protected from internally generated missiles. In addition, for cases where safety-related SSCs are located in areas containing non-safety-related SSCs, CR-3 reviewed the non-safety-related SSCs to ensure that their failure will not preclude the intended safety function of the safety-related SSCs. CR-3's review focused on any CR-3 EPU increases in system pressures or component over-speed conditions that could result during plant operation, anticipated operational occurrences, or changes in existing system configurations such that missile barrier considerations could be affected.

The NRC's acceptance criteria for Internally Generated Missiles are based on:

- GDC-4

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, Missile Protection. [GDC-4].

2.5.1.2.1.2 Technical Evaluation

<u>Introduction</u>

The CR-3 piping and components have been evaluated to determine if the proposed EPU will introduce any new missiles that could potentially damage safety-related SSCs. Included in this evaluation was a review of CR-3 modifications necessary to facilitate implementation of the EPU. The potential missiles considered are those that could be generated by either safety-related or non-safety-related piping system failures, component failures in high energy systems or rotating components subject to over-speed conditions. FSAR Sections 4.2.7, 5.2.3.2.2, 5.4.5.3, 6.1.2.6, and 9.9 discuss the measures taken to protect safety-related SSCs at CR-3 against internally generated missiles.

<u>Description of Analyses and Evaluations</u>

Missiles which are generated either inside or outside containment may cause damage to SSCs that are necessary for the safe shutdown of the reactor or for accident mitigation or may cause damage to the SSCs whose failure could result in a significant release of radioactivity. Examples of the sources of potential missiles are pressure retaining sub-components of SSCs, pressure retaining piping, and rotating equipment subject to over-speed conditions.

The modifications necessary to facilitate the EPU have been reviewed for, and detailed implementation designs are being developed to address, the introduction of new components and/or related system attributes that could potentially introduce a new missile source. New missile sources resulting from modifications, such as the atmospheric dump valve (ADV) air supply, are addressed within the modification and appropriate missile barriers included with the design. The modifications considered are summarized in Appendix E. Major Plant Modifications. The effects of the EPU conditions for existing system configurations such that missile barrier considerations inside or outside the containment building could be affected have also been reviewed. The review of existing SSCs identified no new missile sources from the EPU conditions that could be generated inside or outside of the containment building and potentially damage safety-related SSCs nor any changes affecting any existing missile barriers.

The CR-3 EPU review also focused on any significant increases in system pressures, temperatures, or component over-speed conditions of existing SSCs inside or outside of containment that would be introduced during plant operation or during anticipated operational occurrences. A review of the new operating conditions (presented in Section 1 and Table 1.1-1) and NSSS Design Transients (Section 2.2.2.8) has confirmed that the proposed EPU does not adversely impact the system pressures or temperatures of the systems that could generate missiles. For the NSSS, this conclusion is based on the fact the Reactor Coolant System (RCS) operating pressure is not increasing for EPU, RCS peak pressure during transients is not increasing for EPU and the changes in RCS operating and design transient temperatures are negligible with respect to the generation of missiles.

For non-NSSS systems, a review of the Balance of Plant (BOP) piping systems was conducted to identify those components affected by the EPU. The results and conclusions of this review is that the proposed EPU will not introduce any new missile sources from existing systems and/or SSCs. Although there were minimal increases in select system pressures and temperatures, the increases did not exceed the applicable CR-3 design limits.

With respect to existing pressure retaining hardware and sub-components, it is concluded that the EPU will not introduce any changes that could result in new internally generated missile sources. With respect to rotating component overspeed conditions, there is no change to any existing component over-speed conditions or addition of any new machinery subject to over-speed conditions as a result of the EPU. Thus, the proposed EPU does not change the characteristics of the previously evaluated potential missile sources nor add any new potential high energy missile sources to any existing SSCs.

Refer to Section 2.5.1.2.2, “Turbine Generator,” for evaluation of the impact of turbine missiles.

Results

Since the CR-3 EPU does not adversely impact system pressures or temperatures or component over-speed conditions for unmodified SSCs inside and outside of containment, the characteristics of previously evaluated missile sources are unchanged under the EPU conditions. Therefore, the missile protection measures at CR-3 are adequate with respect to the proposed EPU conditions. New missile sources resulting from modifications, such as the ADVs, are addressed within the modification and appropriate missile barriers included with the design.

The results of the evaluations demonstrate that the CR-3 EPU will not adversely impact safety-related SSCs with respect to concerns of internally generated missiles from any safety or non-safety related SSCs.

2.5.1.2.1.3 Conclusion

The CR-3 staff reviewed the changes in pressures, temperatures and configuration of all safety and non-safety related systems (inside and outside containment) that are required for the proposed EPU and concludes that safety-related SSCs will continue to be protected from internally generated missiles. Based on the above, internally generated missiles protection measures will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of FSAR Section 1.4.40.

2.5.1.2.1.4 References

None.

2.5.1.2.2 Turbine Generator

2.5.1.2.2.1 Regulatory Evaluation

The Turbine Control System, steam inlet stop and control valves, low pressure turbine steam intercept and inlet control valves, and extraction steam control valves control the speed and prevent overspeed of the turbine under normal and abnormal conditions, and are thus related to the overall safe operation of the plant. The CR-3 review of the turbine generator focused on the effects of the proposed EPU on the turbine overspeed protection features to ensure that a turbine overspeed condition above the design overspeed is very unlikely.

The NRC's acceptance criteria for the turbine generator are based on:

- GDC-4, related to the protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, Missile Protection, related to the protection of SSCs important to safety from the effects of turbine missiles by providing a turbine overspeed protection system (with suitable redundancy) to minimize the probability of generating turbine missiles. [GDC-4]

2.5.1.2.2.2 Technical Evaluation

Introduction

The three turbines (one high pressure and two low pressure) are being replaced as part of the EPU project. The main electrical generator has also been rebuilt in support of the EPU. A summary of the electrical generator rebuild and the replacement turbine designs are provided in Attachment E. Major Modifications. While the designs changed from a power conversion efficiency perspective, the functional requirements of the turbine overspeed protection system remain unchanged.

High pressure steam enters the turbine through four throttle valves (or turbine stop valves) and four governor valves (or turbine control valves) at the steam chest. There are a two reheat stop valves and two reheat intercept valves (one from each associated Main Steam Reheater) for each of the two low pressure turbines. The Electro-Hydraulic Control System provides the motive force to position these 16 valves. These valves are normally open and reposition closed upon the release of auto-stop oil and subsequent loss of pressure in the emergency trip header. A major function of these valves is to shut off the flow of steam to the turbine in the event the unit exceeds the setting of the overspeed trip.

The mechanical overspeed trip mechanism consists of an eccentric weight mounted in a transverse hole in the turbine shaft. If the speed of the turbine increases to 110%, the centrifugal force overcomes the spring compression and the weight moves outward, striking a trigger which trips the overspeed trip drain valve and releases the auto-stop oil.

The overspeed protection circuit (OPC) provides an additional means of overspeed protection based on an auxiliary speed signal. Upon activation at 103% of rated speed, the OPC closes the governor valves and reheat intercept valves. These valves are released to their normal position when the turbine speed decreases below 103%.

Description of Analyses and Evaluations

In the event of a loss of electrical load on the turbine generator unit, the restraining torque on the turbine rotor unit is lost. The Turbine Control System is designed to close the inlet and extraction valves of both the high pressure and the low pressure turbines. However, the steam energy entrapped in the turbine unit will cause the rotor to accelerate, potentially causing an overspeed condition. The turbine generator overspeed protection features are retained for the upgraded system and vendor analyses have shown these features to be adequate.

Results

The upgraded high pressure and low pressure turbines have a lower calculated overspeed in response to an electrical load reject due to the increased mass of the new turbines. As a result, no changes are required to the overspeed trip settings.

The cumulative probability of a turbine missile at a 100,000 hr inspection interval is 3.4 E-5 which is lower than the NRC criterion based on the Standard Review Plan, Section 3.5.1.3, of 11.42 E-5 for a 100,000 hr inspection interval.

Appropriate aspects of this design will be confirmed via testing as detailed in Section 2.12.1.

2.5.1.2.2.3 Conclusion

The effects of the proposed EPU on the turbine generator have been evaluated and have adequately accounted for the effects of changes in plant conditions on turbine overspeed. CR-3 concludes that the turbine generator will continue to provide adequate turbine overspeed protection to minimize the probability of generating turbine missiles and will continue to meet the requirements of FSAR Section 1.4.40 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the turbine generator.

2.5.1.2.2.4 References

None

2.5.1.3 Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

2.5.1.3.1 Regulatory Evaluation

CR-3 conducted a review of the plant design for protection from piping failures outside containment to ensure that (1) such failures would not cause the loss of needed functions of safety-related systems and (2) the plant could be safely shut down in the event of such failures. The CR-3 review of pipe failures included high and moderate energy fluid system piping located outside containment. The CR-3 review focused on the effects of pipe failures on plant environmental conditions, control room habitability, and access to areas important to safe control of post accident operations where the consequences are not bounded by previous analyses.

The NRC's acceptance criteria for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment are based on:

- GDC-4, which requires, in part, that SSCs important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40 – Missile Protection, which requires, in part, that SSCs important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures, including the effects of pipe whipping and discharging fluids. [GDC-4]

2.5.1.3.2 Technical Evaluation

Introduction

Piping failures are discussed in Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects.

Per Section 2.2.1 and Impell Report 03-0920-1186, Revision 2, April 1992, "Pipe Rupture Analysis Criteria for Outside the Reactor Building at Crystal River Unit 3", (Reference 1) there are no moderate energy lines outside containment.

The high energy line break (HELB) analysis identifies high energy piping system lines subject to failure and the plant safety-related equipment potentially impacted by piping failures, determines the environmental effects resulting from the piping failures, and identifies the protection measures required to mitigate the effects of the piping failures. Section 2.2.1 provides a discussion of the impact of the EPU on pipe break locations. The environmental conditions resulting from this analysis are provided as input into the environmental qualification program in Section 2.3.1, Equipment Qualification of Electrical Equipment.

The evaluation of pipe breaks outside containment considered the areas within the plant, which contain systems required for safe shutdown and / or systems required to mitigate the effects of postulated pipe breaks.

The EPU will result in pressure increases in the main steam, feedwater, steam generator blowdown and auxiliary steam systems. Due to the increased system pressures, the pipe thrust, jet impingement, building pressurization, and environmental temperature analyses were evaluated to document the changes at the EPU. The fluid systems, along with the original and the EPU initial analysis conditions, which may be affected by changes in initial conditions, are summarized in Table 2.5.1.3-1, HELB Outside Containment Fluid Conditions.

Description of Analyses and Evaluations

High Energy Lines

A review of the current licensing basis identification of high energy line breaks outside containment was performed. The identification of the high energy lines does not change as a result of the EPU. The changes to system process conditions will not add or delete systems from the high energy or moderate energy category. The evaluations for the EPU conditions does not create new or revise pipe break locations from those addressed Section 2.2.1.

Pipe Whip and Jet Impingement

The design of jet impingement shields and pipe rupture restraint protection features are based on the pipe break dynamic effects at operating pressure and temperature. An evaluation was performed to determine the effect on these loads at the EPU operating conditions. The changes in the thrust and jet impingement loads are provided in Table 2.5.1.3-2, HELB Thrust and Jet Impingement Loads Evaluation.

The jet impingement shields and pipe restraints, installed as protection for the effects of piping failures, have been reviewed and remain acceptable for the EPU.

Evaluation of Piping Failures

Intermediate Building (IB) Pressurization Analyses:

The Intermediate Building subcompartment pressurization analyses were performed using the same subcompartment model used for the pre-EPU conditions. The subcompartment model was developed using the NRC approved COMPARE/MOD1 computer code.

Per Section 2.2.1, existing criterion for defining pipe break and crack locations and configurations is unaffected by the EPU. The HELB breaks and break locations are unchanged for the EPU. Therefore, the pre- EPU subcompartment COMPARE/MOD1 model was used to evaluate the mass and energy releases based on the EPU conditions. The main steam, feedwater, steam generator blowdown, and auxiliary steam postulated breaks were evaluated and found acceptable at the EPU conditions.

Main Steam and Auxiliary Steam Systems

The 24 inch main steam line break, at elevation 119 feet in the Intermediate Building, is the bounding steam line break in the Intermediate Building. The mass and energy releases were revised to incorporate the EPU conditions and used as the basis for the pressurization analysis. The pre-EPU subcompartment model was used to determine the effects of the revised mass and energy releases. The peak pressures within the building showed some increases in calculated nodal pressures. See Table 2.5.1.3-3, Maximum Intermediate Building HELB Pressures, for a comparison of Intermediate Building HELB peak pressures.

The peak calculated Intermediate Building pressures were used to evaluate the structural integrity of the building. The building floors, pressure tight doors, and walls were evaluated and found acceptable at the EPU conditions for the main steam line break.

Main Feedwater System

The 18 inch feedwater line break, at elevation 119 feet in the Intermediate Building, is the bounding feedwater line break in the Intermediate Building. The 18 inch feedwater line break results in the peak calculated pressure within any subcompartment in the Intermediate Building for all postulated HELB events. The mass and energy releases were revised to incorporate the EPU conditions and used as the basis for the pressurization analysis. The licensing basis subcompartment model was used to determine the peak pressures in the Intermediate Building (Reference 1). The peak pressures within the building showed some increases at the EPU. See Table 2.5.1.3-3, for a comparison of Intermediate Building HELB peak pressures.

The increased peak calculated pressures were used as input to evaluate the structural integrity of the building structure. The building floors, pressure tight doors, and walls were evaluated and found acceptable at the EPU conditions for the 18 inch feedwater line break.

IB Flooding due to HELB

The Main Feedwater/Main Steam Lines are classified as high energy lines (Reference 1) and are analyzed for the effects of High Energy Line Breaks (HELB). Flooding caused by main feedwater pipe ruptures in the Seismic Class I Intermediate Building above elevation 119 feet (referenced to plant datum) cannot impair the operation of required equipment. Water, from a postulated break in either main feedwater line, flows through wall openings, across a partial slab, then down through the grating of the heater bay into the Turbine Room. Eighteen floor-level rectangular slots (scuppers) are provided at column line 307 in the Turbine Room wall, at elevation 119 feet, to divert the feedwater into the Turbine Room. Thirteen of the eighteen scuppers are available for flood management. Floor openings are protected with metal enclosures and pipe openings with sleeves, both extending 3 feet 6 inches above IB floor to prevent water from flowing to floor elevation 95 feet where the turbine and motor driven Emergency Feedwater Pumps (EFPs) are located. The 3 feet-6 inch flood level equates to a feedwater discharge flow rate that is greater than the run-out flow rate of the main feedwater and feedwater booster pumps. Equipment above elevation 119 feet, susceptible to water damage, is located more than 3 feet 6 inches off the IB floor. No path, at elevation 119 feet, exists for flood water to flow to floor elevation 95 feet (Reference 1). Therefore, flooding in the Intermediate Building from an associated main feedwater line break does not affect safety-related equipment or create environmental concerns due to the EPU conditions.

Steam Generator Blowdown (SGBD) System

The 3 inch SGBD line break was also reevaluated at the EPU conditions. The pre-EPU subcompartment model was used to determine the effects of the revised mass and energy releases. The peak pressures within the building showed some minor increases. See Table 2.5.1.3-3 for a comparison of Intermediate Building HELB peak pressures. The Intermediate Building pressures for the steam generator blowdown breaks are bounded by either the main steam line break or feedwater line breaks in the subcompartment model.

Summary of the EPU Impact on Building Temperature Environments

Intermediate Building

For the Intermediate Building, the Main Steam Line Break (MSLB) conditions result in the bounding conditions for all postulated HELB events outside containment. The subcompartment model used for the pre-EPU and EPU conditions used the NRC approved RELAP5/MOD3.2 computer code.

The 24 inch main steam line break, at elevation 119 feet in the Intermediate Building, is the bounding steam line break in the Intermediate Building. The mass and energy releases were revised to incorporate bounding the EPU conditions and used as the basis for the Intermediate Building environmental temperature analysis. The pre-EPU temperature RELAP subcompartment model was used to determine the effects of the revised mass and energy releases.

The peak temperatures within the Intermediate Building at the EPU conditions, and the resulting temperature profiles at the EPU, are bounded by the pre-EPU temperature profiles. Table 2.5.1.3-4, Maximum Intermediate Building MSLB Temperatures - Elevation 119 Feet, provides a summary of the peak calculated temperatures within the Intermediate Building.

All postulated breaks in the main steam, auxiliary steam, feedwater, and steam generator blowdown systems were evaluated at the EPU conditions. The resulting temperature profiles remain bounded by the pre-EPU 24 inch main steam line break.

Other Buildings

Control Building

Environmental conditions in the control building are not affected by high energy pipe failures. There are no high energy lines that penetrate the control building walls or directly adjacent to the control building.

Turbine Building

The Turbine Building temperature is assumed to be bounded by the temperature profiles generated for the Intermediate Building. With a sizeable relief area, the Turbine Building volume is significantly greater than the IB. Therefore, the turbine maximum temperature environments for postulated MSLB in the Turbine Building are bounded by the results for the IB evaluation.

Turbine Building Flooding

The water sources, which can contribute to flooding in the Turbine Building, are the Condensate Storage Tank (CDT-1), Demineralizer Water System , Condenser Hotwell, Deaerator Storage Tank (FWT-1), Main Feedwater piping above the break elevation, and drains from the Low Pressure Feedwater Heaters (CDHE-3A/B and FWHE-5A/B). For the EPU conditions, the flood-water height increases from 4.0 inches to 6.3 inches in the Turbine Building and Auxiliary Building elevation 95 feet locations. This flooding result occurs from a feedwater line break passing the entire volume of discharged water from the available water sources and assumes the doors separating the Turbine Building and Auxiliary Building are open. Furthermore, the post-EPU flood height remains less than the limiting 7 inch height stated in the FSAR 9.5.2.3.2 for the Auxiliary Building. Seven inches is below the level which safety-related equipment will be affected. The flood volume increases for the EPU due to the time involved to cavitate the Condensate Pumps and subsequent drawdown of the deaerator storage tank level by the Feedwater Booster Pumps, but does not result in a flood level detrimental to safety-related equipment.

Auxiliary Building

The Makeup System is the only source of potential high energy piping within the Auxiliary Building. The Makeup System initial conditions are provided in Table 2.5.1.3-1. There is no change in the Reactor Coolant System pressure and no appreciable change in T_{COLD}. Therefore, the Makeup System pressures and temperatures will remain the same, and there is no impact to the Auxiliary Building due to pipe failures.

Results

The changes to system process conditions will not add or delete systems from the high energy category. The evaluations for the EPU conditions do not create new or revise pipe break locations. The existing high energy pipe break locations are not affected by the EPU operating conditions. The jet impingement shields and pipe restraints installed as protection for the effects of piping failures remain acceptable for the EPU. The Intermediate Building structural integrity due to building pressurization remains acceptable for the EPU. The Intermediate Building environmental temperature profiles for the worst case main steam line break is bounded by the current HELB temperature profiles. The Intermediate Building and Turbine Building Flood levels do not exceed the current limiting values as a result of the EPU.

2.5.1.3.3 Conclusion

CR-3 has reviewed the changes that are necessary for the proposed EPU and the proposed operation of the plant. CR-3 concludes that safety-related SSCs will continue to be protected from the dynamic effects of postulated piping failures in fluid systems outside containment and will continue to meet the CR-3 current licensing basis with respect to the requirements of FSAR Section 1.4.40 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Protection Against Piping Failures in Fluid Systems Outside Containment.

2.5.1.3.4 References

1. Impell Report 03-0920-1186, Revision 2, April 1992, “Pipe Rupture Analysis for Outside the Reactor Building at Crystal River Unit 3

Table 2.5.1.3-1: HELB Outside Containment Fluid Conditions

System/(Building)	Pre EPU Analysis Initial Conditions		EPU Analysis Initial Conditions	
	Pressure [psia]	Temperature [°F]	Pressure [psia]	Temperature [°F]
FW (IB & TB)*	1074.7	455.3	1083	460
MS – AS (IB & TB)*	924.7	590	964	591
SGBD (IB & TB)*	939.7	535	980	542.2
MU (AB)	2139.7	131	2139.7	131

***Bounding initial conditions at steam generator or FW pump discharge

Table 2.5.1.3-2: HELB Thrust and Jet Impingement Loads Evaluation

System	Ratio EPU/pre-EPU
Main Steam	1.04
Aux Steam	1.04
Feedwater	1.0 (No Change)
Steam Generator Blowdown	1.084

Table 2.5.1.3-3: Maximum Intermediate Building HELB Pressures

NODE	24-inch MSLBIB119 NW Pressure [psig]		24-inch MSLBIB119 NW Pressure [psig]		18-inch FWLBIB 119 Penetration Pressure [psig]		18-inch FWLBIB 119 NE Pressure [psig]		3-inch SGBDIB95 NE Pressure [psig]	
	EPU	Pre EPU	EPU	Pre EPU	EPU	Pre EPU	EPU	Pre EPU	EPU	Pre EPU
IB119 NW	1.90	1.28	0.68	0.60	0.65	0.43	0.38	0.31	0.05	0.05
IB119 Penetration	1.73	1.47	6.67	5.98	5.92	6.05	0.41	0.35	0.26	0.30
IB 119 NE	1.85	1.35	0.83	0.77	0.70	0.47	8.85	6.16	0.07	0.07
IB95 NW	1.72	1.18	1.78	1.50	1.53	0.87	0.72	0.61	0.24	0.36
IB 95 NE	1.72	1.18	1.76	1.48	1.53	0.80	0.92	0.77	<0.1	0.45
IB 95 Tendon Access	1.77	1.75	2.00	1.70	2.18	NA	0.79	0.68	0.40	0.32

Table 2.5.1.3-4: Maximum Intermediate Building MSLB Temperatures - Elevation 119 Feet

IB Area	EPU		Pre EPU	
	Max. Temp [Time]	MSLB Break Case	Max. Temp [Time]	MSLB Break Case
IB119 Penetration Area	476.8 °F [125.0 sec]	24” DER in Pen Area	495.1 °F [170.0 sec]	24” DER in Pen Area
IB 119 Northwest	430.4 °F [80.0 sec]	24” DER in IB 119 NW	454.0 °F [42.5 sec]	24” DER in IB 119 NW
IB 119 Northeast	344.3 °F [140.0 sec]	24” DER in IB 119 NW	413.0 °F [125.0 sec]	24” DER in IB 119 NW

2.5.1.4 Fire Protection

2.5.1.4.1 Regulatory Evaluation

The purpose of the Fire Protection Program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The CR-3 review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant will continue to be able to achieve and maintain safe shutdown following a fire.

The NRC's acceptance criteria for the Fire Protection Program are based on:

- 10 CFR 50.48 and associated Appendix R to 10 CFR 50, insofar as they require the development of a FPP to ensure, among other things, the capability to safely shutdown the plant;
- GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.3, Fire Protection, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used [GDC-3]; and
- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. [GDC-5]

Additionally, FSAR Section 9.8, and governing documents identified in FSAR Table 9-18, provide criteria for implementation of the CR-3 FPP, insofar as it requires the development of a FPP to ensure, among other things, fire detection and suppression designed to minimize the adverse effects of fires on SSCs important to safety and the capability to safely shutdown the plant. CR-3 License Condition 2.C.(9) Fire Protection requires implementation of the Fire Protection Program discussion in the FSAR.

CR-3 also conforms to the provisions of the "NRC Enforcement Policy: Extension of Discretion Period of Interim Enforcement Policy" (*Federal Register* 73 FR 52705, September 10, 2008), regarding enforcement discretion for certain fire protection issues, as approved in the NRC letter of February 19, 2010 (Accession No. ML100480260.

2.5.1.4.2 Technical Evaluation

Introduction

The Fire Protection Program at CR-3 consists of activities and functions that are performed to minimize the probability and consequences of a postulated fire. In the event of a fire, the program and system designs ensure the capability to shut down the reactor and maintain it in a safe shutdown condition.

Description of Analyses and Evaluations

The Fire Protection Program is designed to provide reasonable assurance, through defense-in-depth, that a fire will not prevent the performance of necessary safe shutdown functions and that radioactive releases to the environment in the event of a fire will be minimized. CR-3 has reviewed the impact of EPU on various elements of the CR-3 Fire Protection Program. The review concluded that the EPU does not adversely affect the elements of the 'classic' fire protection program (i.e., those related to: (a) Organization; (b) Administrative Controls; (c) Fire Brigade; (d) Fire Protection Responsibilities of Plant Personnel; (e) Fire Detection, Suppression and Barriers; (f) Fire Protection Quality Assurance (QA); and (g) the procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown).

However, there are additional considerations related to the Safe Shutdown analysis strategies due to increased decay heat and resultant need for increased atmospheric dump valve (ADV) capacity, as discussed below.

- EPU does not compromise the ability to achieve safe shutdown. Where safe shutdown systems / components (such as larger ADVs or cables) are impacted, safe shutdown mitigative strategies remain available.

- EPU does not compromise the ability to achieve alternative shutdown methods. EPU does not introduce any plant equipment failure modes which will adversely impact the ability to achieve any of the alternative shutdown functions. Where the function or failure mode of any components in the alternative safe shutdown flow paths are affected, alternative shutdown strategies remain available.

Appendix R Fire Events Analyses

Safe Shutdown (SSD) Thermal-Hydraulic (T-H) results are impacted due to increased decay heat. The analyses performed indicated that the capacity of the ADVs needed to be increased to achieve alternate safe shutdown in 72 hours. This formed the basis for sizing the ADVs.

Several sets of enveloping events were analyzed to account for this increase at EPU conditions. Appendix R Plant Cooldown, Appendix R Overcooling and Appendix R Overheating enveloping events were analyzed separately.

1. Appendix R Plant Cooldown events represent a controlled Reactor Coolant System (RCS) cooldown from Hot Full Power (HFP) temperature conditions to cold shutdown conditions using EFW and steam venting through the ADVs. The Appendix R Plant Cooldown analysis determined adequate ADV capacity and EFW inventory required to comply with Appendix R which requires that Cold Shutdown be achieved within 72-hours after reactor shutdown.

2. The following sets of Appendix R Overcooling events were explicitly analyzed including flow from either the Auxiliary Feed-water (FWP-7) or EFW systems as appropriate:

    (a) Uncontrolled injection of EFW flow from single EFW pump to both OTSGs due to a fire causing the failure of the EFW/FWP-7 control valves with and without Reactor Coolant Pumps operating (to address the impact of a Loss of Offsite Power).
    (b) Uncontrolled injection of EFW flow from single EFW pump to a single OTSG due to a fire causing the failure of the EFW/FWP-7 control valve with and without Reactor Coolant Pumps operating (to address the impact of a Loss of Offsite Power).

These events are terminated by existing Operator Manual Actions (OMA)to isolate EFW/FWP-7 and makeup RCS inventory (due to shrinkage) via High Pressure Injection (HPI).

3. The limiting Appendix R Overheating event was analyzed assuming initiation by losing all main feedwater coupled with fire induced failures affecting both trains of EFW. The steam-driven EFW pump was the only EFW pump which was assumed available for mitigation. An existing OMA is assumed to initiate the steam-driven EFW pump and align the flow path. The event is terminated when HPI is initiated manually by the operator. The overheating scenario was analyzed with and without Reactor Coolant Pumps (to address the impact of a Loss of Offsite Power).

Results

Changes to ADV Capacity/Control Circuits, EFW Inventory, and Operator Actions

Based on the results of the Appendix R enveloping Fire Events Analyses, the following system capacity or procedure changes are required:

Atmosphere Dump Valves

The pre-EPU capacity of each ADV is 301,000 lbm/hr at saturated steam conditions (540°F, 948 psig). A nominal capacity of 620,000 lbm/hr at the same steam conditions is provided to meet the Appendix R requirement of plant cooldown in 72 hours. Section 2.5.5.3, Steam Dump System, and Appendix E Major Modifications further discuss the ADV modification. The ADV modifications are described in Appendix E, Enclosure 2. The modification includes an ADV override feature that when actuated from the control room will preclude or terminate a fire-induced spurious actuation in the case of a confirmed Control Complex fire.

EFW Inventory

The pre-EPU EFW inventory necessary to meet the Appendix R required plant cooldown within 72 hours is 390,943 gallons. As outlined in the CR-3 Improved Technical Specification (ITS) Bases, this is met with a combination of the existing EFW tank capacity (which is adequate for 18 hours of EFW operation) and other onsite replacement sources. At EPU conditions, the total volume necessary increases to 563,773 gallons, which reflects 10 hours of operation on the existing EFW tank capacity. The remaining replacement capacity continues to be available from existing onsite sources. The associated ITS Bases are being revised to reflect the change to the existing EFW tank volume capability from 18 hours to 10 hours (refer to Attachment 4 of this submittal).

Operator Actions

The ADV modifications described in Appendix E, Enclosure 2 includes an ADV override feature. This feature is controlled by a manually operated switch located in the Main Control Room (MCR). The ADV Override switch will be operated by the Control Room Operators in the event of a fire anywhere in the Control Complex as discussed in Section 2.11. This action is a proceduralized Control Room action that is feasible and reliable, and is considered an “Operator Action” and not an “Operator Manual Action” (OMA) as defined in Regulatory Guide 1.189, Section 5.3.1.1. Therefore, operation of the ADV Override Switch is considered an allowed action in accordance with the CR-3 FPP. The manual reset of the ADV Override lockout resets this ADV Override feature, will be installed outside of the MCR but in the Control Complex.

Operation of the ADV Override switch during a fire event requiring MCR evacuation will be accomplished prior to MCR evacuation and is credited in accordance with Regulatory Guide 1.189, Revision 2, Section 5.4.4. This action cannot be negated by subsequent spurious actuation signals resulting from a postulated fire.

Reset of the ADV Override lockout for fire areas inside the control complex; but, not requiring MCR evacuation, is not required because manual operation of the ADVs to get to Cold Shutdown entry conditions is performed local to the valves.

Fires requiring MCR evacuation and shutdown from the Remote Shutdown Panel (RSP) require operation of the ADVs from the Remote Shutdown Panel; therefore, re-setting of the ADV Override Lockouts is required to enable operation from the RSP. This action is not required to achieve and maintain Hot Standby; it is only needed to achieve Cold Shutdown and as such, is a Cold Shutdown manual action. Operation of the ADV Override Reset Lockout Relays is in compliance with Appendix R Section III.G.3 and is an allowable OMA.

The requirements for a fire response, contained in CR-3 Abnormal Operating procedures, are not otherwise impacted.

The results from the analyses of Appendix R fire events, in conjunction with the changes listed above, demonstrate that the EPU has no adverse effect on the ability of the systems and personnel to mitigate the effects of an Appendix R fire event with respect to achieving and maintaining safe shutdown in the event of a fire.

2.5.1.4.3 Conclusion

CR-3 has reviewed the fire-related safe shutdown assessment and accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. CR-3 concludes that the FPP will continue to meet the requirements of 10 CFR 50.48 and associated Appendix R as described in FSAR Section 9.8 and will continue to meet the requirements of FSAR Sections 1.4.3 and 1.4.4 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to fire protection.

2.5.1.4.4 References

None

2.5.2 Reactor Coolant Drain Tank

2.5.2.1 Regulatory Evaluation

The pressurizer relief tank, known as the Reactor Coolant Drain Tank (RCDT) at CR-3, is a pressure vessel provided to condense and cool the discharge from the pressurizer safety and relief valves. The tank is designed with a capacity to absorb discharge fluid from the pressurizer relief valve during a specified step-load decrease. The RCDT is not safety-related and is not designed to accept a continuous discharge from the pressurizer. CR-3 conducted a review of the RCDT to ensure that operation of the tank is consistent with transient analyses of related systems at the proposed EPU level, and that failure or malfunction of the RCDT system will not adversely affect safety-related structures, systems, and components (SSCs). The CR-3 review focused on any design changes related to the RCDT and connected piping, and changes related to operational assumptions that are necessary in support of the proposed EPU that are not bounded by previous analyses. In general, the steam condensing capacity of the tank and the tank rupture disk relief capacity should be adequate, taking into consideration the capacity of the pressurizer power-operated relief and safety valves; the piping to the tank should be adequately sized; and systems inside containment should be adequately protected from the effects of high-energy line breaks and moderate-energy line cracks in the pressurizer relief system.

The NRC's acceptance criteria for the Reactor Coolant Drain Tank are based on:

- GDC 2, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes; and
- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are applicable CR-3 specific criteria:

- FSAR Section 1.4.2, Performance Standards, insofar as it requires that SSCs important to safety be designed to withstand the effects of earthquakes [GDC-2]; and
- FSAR Sections 1.4.23, Protection Against Multiple Disability for Protection Systems, and 1.4.40, Missile Protection, insofar as they require that SSCs important to safety be designed to accommodate and be compatible with specified environmental conditions, and be appropriately protected against dynamic effects, including the effects of missiles. [GDC-4]

2.5.2.2 Technical Evaluation

Introduction

The RCDT is described in FSAR Section 4.2.4.5. The pressurizer safety valves and pressurizer power-operated relief valve (PORV) discharge to the RCDT. Principal design parameters of this tank are given in FSAR Table 11-5.

The pressurizer safety valves are required to have adequate capacity to ensure that the Reactor Coolant System (RCS) pressure does not exceed 110% of system design pressure (i.e., 2750 psig). This is the maximum pressure allowed by the ASME Code (Section III, NB-7300 and NC-7300) for the worst case loss of heat sink event. The design of the surge line, safety valve inlet piping, and safety valve discharge piping are also based on the safety valve design capacity.

The RCDT design (including the tank level setpoints) is also based on the total safety valve capacity and conservatively sized to condense and cool a discharge of pressurizer steam equal to 110% of the steam volume above the full-power pressurizer water level setpoint. This sizing basis was selected to ensure the tank could accept the discharge from the pressurizer safety valves following the worst case loss of external load transient. The RCDT is equipped with a rupture disk that has a relief capacity in excess of the combined capacity of the pressurizer safety valves and the PORV.

Description of Analyses and Evaluations

The RCDT was evaluated based on the results of the loss of external load (turbine trip) analysis described in Section 2.8.5.2.1, Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure. The analysis was performed for the RCS design parameters listed in FSAR Table 4-1 at the EPU conditions.

The results of the analysis confirmed that the installed capacity of the two pressurizer safety valves is adequate to preclude RCS over-pressurization at the EPU conditions. The maximum flow to the RCDT from the pressurizer safety valves and the PORV under the EPU conditions is bounded by the maximum flow analyzed for the current design basis. Since the design of the surge line, safety valve inlet piping, safety valve discharge piping, RCDT, tank rupture disk, and sparger pipe are based on the pressurizer safety valve capacity, it can be concluded that these components are also adequate for the EPU conditions.

In addition, the loss of external electrical load transient (turbine trip) analysis for the EPU determined that the mass and energy of the steam discharged from the pressurizer into the RCDT is less than the design basis discharge. Since the current tank level setpoints ensure adequate coolant is maintained in the tank to condense and cool the design bases discharge, these setpoints remain adequate to preclude the tank temperature and pressure from exceeding the design conditions of 300°F and 100 psig at the EPU conditions.

The evaluation for supports was based on the design capability for the pressurizer safety valves, the PORV and the RCDT.

No credit is taken for the PORV opening in the turbine trip analysis. As a result, the design basis for RCS pressure relief to the RCDT is only from the code safety valves. FSAR Table 4-1 indicates that the flow capacity of the PORV is 165,906 lb/hr at the high setpoint of 2450 psig. This is approximately 25 percent of the flow from two code safety valves. Since the analysis demonstrates that ASME III Code maximum pressure limits are not exceeded based on two safety valves lifting under current setpoints, the required flow for PORV operation at the EPU conditions is enveloped.

Interior missiles were looked at for missile protection of the reactor building liner and compared to the criteria listed in FSAR Section 5.2.3.2.2. The types of missiles for which missile protection is provided are:

- Valve stems up to and including the largest size used
- Valve bonnets
- Instrument thimbles
- Various sizes of nuts and bolts
- Reactor vessel head bolts
- Control Rod Drive Mechanisms (CRDMs)

Results

The current design basis for the RCDT, the RCDT rupture disk, sparger, surge line, PORV and safety valve inlet piping, and PORV and safety valve discharge piping remains bounding for a loss of electrical load (turbine trip) at the EPU conditions. In addition, the current RCDT design bounds the EPU analysis for mass and energy addition such that, following implementation of EPU, the RCDT continues to meet its design basis mass and energy addition requirements without any changes in the tank level or pressure set points.

Because the existing design capability for the pressurizer safety valves, the PORV and the RCDT remain bounding for the EPU conditions, the piping and supports associated with the RCDT remain adequate for EPU conditions as well.

The design feature for missiles is not impacted by the EPU because there are no physical changes being made to the structures surrounding the RCDT or the RCDT itself. Therefore, the existing analysis is still applicable and bounding.

2.5.2.3 Conclusion

CR-3 has reviewed the increase in pressurizer discharge to the RCDT as a result of the proposed EPU and concludes that: (1) the RCDT will operate in a manner consistent with transient analyses of related systems, and (2) safety-related SSCs will continue to be protected against failure of the RCDT consistent with the current CR-3 licensing basis under the criteria outlined in FSAR Sections 1.4.2, 1.4.23, and 1.4.40. Therefore, CR-3 finds the proposed EPU acceptable with respect to the design of the Reactor Coolant Drain Tank.

2.5.2.4 References

None

2.5.3 Fission Product Control

2.5.3.1 Fission Product Control Systems and Structures

2.5.3.1.1 Regulatory Evaluation

The CR-3 review of the Fission Product Control Systems and Structures covered the basis for developing the mathematical model for design basis loss of coolant accident (LOCA) dose computations, the values of key parameters, the applicability of important modeling assumptions, and the functional capability of ventilation systems used to control fission product releases. The CR-3 review primarily focused on adverse effects that the proposed EPU may have on the assumptions used in the analyses for the control of fission products.

The NRC's acceptance for Fission Product Control Systems and Structures are based on:

- GDC-41, insofar as it requires that the containment atmosphere cleanup system be provided to reduce the concentration of fission products released to the environment following postulated accidents.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

Additionally, FSAR Sections 6.2.2.1, 6.2.3.1, and 14.2.2.5.10 provide criteria for the post-accident fission product control capabilities of the Reactor Building Spray (BS) System. The BS System is designed to maintain containment sump pH greater than 7.0 to prevent re-volatilization of fission products. [GDC-41]

2.5.3.1.2 Technical Evaluation

Introduction

This section discusses the post-accident fission product control capabilities of the BS System in support of the EPU operation. The Reactor Building and Fuel Handling Building ventilation systems are not credited with fission product control during post-accident conditions.

The BS System is designed to remove fission products from the containment atmosphere and assure iodine collected in the Emergency Core Cooling System (ECCS) emergency sump is not subject to being re-evolved when sprayed back into the Reactor Building (RB) following a LOCA. The BS System consists of two redundant trains. Each train consists of one BS header, a pump, associated piping, valves, and instrumentation. The BS System functionally includes trisodium phosphate dodecahydrate (TSP-C) material stored in three wire mesh baskets in the RB on the 95' elevation. As shown in FSAR Section 6.2.3, at least one train is required to provide sufficient post-accident fission product control.

In the event of a LOCA, the BS System will actuate and spray the RB with borated water from the borated water storage tank (BWST). The TSP-C baskets become submerged in the sprayed fluid accumulating in the RB. The TSP-C raises the pH of the emergency sump fluid to > 7.0. This ensures the iodine in solution will not re-evolve in containment. Following ECCS switchover from the BWST to the emergency sump, the water solution will contain boric acid and trisodium phosphate dodecahydrate (TSP-C). The mixture of water, boric acid, and TSP-C continues to remove post-accident energy and fission products. Each train is capable of delivering borated water into the containment atmosphere initially from the BWST and after switchover from the emergency sump. The BS System minimizes the control room, exclusion area boundary, and low population zone dose following a large break LOCA (see Section 2.9.2, Radiological Consequences Analyses) in accordance with 10 CFR 50.67 limits. The offsite and control room dose analyses, presented in Section 2.9.2, demonstrate the effectiveness of the BS System to minimize the release of radioactivity to the environment following a large break LOCA. The mass of the TSP-C is sufficient to maintain the containment sump pH between 7.0 and 11.0 during post-LOCA conditions.

Description of Analyses and Evaluations

The spray performance of the BS System is not affected by the EPU. The spray removal performance is a function of containment spray flow rate, droplet size, containment volume, spray fall height, and terminal velocity of the droplets. These variables determine the surface area available for elemental iodine removal and sweep out rates of particulates. The bases for the iodine spray removal effectiveness (spray lambda) are not changed due to the EPU. The flow rates prior to and after switchover from the BWST to the emergency sump and spray nozzle design have not changed due to the EPU. Procedures for containment emergency sump pH control ensure that volume and density of the TSP-C are maintained within Improved Technical Specification Section (ITS) 3.6.7 limits which ensures that the pH is maintained greater than 7.0 following a design basis accident (i.e., LOCA). This ensures that iodine is retained in the sump liquid after switchover from the BWST to the emergency sump. Per CR-3 ITS Bases, B 3.6.7, the containment emergency sump pH control (CPCS) is maintained by the TSP-C contained in the CPCS storage baskets. The TSP-C maintains the pH of the spray solution between 7.0 and 11.0 after the switchover from the BWST to the emergency sump. Therefore, the spray lambdas used in the LOCA analysis, discussed in Section 2.9.2, remain valid and the pH in the sump is sufficient to retain the iodine in the sump liquid after the switchover from the BWST to the emergency sump.

The offsite and control room dose analyses, presented in Section 2.9.2, demonstrate the effectiveness of the BS System to minimize the release of radioactivity to the environment following a LOCA to meet the dose acceptance criteria.

The EPU dose calculations use the methodology discussed in Section 2.9.2, Radiological Consequences Analyses Using Alternative Source Terms.

Results

The effect of the EPU is an increase in source term, which is considered in the new LOCA dose analysis discussed in Section 2.9.2. A review of this Section indicates that the BS System, in conjunction with other SSCs, is effective in limiting both control room and offsite dose to within regulatory guidelines in accordance with the 10 CFR 50.67 limits of 5 rem and 25 rem, respectively.

2.5.3.1.3 Conclusion

CR-3 has performed an assessment of the effects of the proposed EPU on the function of the BS System in reducing the concentration of fission products released to the environment following postulated accidents. CR-3 has adequately accounted for the increase in fission products and changes in expected environmental conditions that would result from the proposed EPU (see Section 2.9.2, Radiological Consequences Analyses). CR-3 further concludes that the fission product control systems and structures will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed EPU. Based on this, CR-3 also concludes that the fission product control systems and structures will continue to meet the current licensing basis with respect to the requirements of FSAR Sections 6.2.3 and 14.2.2.5.10 criteria for the post-accident fission product control capabilities of the BS System. Therefore, CR-3 finds the proposed EPU acceptable with respect to fission product Control Systems and Structures.

2.5.3.1.4 References

None

2.5.3.2 Main Condenser Evacuation System

2.5.3.2.1 Regulatory Evaluation

The Main Condenser Evacuation System (MCES) serves two roles: (1) as the "hogging" or startup system, where it initially establishes main condenser vacuum; and (2) the maintenance of condenser vacuum once it has been established. The CR-3 review focused on modifications to the system that may affect gaseous radioactive material handling and release assumptions, and design features to preclude the possibility of an explosion (if the potential for explosive mixtures exists).

The NRC's acceptance criteria for the Main Condenser Evacuation System are based on:

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and
- GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs) and postulated accidents

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.17, Monitoring Radioactivity Release, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs and postulated accidents [GDC-64]; and
- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include means to control the release of radioactive effluents. [GDC-60]

2.5.3.2.2 Technical Evaluation

Introduction

The MCES at CR-3 consists of the Condenser Air Removal (AR) System. The AR System establishes and maintains condenser vacuum. It consists of two 100% capacity air removal pumps (ARPs) that are used for both system startup and the maintenance of condenser vacuum. The system is not safety-related. The non-condensable gases removed by the ARPs are discharged to the atmosphere and are monitored by a radiation monitor, which is used to detect and monitor potential radioactive releases from primary to secondary leakage.

There is not significant potential for explosive gas conditions that are addressed by this system.

Description of Analyses and Evaluations

The AR System has been evaluated with regards to the greater steam flow and will remain adequate for EPU conditions. The system will not be modified as part of EPU, and the gaseous radioactive material handling and release assumptions will remain unchanged.

2.5.3.2.3 Conclusion

The MCES has been evaluated with regards to the greater steam flow at EPU conditions and has been adequately evaluated. The MCES will continue to maintain its ability to support monitoring for releases of radioactive effluents to the environment following implementation of the proposed EPU. The MCES will continue to meet the requirements of FSAR Sections 1.4.17 and 1.4.70. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Main Condenser Evacuation System.

2.5.3.2.4 References

None

2.5.3.3 Turbine Gland Sealing System

2.5.3.3.1 Regulatory Evaluation

The Turbine Gland Sealing System serves to control the release of radioactive material from steam in the turbine to the environment. CR-3 reviewed changes to the turbine gland sealing system with respect to factors that may affect gaseous radioactive material handling (e.g., source of sealing steam, system interfaces, and potential leakage paths).

The NRC's acceptance criteria for the Turbine Gland Sealing System are based on:

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents; and

- GDC-64, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs) and postulated accidents.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.17, Monitoring Radioactivity Release, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs and postulated accidents. [GDC-64]; and

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include means to control the release of radioactive effluents. [GDC-60]

2.5.3.3.2 Technical Evaluation

<u>Introduction</u>

The Turbine Gland Sealing System reduces the potential for air leakage into the turbine casing and steam leakage from the turbine casing into the turbine building. While such steam is not generally radioactive, the normal functioning of the system does reduce the potential for unmonitored releases to the environs. The turbine rotor is designed with labyrinth type gland seals which provide a high resistance to steam or air flow along the shaft. Gland sealing steam is provided to the gland seal chamber to maintain a slight positive pressure under all operating conditions.

Description of Analyses and Evaluations

The new high and low pressure turbine design change packages (addressed in Appendix E. Major Modifications) include associated improvements to the Turbine Gland Sealing System to support condition changes within the turbines. These changes do not result in changes to the source of sealing steam, system interfaces, or potential leakage paths. Gland sealing design continues to ensure slight positive pressure is maintained under all EPU operating conditions thereby controlling release of radioactivity. The existing provisions that have been established for monitoring effluents from the main condenser will continue to provide appropriate gaseous radioactive material handling. Therefore, the proposed EPU has no effect on its ability to perform these functions.

2.5.3.3.3 Conclusion

CR-3 has assessed the changes to the Turbine Gland Sealing System. CR-3 concludes that the Turbine Gland Sealing System will continue to maintain its ability to control and provide monitoring for releases of radioactive materials to the environment consistent with FSAR Sections 1.4.17 and 1.4.70. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Turbine Gland Sealing System.

2.5.3.3.4 References

None

2.5.4 Component Cooling and Decay Heat Removal

2.5.4.1 Spent Fuel Pool Cooling and Cleanup System

2.5.4.1.1 Regulatory Evaluation

The spent fuel pool provides wet storage of spent fuel assemblies. The safety function of the Spent Fuel Cooling and Cleanup (SF) System is to cool the spent fuel assemblies and keep the spent fuel assemblies covered with water during all storage conditions. The CR-3 review for the proposed EPU focused on the effects of the proposed EPU on the capability of the Spent Fuel Cooling and Cleanup System to provide adequate cooling to the spent fuel during all operating and accident conditions.

The NRC's acceptance criteria for the Spent Fuel Cooling and Cleanup System are based on:

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions;
- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operation and accident conditions be provided; and
- GDC-61, insofar as it requires that fuel storage system be designed with RHR capability reflecting the importance to safety of decay heat removal , and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are applicable CR-3 specific criteria:

- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5];
- FSAR Sections 1.4.41, Engineered Safety Features Performance Capability, 1.4.44, Emergency Core Cooling Systems Capability, and 1.4.52, Containment Heat Removal Systems, insofar as they require that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operation and accident conditions be provided [GDC-44]; and
- FSAR Sections 1.4.67, Fuel and Waste Storage Decay Heat, 1.4.68, Fuel and Waste Storage Shielding, and 1.4.69, Protection Against Radioactivity Release From Spent Fuel and Waste Storage, insofar as they require that fuel storage system be designed with RHR capability

reflecting the importance to safety of decay heat removal, and measures to prevent a significant loss of fuel storage coolant inventory under accident conditions. [GDC-61]

2.5.4.1.2 Technical Evaluation

Introduction

The FSAR states that the Spent Fuel Pool Cooling and Cleanup System is designed to provide a reliable means of decay heat removal and to maintain the water clarity in the spent fuel pools. Redundancy is provided with two pumps, two heat exchangers and multiple injection points into the spent fuel pool; however, the system is not required to meet single failure criteria.

As stated in FSAR Section 9.3.1, the SF System is designed to remove decay heat from fuel assemblies stored in the spent fuel pool. The SF System operates in two modes: 1) Normal Storage mode to remove decay heat from newly discharged fuel assemblies plus previously discharged assemblies. Normal storage means discharging 89 fuel assemblies into the pool; and 2) Refueling Storage mode to remove decay heat from a full core offload plus previously discharged fuel assemblies. Refueling Storage (Full core discharge) occurs when all the fuel in the reactor (all 177 fuel assemblies) is placed in the spent fuel pool.

The system also purifies and maintains water clarity in the spent fuel pool. Borated water in the spent fuel pool provides radioactive shielding and reactivity control. The spent fuel pool cooling and cleanup piping is arranged so that failure of any line does not drain the spent fuel pool. The heat from the pool is rejected to the service water system and then to the ultimate heat sink.

The spent fuel pool cooling and cleanup system consists of two individual loops, each with pumps, heat exchangers, and associated piping, valves, and hoses. The loops are safety-related and qualified to meet Seismic Category I requirements. The non-safety related portions of the system include the demineralizer, spent fuel coolant filters and associated piping. The heat load for the "Normal Storage" mode before the EPU is 8.74E+06 Btu/hr 30 days after shutdown per FSAR Section 9.3.1. The spent fuel pool heat load for the "Normal Storage" mode after implementation of the EPU will be 12.98E+06 Btu/hr 30 days after shutdown. The evaluation below addresses the ability of the SF cooling system to remove the additional heat load resulting from the EPU.

For the "Refueling Storage" mode, both SF cooling loops are required to operate at the same time to provide decay heat cooling for a full-core offload. The maximum decay heat load for this mode is currently 29.6E+06 Btu/hr as noted in FSAR Section 9.3.1. After the EPU is implemented, the full offload decay heat load is evaluated to be 38.69E+06 Btu/hr. This evaluation also addresses the ability of the SF cooling system to handle the added heat load for the "Refuel Storage" mode under the EPU conditions.

The maximum design temperature of the service water is 110°F for Normal Storage, which is the CR-3 design basis temperature used for safety related evaluations. The maximum assumed service water temperature is 100°F for Refueling Mode. The maximum assumed ultimate heat sink temperature (sea water) is 95°F.

To provide additional defense in depth, the system was designed to allow an alternate path for using the Decay Heat Removal (DH) System to cool the spent fuel pool, by aligning the (DH) System pumps with SF System piping.

The heat removal criteria of the Spent Fuel Pool Cooling and Cleanup System, as described in FSAR Section 9.3.1, are that the system be capable of maintaining the spent fuel pool temperature less than or equal to 160°F during normal plant operation and normal storage operations. Refueling operations are conducted approximately every 24 months and are defined for the purpose of this evaluation as removing approximately one-third of the core (approximately 73 fuel assemblies) from the reactor and placing them in the spent fuel pool. It is noted that, following the EPU, up to 89 fuel assemblies will be transferred to the spent fuel pool each cycle. As a result, heat load calculations conservatively assumed 89 fuel assemblies.

Description of Analyses and Evaluations

The spent fuel pool cooling and cleanup system and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. The evaluations were performed using ORIGEN for decay heat generation in the spent fuel pool (SFP). Appendix B of the EPU License Amendment Request (LAR) discusses the use of ORIGEN for this purpose. The evaluations were conservatively performed for an analyzed NSSS core power of 3014 MW_t. The major impact of the EPU is the potential increase in shutdown time required before core off-loads can be initiated due to the increased decay heat. Alternate heat removal paths were not credited in the evaluation. The evaluation was performed for the SF System and components to ensure design and licensing requirements are met for the following parameters:

- SF System Design Pressure/Temperature
- SF System Flow
- Cooling Capacity - Normal Storage Mode Cooling Capacity - Refueling Storage Mode
- Loss of Cooling
- Purification Subsystem

SF System Design Pressure/Temperature

The current spent fuel pool cooling and cleanup system design temperature and pressure are 250°F and 125 psig respectively. The maximum normal operating temperature currently is 160°F. The normal system pump discharge is 110 feet of head (approximately 48 psig). After implementation of the EPU, the maximum normal operating pool temperature will remain 160°F. This design limit will not be changed as a result of the EPU. Therefore, the existing design pressure and temperature of the system components including: heat exchangers, pumps, valves, piping, demineralizers, strainers, and filters are acceptable at the EPU conditions.

SF System Flow

The current spent fuel pool cooling flow rate to provide for acceptable heat removal in the spent fuel pool heat exchangers is 1300 gpm. Maximum system flow (i.e. design flow) is 1500 gpm per heat exchanger or per loop. The analysis for spent fuel pool heat loads at the EPU conditions continues to use a pump flow of 1300 gpm resulting in a 200 gpm margin in SF system flow capability. SF System flow is acceptable at the EPU conditions.

Cooling Capacity

At the end of Fuel Cycle 18 (R17) and thereafter, up to 89 fuel assemblies will be added to the spent fuel pool during refueling. The analysis determined refueling time to maintain the SFP temperature below 160°F for various Nuclear Services Closed Cycle Cooling Water (SW) temperatures conditions and minimum SW and SF System flows of 900 gpm and 1300 gpm, respectively. Table 2.5.4.1–1: Normal Off-load, shows SF System decay heat removal capacity per loop and the time required after shutdown to ensure that the temperature remains below 160°F with the 89 EPU fuel assemblies added to the spent fuel pool. A heat load variation study applying +/- 4 fuel assemblies is also presented. In addition, the evaluation determined the time to maintain the SFP temperature with increased SW and SF flows with a SW temperature of 100°F. The results are presented in Table 2.5.4.1-2 The SF System is capable of providing cooling for a normal storage at the EPU conditions.

The analysis determined the time after shutdown when the SF cooling System has sufficient capacity to prevent water temperature from exceeding 160°F with a full core discharge of 177 assemblies. The time required after shutdown to prevent exceeding 160°F is presented in Table 2.5.4.1-3. The results in Table 2.5.4.1-3 are conservative in assuming a nominal SF System flow rate of 1300 gpm, SW flow rate of 1250 gpm, SW temperature of 100°F, and a design heat exchanger heat transfer coefficient. Plant data shows that SF System flow rates greater than the nominal 1300 gpm are achievable and that the heat exchangers are fouled less than design, thereby realistically resulting in better heat transfer performance.

Loss of Cooling

FSAR Section 9.3.2.2 states that "ample time is available to assure that protective actions can be taken even in the unlikely event of multiple component failures or complete cooling loss". The actions taken would be the same regardless of whether the EPU is implemented and are not impacted by the EPU implementation. The time to increase the spent fuel pool temperature from 160°F to 190°F as noted in FSAR Section 9.3.2.6.1 is 8 hours prior to the EPU. Under the EPU conditions, with a failure of one SF System train, a full core off-load in the pool, and a starting pool temperature of 160°F, the analysis determined the spent fuel pool would reach 190°F in 9.59 hours. The analysis also determined the heat removal capabilities using one SF pump to supply both SF heat exchangers in parallel. The time required for the SF System heat removal rate to reach the full core offload heat decay load (i.e. cooling) is presented in Table 2.5.4.1-5.

The Borated Water Storage Tank (BWST), Demineralized Water System, and Fire Service are available as make up supplies. These will remain available post-EPU.

Purification Subsystem

The EPU has no impact on the hydraulic portions of the purification subsystem. The current purification flow rate is adequate for the EPU conditions. No equipment changes in the purification loop are required to support the uprate. The purification subsystem may experience a slight increase in the frequency of demineralizer resin replacement due to higher levels of impurities in the pool. However, the Reactor Coolant System (RCS) Cleanup System is adequate to mitigate expected increase in impurity inventory in the primary coolant system due to the EPU prior to transmission to the spent fuel pool.

Results

Continued compliance with the Spent Fuel Cooling and Cleanup System performance requirements was demonstrated at the EPU conditions with no system changes necessary.

The EPU cooldown calculation results are as follows:

- SF System decay heat removal capacity to ensure SFP temperature remains below 160°F with the 89 EPU fuel assemblies placed in the spent fuel can be achieved in 19.91 days with SW temperature at 97°F or lower with one train of SF cooling. This meets the expectation to keep the temperature below 160°F in 23 days in order to facilitate refueling activities.
- With a full core off-load after operating at the EPU conditions for a full fuel cycle, both trains of SF cooling capacity is greater than the decay heat load at 11.24 days after shutdown.
- Starting at 160°F, the time to increase the SFP temperature to 190°F is increased from 8 hours after operating at the pre-EPU conditions to 9.59 hours after operating at the EPU conditions. This is because the core offload for the EPU is delayed from 156 hours to 270 hours.

2.5.4.1.3 Conclusion

CR-3 has reviewed the Spent Fuel Pool Cooling and Cleanup System and concludes that the effects of the proposed EPU on the spent fuel pool cooling function of the system has been adequately addressed. Based on this review, CR-3 further concludes that the Spent Fuel Pool Cooling and Cleanup system will continue to provide sufficient cooling capability to cool the spent fuel pool following implementation of the proposed EPU and will continue to meet the requirements of FSAR Sections 1.4.4 and 1.4.67. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Spent Fuel Pool Cooling and Cleanup System.

2.5.4.1.4 References

None

Table 2.5.4.1 – 1: Normal Off-load minimum SW and SF Flows

SW Temperature (°F)	SW Flow (gpm)	SF Flow (gpm)	Heat Removal Rate (MBtu/hr)	Days after shutdown to maintain SFP temp below 160°F w/ 89 FAs discharged (Days)	Days after shutdown to maintain SFP temp below 160°F w/ 85 FAs discharged (Days)	Days after shutdown to maintain SFP temp below 160°F w/ 93 FAs discharged (Days)
110	900	1300	10.23	43.33	38.76	47.50
100	900	1300	12.22	23.44	21.06	25.61
97	900	1300	12.81	19.91	17.37	22.23
95	900	1300	13.20	17.58	14.97	20.00
90	900	1300	14.18	13.51	12.18	14.72

Table 2.5.4.1 – 2: Normal Off-load with increased SW and SF flows

SW Temperature (°F)	SW Flow (gpm)	SF Flow (gpm)	Heat Removal Rate (MBtu/hr)	Days after shutdown to maintain SFP temp below 160°F w/ 89 FAs discharged (Days)	Days after shutdown to maintain SFP temp below 160°F w/ 85 FAs discharged (Days)	Days after shutdown to maintain SFP temp below 160°F w/ 93 FAs discharged (Days)
100	1500	1300	14.72	12.04	10.64	13.32
100	1250	1300	13.85	14.41	13.12	16.28
100	1500	1500	15.41	10.17	9.25	11.52
100	1250	1500	14.46	12.75	11.38	13.99
100	900	1500	12.68	20.69	18.18	22.98

Table 2.5.4.1 – 3: Full Core Off-load (two trains)

	Time after shutdown to prevent exceeding 160°F (Days)
A and B Loops (Refuel mode – pre-EPU)	6.5
A and B Loops (Refuel mode – post-EPU)	11.24

Table 2.5.4.1 – 4: Full Core Off-load (single train)

	Time from 160°F to 190°F with single train (hours)
Pre-EPU	8.5 @ 156 hours after Shutdown
Post-EPU	9.59 @ 270 hours after Shutdown

Table 2.5.4.1 – 5: One SF Pump Supplying Two Heat Exchangers

SW Temperature (°F)	SW Flow per heat exchanger (gpm)	SF Temperature (°F)	SF Flow per heat exchanger(gpm)	Time before Decay Heat Load is equal to Heat removal pre-EPU (Days)	Time before Decay Heat Load is equal to Heat removal post-EPU (Days)
100	900	190	500	7.63	13.98
100	900	190	600	6.30	10.97
100	900	190	650	5.80	9.81
100	1250	190	600	4.80	8.70
100	1250	160	500	15.08	35.97
100	1250	160	600	12.30	27.26
100	1250	160	650	11.30	25.05
100	1250	177	650	6.25	12.02

2.5.4.2 Nuclear Services and Decay Heat Seawater (RW) System

2.5.4.2.1 Regulatory Evaluation

The Nuclear Services and Decay Heat Seawater (RW) System functions are to provide a cooling source to the Decay Heat Closed Cycle Cooling (DC) System which provides cooling to safety-related pumps and heat removal from the Reactor Coolant System. The functions of the RW System also include providing a heat sink to the Nuclear Services Closed Cycle Cooling Water (SW) System by removing process and operating heat from safety-related components during normal operation, transients, and accidents from the SW System via the SW heat exchangers (SWHEs).

The CR-3 review of the RW System covered the characteristics of the RW System components with respect to their functional performance as affected by adverse operational (i.e., water hammer) conditions, abnormal operational conditions, and accident conditions (e.g., a loss of coolant accident coincident with the loss of offsite power). The CR-3 review focused on the additional heat load that would result from the proposed EPU.

The NRC's acceptance criteria for the Nuclear Services Closed Cycle Cooling Water System are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (e.g., water hammer), maintenance, testing, and postulated accidents;
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and
- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5]; and
- FSAR Sections 1.4.41, Engineered Safety Features Performance Capability, 1.4.44; Emergency Core Cooling Systems Capability; and 1.4.52, Containment Heat Removal Systems, insofar as

they require that a system with the capability to transfer heat loads from SSCs to a heat sink under both normal operating and accident conditions be provided. [GDC-44].

Additionally, FSAR 9.5.1 provides the design basis for the RW System insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (e.g., water hammer), maintenance, testing, and postulated accidents. [GDC-4]

CR-3 response to the requirements of GL 89-13 was submitted to the NRC in FPC Letter 3F1295-11 dated December 6, 1995. The requirements of GL 96-06 and impact as a result of the EPU are discussed in Section 2.5.4.3, Reactor Auxiliary Closed Cycle Cooling Water System.

2.5.4.2.2 Technical Evaluation

Introduction

The Decay Heat Seawater function serves as the heat sink for the Decay Heat Closed Cycle Cooling Water (DC) System, which facilitates the removal of decay heat from the reactor core and the removal of process and operating heat from safety related components associated with decay heat removal following a transient or accident. During plant cooldown below approximately 250°F, the Decay Heat Seawater System removes heat which has been rejected to the DC System by the Decay Heat Removal (DHR) System. The maximum temperature of the intake canal used for the evaluation of safety-related design features cooled by the RW System is based on the Improved Technical Specification 3.7.11 of 95°F.

All structures and components in the decay heat seawater pump suction path are shared by the RW System. The two decay heat seawater pumps are located inside two separate compartments of the nuclear services seawater sump pit. One compartment contains the A train decay heat seawater pump and an emergency nuclear services seawater pump. The other compartment contains the B train decay heat seawater pump, an emergency nuclear services seawater pump, and the normal duty nuclear services seawater pump. A separate underground intake conduit for each compartment connects the associated pump suctions to the intake canal. The system provides cooling water to the tube side of two heat exchangers removing heat from the DC System and subsequently rejecting it to the ultimate heat sink (the Gulf of Mexico) by way of the discharge canal. The two decay heat seawater pumps are nominally 100 percent capacity, each providing sufficient flow for the maximum heat load expected for normal cooldown or following an emergency. Each of the pumps is powered from a separate 4160 volt engineered safeguards (ES) bus.

The Nuclear Services Seawater function serves as the heat sink for the SW System, removing process and operating heat from safety-related components during normal operation, transients and accidents. During normal operation, the normal duty (i.e., non- ES) SW pump provides cooling to essential and non-essential components. Additionally, the SW System provides cooling water to the spent fuel pool cooling heat exchangers during all operating conditions.

The Nuclear Services Seawater function is provided by one normal duty pump and two emergency service pumps located inside two separate compartments of the nuclear services seawater sump pit. One compartment contains the normal service pump, the B train emergency pump, and a decay heat seawater pump. The other compartment contains the A train emergency pump and a decay heat seawater pump. A separate underground intake conduit for each compartment connects the associated pump suctions to the intake canal. The system provides cooling water to the tube side of four heat exchangers removing heat from the SW System and subsequently rejecting it to the ultimate heat sink by way of the discharge canal. The normal duty nuclear services seawater pump is powered by a non-safety related power source and is designed to provide sufficient flow to ensure heat removal during normal operation. The two emergency nuclear services seawater pumps are each nominally 100 percent capacity, providing sufficient flow for the maximum heat load expected following an emergency. Each of the emergency pumps is powered from a separate 4160 volt (ES) bus.

The Nuclear Services Seawater System is designed to Seismic Category I requirements, except for the standpipe drain line. The design and operation along with a list of components served by SW during normal and emergency conditions can be found in the FSAR Section 9.5. Following an Engineered Safeguards Actuation System (ESAS) actuation, SW System flow paths are realigned to provide a reliable source of cooling to essential safeguards equipment which may be supplied by non-safety cooling water systems during normal operations. To ensure these additional heat loads can be accommodated, both emergency pumps are started simultaneously by an ESAS signal to provide adequate cooling in the event of a single active failure which disables one emergency pump.

The CR-3 revised response to GL 89-13 documents the actions performed to comply with GL 89-13. These actions include a commitment to operate the RW System under a program of surveillance and control techniques which minimize the incidence of heat exchanger blockage.

Description of Analyses and Evaluations

The RW System and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. The evaluations compared the existing design parameters of the systems/components with the EPU conditions for the following design aspects:

- RW flow and heat removal requirements
- RW System temperature limits
- Design pressure / temperature of piping and components versus the EPU operating pressures and temperatures
- RW fouling in the DC and SW heat exchangers (DCHEs and SWHEs) (NRC GL 89-13)

Results

The RW System previously analyzed failure effects are not affected by the EPU conditions since the RW System flow rate and pressure does not change at the EPU and no physical changes are being made. The implementation of the EPU does not affect the capability of the RW System to perform its function as demonstrated by the system and component evaluation results described below using the RW System during the postulated cooldown and accident scenarios.

RW Heat Removal Requirements

The RW System inlet temperature (from the intake canal) remains unaffected by the EPU; therefore, the temperature limits for the RW intake will not be exceeded by the EPU conditions. The RW System flow requirements remain unchanged for the EPU conditions. Overall, the heat removal requirements of the RW System increase for both normal full power operation and during accident conditions. The increased heat removal requirements will have a slight affect on the discharge temperatures from the Decay Heat Closed Cycle Cooling Heat Exchangers (DCHEs) and Nuclear Services Closed Cycle Cooling Water Heat Exchangers (SWHEs). Since the heat removal requirements of the RW System are increasing, the outlet temperatures from the DCHEs and SWHEs to the discharge canal will increase due to the EPU conditions. The outlet temperatures of the DCHEs and SWHEs for the EPU operation do not exceed the specified design conditions.

RW System EPU Operating Conditions versus Design Conditions of Piping and Components

The RW System flow rate does not change at the EPU conditions and no physical changes are being made to the system. Therefore, the RW System operating pressures are not affected by the EPU conditions and the existing component design pressures are acceptable.

The higher heat loads from the SW and DC Systems cause the RW outlet temperatures from the DCHEs and the SWHEs to be higher. Since the piping downstream of the DCHEs to the common ductile iron piping is designed to 140°F and the piping downstream of the SWHEs to the ductile iron is designed to 150°F, the design temperature of the RW piping and components downstream of the DCHEs and SWHEs continue to bound the maximum RW operating temperatures at the EPU conditions.

Since the RW System flows, pressures, and temperatures for the EPU operating conditions remain within current design operating conditions and no RW modifications are being implemented for the EPU, the functional performance of the RW System is not affected by the EPU operating conditions. The results of evaluating the RW System operating temperatures for both the pre-EPU and post-EPU operating conditions conclude that the RW System will provide sufficient cooling for SSCs important to safety following the implementation of the proposed EPU.

NRC Generic Letter 89-13

CR-3 operates the RW System under a program of surveillance and control techniques that comply with NRC GL 89-13 requirements. The surveillance program consists of periodic preventative maintenance that inspects and cleans heat exchangers, inspects RW piping lining, and inspects and cleans the RW intake pits. The control technique implemented at CR-3 to minimize macroscopic fouling of the SW and DC heat exchangers is frequent regular maintenance. The established program monitors as-found heat exchanger blockage and evaluates the data against established criteria for removing macrofouling in additional heat exchangers. In addition, RW pump discharge pressure limits have been established to alert operators that excessive macrofouling of the heat exchangers is occurring.

The EPU does not result in changes to the flow rate through the RW System. Accordingly, the surveillance and control techniques used to monitor and reduce bio-fouling induced flow blockage do not require a change as a result of the EPU. Operation at the EPU conditions does not change the maintenance practices and training procedures.

The EPU does not affect the programs, procedures, and activities in place at CR-3 in support of implementation of the requirements of GL 89-13. The program continues to ensure that the RW System remains reliable and operable after EPU.

2.5.4.2.3 Conclusion

CR-3 has assessed the effects of the proposed EPU on the RW System and concludes the assessment has adequately accounted for the effect of the increased heat loads on system performance that would result from the proposed EPU on system performance. CR-3 concludes that the RW System will provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, CR-3 concludes that the RW System will continue to meet the current licensing basis with respect to the requirements of FSAR Sections 1.4.4, 1.4.41, 1.4.44, 1.4.52, and 9.5.1. Additionally, the actions of GL 89-13 will continue and remain valid. Based on the results above, CR-3 finds the proposed EPU acceptable with respect to the Nuclear Services Closed Cycle Cooling Water System.

2.5.4.2.4 References

None

2.5.4.3 Reactor Auxiliary Closed Cycle Cooling Water Systems

2.5.4.3.1 Regulatory Evaluation

The CR-3 review covered Reactor Auxiliary Closed Cycle Cooling Water Systems that are required for (1) safe shutdown during normal operations, anticipated operational occurrences, and mitigating the consequences of accident conditions, or (2) preventing the occurrence of an accident. The Reactor Auxiliary Closed Cooling Water System is comprised of two separate CR-3 systems, the Decay Heat Closed Cycle Cooling Water (DC) System and the Nuclear Services Closed Cycle Cooling Water (SW) System. The functions of the DC System are to remove decay heat from the reactor core and heat from safety related components during normal plant cooldown and following a transient or accident. The functions of the SW System are to remove process and operating heat from safety-related components during normal operations as well as during transient or accident conditions (including the containment coolers). Emphasis was placed on the closed cooling water systems for safety-related components (e.g., Emergency Core Cooling System equipment, ventilation equipment, and reactor shutdown equipment). The CR-3 review focused on the additional heat load that would result from the proposed EPU.

The NRC's acceptance criteria for the Reactor Auxiliary Closed Cycle Cooling Water Systems are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents;
- GDC-5, as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and
- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

Specific criteria of NRC Generic Letter (GL) 89-13, Service Water System Problems Affecting Safety-Related Equipment, and GL 96-06, Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions, are also included in this review.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires that structures, system, and components important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5]; and

- FSAR Sections 1.4.41, Engineered Safety Features Performance Capability, 1.4.44, Emergency Core Cooling Systems Capability, and 1.4.52, Containment Heat Removal Systems, insofar as they require that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided [GDC-44].

Additionally, FSAR Sections 9.5.2.1 and 9.5.2.2 provide acceptance criteria for SW and DC Systems, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation including flow instabilities and attendant loads (i.e., water hammer), maintenance, testing, and postulated accidents. [GDC-4]

The requirements of GL 89-13 and impact as a result of the EPU are discussed in Section 2.5.4.2, Nuclear Services and Decay Heat Seawater (RW) System. The CR-3 implementation of the requirements of GL 96-06 was evaluated and accepted by the NRC as cited in NRC to FPC Response Letter 3N0400-10, dated April 27, 2000.

2.5.4.3.2 Technical Evaluation

Introduction

The decay heat services function is described in FSAR Section 9.5.2.2. The decay heat closed cycle cooling heat exchangers (DCHE) are designed to remove heat from plant components during plant operation, plant cooldown, and post accident conditions. The DC System circulates water through parallel flow paths into various components, where it picks up heat from other systems and transfers the heat to the Decay Heat Sea Water (RW) System via the DC heat exchangers. The maximum temperature of the intake canal used for the evaluation of safety related design features is based on the Improved Technical Specification (ITS) 3.7.11 of 95°F.

The nuclear services cooling water function is described in FSAR Section 9.5.2.1. The SW System is designed to remove heat from plant components during plant operation, plant cooldown, and post accident conditions. The SW System circulates water through parallel flow paths into various components, where it picks up heat from other systems and transfers the heat to the RW System via the SW exchangers. The maximum temperature of the intake canal, used for the evaluation of safety related design features, is based on ITS 3.7.11 of 95°F.

Description of Analyses and Evaluations

The DC System and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. The evaluations compared the existing design parameters of the system/components with the EPU conditions for the following design aspects:

- DCHE performance (flow rates, duty and temperatures) at the increased EPU heat loads during normal cooldown, abnormal transient, and accident conditions

- DC System temperature limits

- Design pressure / temperature of piping and components versus the EPU operating pressures and temperatures

- DC relief valve capacities
- Protection of isolated piping sections from heatup effects (NRC Generic Letter 96-06)

The service water systems and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. The evaluations compared the existing design parameters of the systems/components with the EPU conditions for the following design aspects:

- SW flow and heat removal requirements
- SW System temperature limits
- Design pressure / temperature of piping and components versus the EPU operating pressures and temperatures
- SW relief capacities
- Over pressurization of isolated piping inside containment and boiling / flow blockage / water hammer effects in service water piping to the containment recirculation fan coolers (NRC Generic Letter 96-06)

Results

The DC System previously analyzed failure effects are not affected by the EPU conditions since the DC System flow rate and pressure at the EPU conditions remain bounded by current design and no physical changes are being made.

The DC System provides heat removal from the reactor and transfers the heat ultimately to the environment via the RW System. The DC System provides this capability under both normal operating and accident conditions and is capable of achieving this function considering a single failure. The implementation of the EPU does not affect the capability of the system to perform this function as demonstrated by the system and component evaluation results described below and by the analysis results discussed in Section 2.8.4.4, Residual Heat Removal System, using the RW System during the postulated cooldown and accident scenarios.

DCHE Performance

The DC System is capable of removing the required EPU heat loads with the existing DC System supply flow rates, as discussed in Section 2.8.4.4. Since none of the cooled components require more cooling flow, the existing DC and RW flow rates through the DC heat exchangers are not changed by the EPU.

During normal cooldown, abnormal transient, and accident conditions, the DCHEs are capable of maintaining the cooling water supply temperature to individual cooled components below 115°F. During normal plant cooldown, the EPU heat loads are higher, primarily caused by the decay heat removal heat exchanger (DHHE) that has a higher duty due to the higher reactor decay heat at the EPU power level. The maximum DC heat load during normal cooldown occurs when the residual heat removal (DH) system is first placed in service, approximately six hours after reactor shutdown. During cooldown, the reactor coolant flow through the DHHEs is throttled to limit cooldown of the reactor coolant system to 25°F per half hour (from 280°F to 150°F) and to limit the DCHE outlet temperature to 115°F. As a result of maintaining these limits with the higher EPU heat loads, the normal cooldown is lengthened as described in Section 2.8.4.4.

During accident conditions, the DCHEs remove heat from the containment sump and reactor coolant system via the DHHEs. Similar to normal cooldown, the accident heat loads at the EPU conditions are higher due to the higher reactor decay heat at the EPU power level. The EPU analyses described in Section 2.6.1, Primary Containment Functional Design, confirm that the DCHEs provide sufficient heat removal for mitigation of postulated accidents. This evaluation also confirmed that, during the recirculation mode following postulated loss of coolant accidents, (LOCA) the operation of one RW pump in conjunction with one DCHE and one DC pump, supplies sufficient flow to remove the required heat via the DCHE, including the safety related pump coolers and DC pump air handling units.

The DCHE retains adequate margin for macrofouling when the UHS temperature is at the assumed maximum of 95°F.

<u>DC System Temperature Limits</u>

The DCHE outlet temperature for normal cooldown and accident conditions remain within current operating limits. Since some of the heat loads of the components cooled by the DC are increasing due to the EPU, the DC System outlet temperatures from the DC-cooled components will have a small increase compared to current operating conditions; however, the outlet temperatures from these components are still within current operating limits.

<u>DC System EPU Operating Conditions versus Design Conditions of Piping and Components</u>

The DC System flow rate at the EPU conditions remains bounded by current design and no physical changes are being made to the system. Therefore, the DC System operating pressures are not affected by the EPU conditions and the existing component design pressures are acceptable. The higher heat loads at normal cooldown and accident conditions cause the DC outlet temperatures from cooled components to be higher which, in turn, causes the inlet temperature to the DC heat exchanger to be higher.

The design temperature of the DC heat exchangers, pumps, and piping remain bounding of the maximum DC operating temperatures at the EPU conditions.

<u>DC System Relief Valve Capacities</u>

The DC System fluid temperatures are bounded by the relief valve design. Because the EPU condition is below the system design temperature/pressure, no additional analysis is required to demonstrate acceptability of the system relief valves. The associated DC System volume for the EPU conditions is not increased. Therefore, the surge tank for the DC System is adequate for the EPU operating conditions.

The DC System provides heat removal from the reactor building and transfers the heat ultimately to the environment. The DC System provides this capability under accident conditions and is capable of achieving this function considering a single failure. The implementation of the EPU does not affect the capability of the system to perform this function as demonstrated by the system and component evaluation results described above and by the analysis results discussed in Section 2.6.1.

SW Heat Exchanger Performance

The SW System is capable of removing the required EPU heat loads with the existing SW System supply flow rates. Since none of the cooled components require more cooling flow, the existing SW flow rates through the SW heat exchangers are not changed by the EPU.

The evaluation demonstrated that, during the recirculation phase following a postulated LOCAs, the operation of one emergency SW pump motor and one RW pump in conjunction with three SW heat exchangers are required to remove the heat from the containment recirculation fan coolers, spent fuel coolers, and miscellaneous safety related pumps and fan coolers at the EPU conditions.

The majority of the cooled components are unaffected by the EPU conditions since their functions and heat removal requirements are unrelated to the reactor power level or turbine cycle performance. The components affected by the EPU include the following:

- Spent fuel pool heat exchangers (SFHEs) - removes the higher fuel decay heat at the EPU power level from the spent fuel stored continuously in the spent fuel pool, as discussed in Section 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System

- Containment recirculation fan coolers (RBCUs) - during accident events, removes the additional energy released to containment due to the higher EPU power level, as discussed in Section 2.6.5, Containment Heat Removal

SW System Temperature Limits

During normal operation, the current SWHE outlet temperature has an operational limit. As discussed above, only the SFHEs' and RBCUs' heat loads are affected by the EPU. During normal operation, the RBCU heat loads are removed by the Chilled Water (CH) System. Although there is an increase in the SFHEs heat load due to the EPU, existing margin is adequate to accommodate the increased heat load associated with the SFHEs. Therefore, the current SWHE outlet temperatures limit for normal operating conditions will not be exceeded due to the EPU.

During accident scenarios, the current SWHE outlet temperature has a limit of 110°F. The higher heat due to the SFHEs and RBCUs with the existing SW flow rates causes their SW System outlet temperatures to be higher. The worst case outlet temperature occurs when the Gulf of Mexico and, correspondingly, the RW system supply temperature, is at its temperature limit of 95°F. Currently, CR-3 uses an operating curve to determine the allowable macrofouling of the SWHE based on the ultimate heat sink (UHS) (Gulf of Mexico) water temperature. To accommodate the increased heat loads from the SFHEs and RBCUs, existing margin in plant calculations assists in managing the allowable macrofouling of the SWHE for the post EPU conditions such that the SW outlet temperatures for normal and accident conditions remain within current operating limits.

SW System EPU Operating Conditions versus Design Conditions of Piping and Components

The SW System flow rate does not change at the EPU conditions and no physical changes are being made to the system. Therefore, the SW System operating pressures are not affected by the EPU conditions and the existing component design pressures are acceptable.

The higher heat loads at accident conditions cause the SW outlet temperatures from cooled components to be higher which, in turn, causes the inlet temperature to the SW heat exchanger to be higher.

The design temperature of the SW heat exchangers, pumps, surge tank, and piping remain bounding of the maximum SW operating temperatures at the EPU conditions.

The current SW System inventory transient analysis due to increased temperatures experienced during an accident scenario bounds the EPU operating conditions. Therefore, the SW System inventory (volume) will not exceed current design conditions.

<u>SW System Relief Valve Capacities</u>

The EPU heat loads into the SW System are bound by the current heat loads from the volumetric expansion analysis for the Nuclear Service Closed Cycle Surge Tank. Therefore, the SW System fluid temperatures that affect thermal expansion in the surge tank are bounded by the relief valve design. Because the EPU condition is below the system design temperature/pressure, no additional analysis is required to demonstrate their acceptability of the system relief valves. The associated SW System volume for the EPU conditions is not increased. Therefore, the surge tank for the SW System is adequate for the EPU operating conditions.

<u>NRC Generic Letter 96-06</u>

The issue in NRC GL 96-06 related to the heatup/overpressurization of isolated component cooling water piping inside containment was evaluated by CR-3 with no concerns identified for the component cooling water piping inside containment. This conclusion is not affected by the EPU conditions since there are no physical changes or operational changes required by the EPU that would affect the containment penetration piping or isolation valves. The small increase in post-accident temperature at the EPU conditions raises the containment temperature to less than the value of 281°F used in the original analysis. Since there are no additional DC or SW System lines being added for the EPU, there are no additional lines (from the DC or SW System) that will penetrate the containment. Therefore, no new relief valves are required, and the existing relief valves remain acceptable.

2.5.4.3.3 Conclusion

CR-3 reviewed the assessment of the effects of the proposed EPU on the Reactor Auxiliary Cooling Water System and concludes that the assessment has adequately accounted for the increased heat loads from the proposed EPU on system performance. CR-3 concludes that the Reactor Auxiliary Cooling Water System will continue to be protected from the dynamic effects associated with flow instabilities and provide sufficient cooling for SSCs important to safety following implementation of the proposed EPU. Therefore, CR-3 has determined that the Reactor Auxiliary Cooling Water System will continue to meet the requirements of FSAR Sections 1.4.4, 1.4.41, 1.4.44, and 1.4.52. Based on the discussion above, CR-3 finds the proposed EPU acceptable with respect to the station Reactor Auxiliary Cooling Water Systems.

2.5.4.3.4 References

None

2.5.4.4 Ultimate Heat Sink

2.5.4.4.1 Regulatory Evaluation

The Ultimate Heat Sink (UHS) is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. The CR-3 review focused on the impact that the proposed EPU has on the decay heat removal capability of the UHS. Additionally, the CR-3 review included evaluation of the design-basis UHS temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed.

The NRC's acceptance criteria for the Ultimate Heat Sink System are based on:

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions, and

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of the FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5]; and

- FSAR Sections 1.4.41, Engineered Safety Features Performance Capability, 1.4.44, Emergency Core Cooling Systems Capability, and 1.4.52, Containment Heat Removal Systems, as they require that a system with the capability to transfer heat loads from important to safety SSCs to a heat sink under both normal operating and accident conditions be provided. [GDC-44]

2.5.4.4.2 Technical Evaluation

<u>Introduction</u>

The UHS is the Gulf of Mexico, which provides water to the Nuclear Services and Decay Heat Seawater (RW) and the Circulating Water System (CW) via intake and discharge canals which are linked to the RW and CW Systems at the intake structure. The RW System provides cooling water for heat removal from the Nuclear Services Closed Cycle Cooling Water (SW) and Decay Heat Closed Cycle Cooling Water (DC) Systems and transfers it to the UHS. The SW and DC Systems provide cooling water for heat removal from safety-related heat exchangers.

A maximum gulf water temperature of 95°F and gulf water level at the intake structure ≥ 79 feet plant datum allowed by Improved Technical Specification 3.7.11 are used for the safety-related analyses, which rely on the UHS for heat removal. See Section 2.5.4.2, Nuclear Services and Decay Heat Seawater (RW) System, which describes the cooldown and postulated accident scenarios using the UHS for heat rejection.

The Gulf of Mexico is also used by the non-safety related CW System to provide cooling water for heat removal from the turbine cycle during normal plant power operations.

Description of Analyses and Evaluations

The UHS was evaluated to ensure it is capable of performing its intended function of supplying a reliable water source and heat removal capacity for normal and accident conditions following the EPU. Evaluations related to the UHS are addressed in Section 2.5.4.2.

Gulf of Mexico conditions continue to support previously assumed UHS requirements of a gulf temperature of ≤ 95°F and a gulf water level of ≥ 79 feet plant datum.

Results

The ultimate heat sink continues to provide the required water supply and heat sink capacity for safety related systems at the EPU conditions. The RW System flow requirements for cooling of safety-related heat exchangers are not changed by the EPU.

The RW System water returned to the UHS is at a slightly higher temperature due to the increased heat loads from the EPU NSSS thermal power level at normal operating conditions, during normal cooldown, and from the higher reactor decay heat. However, this higher discharge temperature does not impact the UHS since the RW System intake is independent of the RW and CW Systems discharge. Furthermore, no changes to improved Technical Specification 3.7.11 are required as a result of the EPU.

2.5.4.4.3 Conclusion

The CR-3 assessment has adequately accounted for the effects that the proposed EPU would have on the UHS safety function, including validation of the design-basis UHS temperature limit based on the post-EPU data. CR-3 concludes that the proposed EPU will not compromise the design-basis safety function of the UHS, and that the UHS will continue to satisfy the requirements of FSAR Sections 1.4.4, 1.4.41, 1.4.44, and 1.4.52, following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU is acceptable with respect to the Ultimate Heat Sink.

2.5.4.4.4 References

None

2.5.4.5 Emergency Feedwater System

2.5.4.5.1 Regulatory Evaluation

The Emergency Feedwater (EF) System functions as an emergency system for the removal of heat from the primary system when the Main Feedwater System is not available. The EF System may also be used to provide decay heat removal necessary for withstanding or coping with a station blackout (SBO). CR-3 review for the proposed EPU focused on the system's continued ability to provide sufficient emergency feedwater flow at the expected conditions (e.g., steam generator pressure) to ensure adequate cooling with the increased decay heat. The review also considered the effects of the proposed EPU on the likelihood of creating flow instabilities (e.g., water hammer) during EF operation.

The NRC's acceptance criteria for the Emergency Feedwater System are based on the following:

- GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whip, and discharging fluids that may result from equipment failures;

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions;

- GDC-19, insofar as it requires that equipment at appropriate locations outside the control room be provided with (a) the capability for prompt hot shutdown of the reactor, and (b) a potential capability for subsequent cold shutdown of the reactor;

- GDC-34, insofar as it requires that a Residual Heat Removal system be provided to transfer fission product decay heat and other residual heat from the reactor core, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure; and

- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SCCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found to be acceptable by the NRC for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.23 , Protection Against Multiple Disability for Protection Systems; and FSAR Section 1.4.40, Missile Protection, insofar as they require that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whip, and discharging fluids that may result from equipment failures [GDC-4];

- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5];
- FSAR Section 1.4.11, Control Room, insofar as it requires that equipment at appropriate locations outside the control room be provided with (a) the capability for prompt hot shutdown of the reactor, and (b) a potential capability for subsequent cold shutdown of the reactor [GDC-19];
- FSAR Section 1.4.37, Engineered Safety Features Basis for Design; and FSAR Section 1.4.42, Engineered Safety Features Components Capability, insofar as they require that a Residual Heat Removal system be provided to transfer fission product decay heat and other residual heat from the reactor core, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure [GDC-34]; and
- FSAR Section 1.4.41, Engineered Safety Features Performance Capability, FSAR Section 1.4.44, Emergency Core Cooling Systems Capability, and FSAR Section 1.4.52, Containment Heat Removal Systems, insofar as they require that a system with the capability to transfer heat loads from safety-related SCCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished, assuming a single failure [GDC-44];

2.5.4.5.2 Technical Evaluation

<u>Introduction</u>

As described in FSAR Section 10.5, the EF System is sized to provide sufficient decay heat removal to cooldown the Reactor Coolant System (RCS) to the required pressure and temperature for the Decay Heat Removal (DH) System in response to the following events:

a. Loss of Feedwater (LOFW)

b. FW line break (FWLB)

c. Loss of FW with loss of onsite and offsite AC power (LOOP)

d. Main steam line break (MSLB)

e. Small break loss of coolant accident (SBLOCA)

f. Anticipated Transient Without Scram (ATWS)

g. EF System Shutdown from Outside Control Room

h. Appendix R Cooldown with EF System

i. Hot Standby Operation using EF System

The EF System is seismically qualified and will remain functional following a single failure for any of the above listed cases. See Section 2.2.4, Safety Related Valves and Pumps, for effect of the EPU on safety related valves.

The Emergency Feedwater System flow is increasing for the EPU conditions and contains fast acting valves. The flow rates used in the previous evaluations preformed for the water hammer loads associated with fast closure of these valves envelopes the new EPU flow rates. See Section 2.2.2.2, BOP Piping Components and Supports.

The safety-related EF System consists of one turbine driven pump (EFP-2) and one diesel driven pump (EFP-3). The motor driven pump (EFP-1) is not credited in safety analyses and is not addressed in detail below. The EFP-2 and EFP-3 start automatically to supply EF to both steam generators when a signal is received from the Emergency Feedwater Initiation Control (EFIC) System. Each pump supplies both steam generators through a normally open, motor operated discharge valve in series with normally open solenoid operated control valves, which EFIC uses to control level. The EFP-1 and EFP-2 are located in the Intermediate Building while EFP-3 is located in its own building.

The EFP-1 is interlocked to prevent starting if the EFP-3 is running. All 3 EF system pumps can be aligned to supply water to either once through steam generator (OTSG); however, the motor driven pump must be manually initiated. EFP-1 is a “defense in depth” component and it is not auto-started to mitigate the consequences of an accident. Emergency operating procedures direct plant personnel to operate EFP-1 in a LOFW event should either EFP-2 or EFP-3 be unable to provide sufficient flow to the OTSGs.

The EF System operates following abnormal or accident conditions to provide EF flow to one or both OTSGs. The system does not operate during startup or during plant operation with normal feedwater in service.

<u>Description of Analyses and Evaluations</u>

Hydraulic modeling of the modified EF System was performed using FATHOM. This analysis conservatively assumed the longest flow path, resulting in the highest line losses. As a result this provides a conservative flow assumption for the analyses. The primary impact of the EPU on the EF system is the increased decay heat removal requirements during abnormal and accident conditions. Due to increased decay heat load, EF System flow will be increased as discussed in Appendix E, Major Plant Modifications, to achieve the system capabilities as shown in Table 2.5.4.5-1 for EFP-2 and EFP-3.

The EFP-3 utilizes a Fuel Oil Storage Tank (DFT-4) to store the required fuel oil to operate the pump. With implementation of the EPU, the minimum required capacity of DFT-4 will increase as shown in Table 2.5.4.5-2. Given the 13,000 gallon usable capacity, the existing tank is of adequate size for the EPU conditions. Improved Technical Specification (ITS) Sections 3.7.19 and B3.7.19 specify a minimum capacity that must be maintained for operation to ensure the capability to operate the diesel for 6 days. Analysis was performed to determine the storage requirements for 6-day and 7-day operation of EFP-3. A comparison of Technical Specification requirements before and after implementation of the EPU is provided in Table 2.5.4.5-2.

The EFP-3 lube oil inventory requirements in ITS 3.7.19 remain bounding for the EPU operation. The ITS lube oil inventory requirements for EFP-3 were approved in Amendment 192 on July 17, 2000 (ML0037334850). This volume reflected increased lube oil consumption that was typical of the new diesel engine. Given that lube oil consumption rate decreases as a new engine accumulates operating time (wears-in), and that piston ring seating interface with the cylinders improves with increased loading, (due to increased seating pressure behind the inside diameter of the rings), the lube oil consumption rate at the EPU operating conditions will not exceed the current consumption rate. Based on this, the required EFP-3 lube oil inventory as specified currently in ITS 3.7.19 remains adequate for the increased EFP-3 loading conditions at the EPU.

As a result, Technical Specification and Bases revisions are proposed to support the EPU requirements with respect to the Fuel Oil Storage Tank DFT-4. The ITS and ITS Bases are revised as shown in Attachment 2.

Each of the events listed above in the "Introduction" section have been analyzed under the EPU conditions. Safety analyses were performed using the previously approved RELAP5/MOD2-B&W computer program. A summary of these analyses for events requiring EF system operation is described below:

(a) Loss of Feedwater (LOFW)

The LOFW event is the limiting transient in terms of establishing the minimum EF flow requirements. Since the LOFW bounds the other events, the pumps are evaluated as part of the LOFW event discussion. The EFP-2 and EFP-3 flow requirements were reevaluated for the EPU conditions. Due to the new core thermal output, the required EF flow will increase from 550 gpm within 60 seconds after the initiation setpoint is reached to 660 gpm within 40 seconds after the setpoint is reached. LOFW event details are discussed in Section 2.8.5.2.3, Loss of Normal Feedwater.

(b) Feedwater Line Break (FWLB)

The FWLB event is also affected by the increased heat from the reactor core under the EPU conditions. EF System flow requirements for a FWLB are bounded by those for a LOFW event as addressed in (a) above. FWLB event details are discussed in Section 2.8.5.2.4, Feedwater System Pipe Breaks Inside and Outside Containment.

(c) LOFW with Loss of Offsite Power (LOOP)

EFP-2 is steam driven and EFP-3 is diesel driven, therefore they are not dependent on offsite power. A LOFW with LOOP event is identical to (a) except that the Reactor Coolant Pumps trip, reducing the heat that must be removed by the EF System. Therefore, the minimum EF flow is bounded by (a). LOFW event details are discussed in Section 2.8.5.2.3, Loss of Normal Feedwater Flow.

(d) Main Steam Line Break (MSLB)

An analysis for a MSLB and the capability of the EF System in response under the EPU conditions was performed. The Emergency Feedwater Initiation and Control (EFIC) System

(FSAR section 7.2.4) initiates emergency feedwater from low pressure in either OTSG, however, the Feed Only Good Generator (FOGG) logic prevents emergency feedwater from being provided to the faulted OTSG. The analysis indicates that with EF flow to only the intact generator, (1) the core remains intact and in a coolable geometry (core power does not exceed 112% rated power at the EPU conditions), (2) the reactor does not return to critical when excess cooling is applied, and (3) the Reactor Coolant System (RCS) pressure does not exceed ASME Code design limits. Therefore, the EPU has no effect on the EF System capability with respect to response to the MSLB accident. MSLB event details are discussed in Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment.

(e) Small Break Loss of Coolant Accident (SBLOCA)

The SBLOCA event analysis requires an increase in flow from 200 gpm to each OTSG within 60 seconds of initiation (pre-EPU), to 300 gpm (600 gpm total) within 40 seconds of initiation (post-EPU). This flow is bounded by the LOFW event in (a) above. SBLOCA event details are discussed in Section 2.8.5.6.3, Emergency Core Cooling System and Loss of Coolant Accidents.

(f) ATWS

The ATWS event analysis indicates that EF flow of 275 gpm per OTSG with a maximum injection delay of 40 seconds is adequate. The EFP-2 and EFP-3 operational parameters required for this accident are bounded by those required for the LOFW in (a) above. ATWS event details are discussed in Section 2.8.5.7, Anticipated Transients Without Scrams.

(g) EF System Shutdown Equipment (Outside Control Room)

FSAR Table 7-10 lists CR-3 plant equipment with controls on the Remote Shutdown Panel (RSP) that can be used to perform a safe plant shutdown if habitability of the Control Room is not possible. Operation of EF System equipment from the RSP will be performed in the same manner after the EPU implementation. Remote Shutdown System (RSS) is described in Section 2.4.3, Remote Shutdown System.

(h) Appendix R Cooldown with EF System

Regulatory requirements under 10CFR50 Appendix R require that provisions must be made for cooling down the RCS from normal operating temperature (approximately 545°F) to 200°F within 72 hours from the onset of a fire that has the potential to challenge safe shutdown of the plant. According to engineering analysis, the pre-EPU conditions, the quantity of water that must be provided to the OTSGs by the EF system to accomplish this cooldown is 420,000 gallons. After implementation of the EPU, the quantity of water required for cooldown is 471,422 gallons. The sources available to the EF System total 1,020,000 gallons according to FSAR Table 10-2; therefore, sufficient water inventory is available for the EF System after the EPU implementation. Refer to Section 2.5.1.4, Fire Protection, for details of the Appendix R Cooldown analysis.

(i) Hot Standby Using EF System

The dedicated safety related EF System suction supply tank EFT-2 provides inventory sufficient to remove decay heat from the RCS until other water sources are available. With implementation of the EPU, analysis indicates that the time period the EF System can provide water to the OTSGs from EFT-2 is 10 hours rather than 18 hours. The 10 hours allowed to align alternate sources of water remains sufficient time. The reference in the Technical Specification Bases to this supply duration is revised as shown in Attachment 4. ITS Bases Markup.

Results

Based on the evaluation above, the EF System continues to supply adequate water following the implementation of the EPU. The specific results of the analyses are as follows:

- The motor driven EF pump (EFP-1) continues to provide defense-in-depth supplemental flow to the OTSG under the EPU conditions.
- The turbine driven pump (EFP-2) is capable of supplying the higher flow requirement to the OTSGs under EPU conditions with modifications made to the EF System.
- The diesel driven EF pump (EFP-3) is adequate to provide the higher flow to the OTSGs under the EPU conditions; however modifications made to the EF System provide additional margin for EFP-3.

2.5.4.5.3 Conclusion

The effects of the EPU on the Emergency Feedwater System have been evaluated by the CR-3. The evaluation adequately accounted for the effects of the increase in decay heat and other changes in plant conditions on the ability of the EF System to supply adequate water to the OTSGs to ensure adequate cooling of the core. CR-3 concludes that the EF System will continue meet its design functions following implementation of the proposed EPU. Based on the above discussion, the CR-3 further concludes that the EF System will continue to meet the CR-3 criteria listed in FSAR Sections 1.4.4, 1.4.11, 1.4.23, 1.4.37, 1.4.40, 1.4.41, 1.4.42, 1.4.44 and 1.4.52. Therefore, the CR-3 finds the proposed EPU acceptable with respect to the Emergency Feedwater System.

2.5.4.5.4 References

None

Table 2.5.4.5-1 Emergency Feedwater Parameters

Table 2.5.4.5-1 **Emergency Feedwater Parameters** (1)		
Parameter	**Current Value**	**EPU Value**
Minimum EF Flow (gpm)	550	660
EF Time to Achieve Full Flow (sec)	≤ 60	≤ 40
Water to OTSGs for Appendix R Cooldown (gal)	420,000	471,422
Operator Response Time for Aligning Alternate Water Sources (hrs)	18	10

FOOTNOTE (1): Applicable to EFP-2 and EFP-3. EFP-1 is manually initiated and not credited

Table 2.5.4.5-2 Technical Specification Requirement Comparison Before and After the EPU implementation

Criterion	**DFT-4 Required Fuel Oil Volume, Pre- EPU (gal)**	**DFT-4 Required Fuel Oil Volume, Post- EPU (gal)**
7-day Operation	9,480	9800
6-day Operation	8,335	8600

2.5.5 Balance-of-Plant Systems

2.5.5.1 Main Steam System

2.5.5.1.1 Regulatory Evaluation

The Main Steam (MS) - System transports steam from the NSSS to the power conversion system and various safety-related and non-safety-related auxiliaries. The CR-3 review focused on the effects of the proposed EPU on the system's capability to transport steam to the power conversion system, provide heat sink capacity, supply steam to drive safety system pumps, and withstand adverse dynamic loads (e.g., water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads).

The NRC's acceptance criteria for the Main Steam System are based on the following:

- GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures;
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and
- GDC-34, Residual Heat Removal; insofar as it requires that a Residual Heat Removal System be provided to transfer fission product decay heat and other residual heat from the reactor core.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria:

- FSAR Section 1.4.4, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5];
- FSAR Section 1.4.40, Missile Protection; insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures [GDC-4]; and
- FSAR Section 1.4.42, Engineered Safety Features Components Capability; insofar as it requires that a Residual Heat Removal System be provided to transfer fission product decay heat and other residual heat from the reactor core. [GDC-34]

2.5.5.1.2 Technical Evaluations

Introduction

The Main Steam System is described in FSAR Section 10.2.1.4. The system provides heat removal from the Reactor Coolant System during normal, accident and post accident conditions. During off-normal conditions, the system provides emergency heat removal from the Reactor Coolant System using secondary heat removal capability. System components are also credited for safe shutdown following station blackout events and some fire events.

The MS System is designed to produce dry steam with a small amount of superheat in the steam generators and direct it to the high pressure turbine, as well as other steam driven components and auxiliary systems. The Main Steam System includes the steam piping, main steam safety valves, main steam isolation valves, and other miscellaneous valves and piping. The Main Steam System also provides a flow path for steam from the steam generators to the Steam Dump System (Turbine Bypass and Atmospheric Dump Valves) discussed in Section 2.5.5.3 Steam Dump System.

Description of Analyses and Evaluations

The Main Steam System was evaluated to ensure any changes to the design or operation of the system would not impact the capability of steam-driven equipment to function in accordance with safe shutdown and/or accident analysis assumptions. In addition, the evaluation determined that there were no increased challenges to the Steam Dump System, which could result in increased challenges to reactor safety systems. The evaluations were performed based on the data provided by a PEPSE thermal modeling program to support operation at the EPU conditions.

Additional analyses and evaluations of the Main Steam System are found in the following sections:

- Protection against dynamic effects, including missiles, pipe whip and discharging fluids are evaluated in Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects and Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment;

- Capability to withstand water steam hammer resulting from rapid valve closure and relief valve fluid discharge loads are evaluated in Section 2.2.2.2, Balance of Plant Piping Components and Supports;

- Functions of the Turbine Bypass and Atmospheric Dump Valves are evaluated in Section 2.5.5.3, Steam Dump System; and

- Capability of supporting accident analysis assumptions is evaluated in Section 2.5.4.5, Emergency Feedwater System.

Results

As described in Section 2.5.5.3, the atmospheric dump valves (ADVs), one per steam generator, will be replaced during the 17R outage and are each designed to pass 620,000 pounds per hour of total steam flow at the EPU operating conditions. The replacement ADVs are sized and qualified as safety-related for the mitigation of a Small Break Loss-of-Coolant (SBLOCA) event. These valves are also credited for mitigating Appendix R events using direct manual operation. The changes in the design and operation of the ADVs support the function in accordance with safe shutdown and accident analysis assumptions.

The design capacity the turbine bypass valves will pass from the main steam headers to the main condenser is 22.7% of the EPU total steam flow at 100% rated thermal power. The capacity of the Turbine Bypass System, through the bypass to the condenser, includes 4 new bypass valves rated at 735,000 pounds per hour each (at900 psia and 586°F). Refer to Appendix E, Major Plant Modifications, for further information related to the modification of the turbine bypass valves and refer to Section 2.5.5.3, Steam Dump System, for additional discussion of the Turbine Bypass System.

The Main Steam System supplies steam to the emergency feedwater pump turbine (EFTB-1) and was determined to be acceptable for the EPU conditions as described in Section 2.5.4.5, Emergency Feedwater System.

The Main Steam System's ability to supply steam to auxiliary components, including the turbine gland steam supply and turbine driven emergency feedwater pump, will not be affected by the EPU. None of these steam flow requirements change appreciably due to the EPU conditions. The EPU heat balances (PEPSE Program) include these required auxiliary flows and confirm that sufficient main steam flow exists to ensure the high pressure turbine performance meets the desired EPU power generation requirements.

Summary

As described in Sections 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, 2.2.2.2, Balance of Plant Piping Components and Supports, 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment, and 2.5.4.5, Emergency Feedwater System, the Main Steam System is capable of withstanding steam water hammer and the safety-related portions of the system continue to be protected against the dynamic effects of the EPU, including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures.

As described in Section 2.5.5.3, the changes in the design and operation of the ADVs support the function in accordance with safe shutdown and accident analysis assumptions.

As described in Section 2.5.4.5, Emergency Feedwater System, the changes in the design and operation of the turbine-driven feedwater pump support the function in accordance with safe shutdown and accident analysis assumptions.

2.5.5.1.3 Conclusions

CR-3 reviewed the assessment of the effects of the proposed EPU on the MS System and concludes that the assessment adequately accounted for the effects of changes in plant conditions on the design of the MS System, with the modifications as described. CR-3 concludes that the MS System will maintain its ability to transport steam to the power conversion system, provide heat sink capacity, supply steam to steam-driven safety pumps, and withstand steam hammer. Therefore, CR-3 finds the proposed EPU is acceptable with respect to the Main Steam System.

2.5.5.1.4 References

None

2.5.5.2 Main Condenser

2.5.5.2.1 Regulatory Evaluation

The Main Condenser System is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the Turbine Bypass System. The CR-3 review focused on the steam bypass the effects of the proposed EPU load rejection assumptions, and on the ability of the Main Condenser System to withstand the blowdown effects of steam from the Turbine Bypass System.

The NRC's acceptance criteria for the Main Condenser System are based on the following:

- GDC-60, Control of Releases of Radioactive Materials to the Environment; insofar as it requires that the plant design include means to control the release of radioactive effluents.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria for the Main Condenser System:

- FSAR Section 1.4.70; Control of Releases of Radioactivity to the Environment; insofar as it requires that the plant design include means to control the release of radioactive effluents. [GDC-60]

2.5.5.2.2 Technical Evaluation

Introduction

The CR-3 main condenser is a two-shell, single-pressure, deaerating type surface condenser. The condenser extracts the latent heat from the low pressure turbine exhaust steam, the steam from the Turbine Bypass System (when in operation), and miscellaneous flows, drains and vents during normal plant operation. This heat is transferred to the Circulating Water System. The evaluation of the EPU effect on the circulating water system is described in Section 2.5.1.1.3, Circulating Water. The resulting condensate is collected in the condenser hotwell before entering the Condensate and Feedwater System. The condensate hotwell level control system maintains sufficient level to provide the suction head for the condensate pumps. Deaeration is provided by the Main Condenser Evacuation System with continuous monitoring to detect high radiation levels.

Upon loss of electrical load, energy in the Reactor Coolant System will be dissipated by relieving steam to the condenser and/or the atmosphere. The design capacity the turbine bypass valves will pass from the main steam headers to the main condenser is 22.7% of the EPU total steam flow at 100% rated thermal power. Larger turbine bypass valves will be installed and will provide the Turbine Bypass System with a greater bypass capacity with respect to total steam flow. The capacity of the Turbine Bypass System, through the bypass to the condenser, includes 4 new bypass valves rated at 735,000 pounds per hour each (at 900 psia and 586°F). Refer to Appendix E, Major Plant Modifications, for further information related to the modification of the turbine bypass valves and refer to Section 2.5.5.3, Steam Dump System, for additional discussion of the Turbine Bypass System.

Description of Analyses and Evaluations

The main condenser will experience higher steam flows due to the increase in low pressure turbine exhaust flow at the EPU power level during normal power operation. The main condenser will also experience higher steam demands from the Turbine Bypass System. The following evaluations determined the impact of the EPU conditions on the performance and integrity of the condenser:

- Evaluate the steam blowdown effects of increased steam flow at the proposed EPU power operation and during steam dump to the condenser following load rejection on condenser tube vibration,
- Evaluate the impact of the increase steam flow on the condenser spargers, baffles, and impingement plates, provided to protect the condenser tube and internal components from damage due to incoming steam and water flows,
- Evaluate the impact on the condensation / deaeration capability to maintain condenser back press and condensate oxygen content within operating limits, and
- Evaluate the impact of the increased steam flow on the plant design to control the release of radioactive effluents in accordance with FSAR Section 1.4.70.

Results

The evaluation determined that the condenser satisfactorily removes the increased EPU heat loads, condenses the required steam flows.

Table 2.5.5.2-1 describes the key design parameters of the main condenser for each condenser shell and compares its performance at current operating and the proposed EPU conditions.

A main condenser tube vibration evaluation using Heat Exchanger Institute (HEI) methodology determined that the existing tube span lengths for the spacing associated with intermediate support plates and for the end plate spans is adequate for the proposed EPU operating conditions. To meet the minimum spacing requirements stakes are installed throughout the tubes except for the inner bundles below the rain trays and the inner bundles above the rain trays less the first 10 rows. This spacing provided the staked tubes with an unsupported span of 42.375 inches. No modifications are required for the existing condenser tube supports for the EPU operation. The vibration analysis demonstrated that the predicted steam cross flow velocities in the un-staked areas of the tube bundles were satisfactory.

The increased steam flow rates at the EPU conditions of normal operation and steam dump may increase the wear of condenser internal spargers, baffles, and impingement plates. Modifications were previously implemented to ensure that the spargers, baffles, and impingement plates are adequate for the EPU operation. The diffusers in the condenser were replaced with a larger and more robust model that can sustain the increased steam flow in preparation for the EPU. These components are inspected every refueling outage for unacceptable wear.

CR-3 analyzed the condenser performance at various Circulating Water flow rates, different condenser configurations, and over a range of Circulating Water inlet temperatures. Analyses were performed to determine the allowable operating range of unit power and circulating water temperature with a circulating water pump out of service to establish operational limits. The analyses were based on the EPU design heat load, a condenser cleanliness factor of 0.9 and circulating water temperatures which varied from 50°F to 95°F. Each analysis yielded condenser inlet pressure values as a function of circulating water temperature based on the input variables for that analysis. The analyses determined that the backpressures over the range of operating circulating water temperatures remained below the 4.5 inHg A limit for the turbine "Do Not Operate" region at low power, and also, well below the turbine trip set point of 9.0 inHg A when operating at full power.

The design of the main condenser does not change following the implementation of the EPU. Therefore, the EPU does not impact the ability of the CR-3 regarding the control of radioactive material in accordance with FSAR Section 1.4.70. Monitoring of the air and non-condensibles leaving the condenser is accomplished by a radiation monitor in the Condenser Evacuation System, described in FSAR Section 11.4. The impact of the EPU on radiological effluent releases from the CR-3, radiation monitoring setpoints and compliance with 10 CFR Part 50, Appendix I, are discussed in Section 2.10.1, Occupational and Public Radiation Doses.

2.5.5.2.3 Conclusion

CR-3 has reviewed the assessment of the effects of the proposed EPU on the Main Condenser System and concludes the assessment adequately accounted for the effects of changes in plant conditions on the design of the main condenser. CR-3 also concludes that the Main Condenser System will continue to maintain its ability to withstand the blowdown effects of the steam from the Turbine Bypass System (Steam Dump) at the EPU conditions and thereby, continue to meet the CR- 3 current licensing basis with respect to the requirements of FSAR Section 1.4.70, for prevention of the consequences of failures in the system. Therefore, the CR-3 finds the proposed EPU is acceptable with respect to the main condenser.

2.5.5.2.4 References

None

Table 2.5.5.2-1 Main Condenser Performance Characteristics

	Current Operating Value 100% Power	EPU Operating Value 100% Power @ 75°F Circulating Water	EPU Operating Value 100% Power @ 95°F Circulating Water
Condenser Duty / Shell [Btu/hr]	5.819×10^9	7.10×10^9	7.08×10^9
Circulating Water Inlet Temperature [°F]	75	75	95
Circulating Water Temperature Rise [°F]	17.2	19.9	20.1
Condenser Backpressure [inHg A]	2.67	2.7	4.41

2.5.5.3 Steam Dump System

2.5.5.3.1 Regulatory Evaluation

The Steam Dump System consists of the Turbine Bypass valves (TBVs) and the Atmospheric Dump Valves (ADVs). The Turbine Bypass System controls steam pressure by regulating the steam flow that bypasses the main turbine and flows directly into the main condenser.

The Atmospheric Dump Valves also control the steam pressure by regulating the steam flow directly to the atmosphere. The TBVs, ADVs along with the MSSVs are used to dissipate the heat energy in the Reactor Coolant System (RCS) upon a turbine trip or loss of electrical load. The steam dump system is also used during plant startup and plant cooldown in controlling steam generator pressure which thereby controls the heat transfer rate of the steam generators.

The CR-3 review is focused on the effects that the EPU has on load rejection capability, analysis of postulated steam piping failures, and on the consequences of inadvertent Steam Dump System operation.

The NRC's acceptance criteria for the Steam Dump System are based on the following:

- GDC-4, insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures; and

- GDC-34, insofar as it requires that an RHR be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits (SAFDLs) and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4 the design criteria used during the licensing of CR-3 predates the general design criteria (GDC) provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria:

- FSAR Section 1.4.23, Protection Against Multiple Disability for Protection Systems; and FSAR Section 1.4.40, Missile Protection; insofar as it requires that SSCs important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures [GDC-4]; and

- FSAR Section 1.4.37, Engineered Safety Features Basis For Design and FSAR 1.4.42, Engineered Safety Features Components Capability, insofar as they require that an decay heat removal System be provided to transfer fission product decay heat and other residual heat from the reactor core at a rate such that SAFDLs and the design conditions of the RCPB are not exceeded [GDC-34].

2.5.5.3.2 Technical Evaluation

Introduction

The Steam Dump System is described in the FSAR Section 10.2.1.4. This system consists of the Turbine Bypass Valves (TBVs), Atmospheric Dump Valves (ADVs), and Main Steam Safety Valves (MSSVs). The purpose of this system is to remove energy from the Main Steam System:

- when the turbine has tripped or during large plant load reduction transients,
- when steam generation exceeds the steam demand; and
- when the turbine is shutdown and steam is used to remove the heat generated from the RCS by transferring its heat energy to the secondary side of the steam generator

Other evaluations of the Steam Dump System can be found in the following sections:

- Steam Generator Tube Rupture – Section 2.8.5.6.2
- Small Break Loss of Coolant Accident – Section 2.8.5.6.3
- Appendix R Cooldown – Section 2.5.1.4
- Station Blackout – Section 2.3.5

At the EPU conditions, the total steam flow increases as shown in Table 1.1-1. The Steam Dump System modifications to support these conditions included:

- The TBVs were replaced with higher capacity valves during the Fall 2009 outage (Refuel 16); and
- The ADVs will be replaced as safety-related components in the subsequent outage prior to the EPU. These modifications are further described in Appendix E. Major Modifications. The pre-EPU and post-EPU flow rates can be found in Table 2.5.5.3-1.

Description of Analyses and Evaluations

The Steam Dump System and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. This evaluation addressed the following functions of the Steam Dump System:

- Ability to handle a load rejection, as demonstrated by bypass capacity
- Impact of postulated steam piping failure, as demonstrated in Section 2.2.1
- Impact of inadvertent operation, as bounded by the small steam line break described in Section 2.8.5.1.1

- Impact of dynamic effects, specifically missile protection and steam hammer
- Ability to remove residual heat, as demonstrated by Appendix R Cooldown

Results

Although steam flow will increase, the larger ADVs and TBVs will give the Steam Dump System a greater bypass capacity with respect to total steam flow. This will improve the system's ability to accept a load rejection and decrease the reliance on the MSSVs.

The impact acceptability of postulated steam piping failures is addressed and found to be adequate in Section 2.2.1, Pipe Rupture Locations and other Dynamic Effects.

The fuel impacts due to Inadvertent operation of the ADVs is bounded by the small steam line break (increase in steam flow) evaluated in Section 2.8.5.1.1, Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve. The cooldown impact of ADV opening due to Fast Cooldown System actuation was evaluated and shown to be within analyzed conditions or limits. In addition the TBVs and ADVs are designed to fail closed on loss of control air or control signal. In the event a TBV or ADV controller malfunction occurs, preventing valve opening when demanded, the MSSVs function to control Main Steam System pressure within design limits.

Dynamic effects from missiles are addressed in Section 2.5.1.2.1, Missile Protection and found to be adequate for existing systems or SSCs. Dynamic effects from steam hammer are addressed in Section 2.2.2.2, BOP Piping and Supports.

The use of the steam dump system for decay heat removal, with the new higher capacity ADVs, was found to be adequate as addressed in Section 2.5.1.4, Fire Protection.

2.5.5.3.3 Conclusion

CR-3 reviewed the assessment of the effects of the proposed EPU on the Steam Dump System and concludes that the assessment adequately accounted for the effects of changes in plant conditions on the design of the Steam Dump System, with the modifications as described. CR-3 concludes that the steam dump system will continue to provide a means for shutting down the plant during normal and emergency operations. CR-3 further concludes that the steam dump system will not adversely affect essential systems or components. Based on this, CR-3 concludes that the Steam Dump System will continue to meet FSAR Sections 1.4.23, 1.4.37, 1.4.40, and 1.4.42. Therefore, CR-3 finds the proposed EPU is acceptable with respect to the Steam Dump System.

2.5.5.3.4 References

None

Table 2.5.5.3-1: Steam Dump System Bypass Flow Capacity

Component	Flow Capacity Pre-EPU 2609 MWt [lb_m/hr]	Flow Capacity Post-EPU 3014 MWt [lb_m/hr]
TBVs (4 valves)	1,672,000 (418,000 each valve @ 910 psia and 600°F)	2,940,000 (735,000 each valve @ 900 psia and 586°F)
ADVs (2 valves)	602,592 (301,296 each valve @ 962.7 psia and 540°F)	1,240,000 (620,000 each valve @ 962.7 psia and 540°F)

2.5.5.4 Condensate and Feedwater

2.5.5.4.1 Regulatory Evaluation

The Condensate and Feedwater Systems provide feedwater at the appropriate temperature, pressure, and flow rate to the steam generators. Portions of the Feedwater System are classified as safety-related including the piping from the main feedwater block valves to the steam generators and the main feed pump suction isolation valves. The CR-3 review focused on the effects of the proposed EPU on previous analyses and considerations with respect to the capabilities of the Condensate and Feedwater Systems to supply adequate feedwater during plant operation and shutdown, and to isolate components and piping in order to preserve the Feedwater System's safety function. The CR-3 review also considered the effects of the proposed EPU on the Feedwater System, with regard to possible fluid flow instabilities during normal plant operation, as well as during upset or accident conditions.

The NRC's acceptance criteria for the Condensate and Feedwater Systems are based on the following:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects;
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and
- GDC-44, insofar as it requires that a system with the capability to transfer heat loads from safety-related SSCs to a heat sink under both normal operating and accident conditions be provided, and that suitable isolation be provided to assure that the system safety function can be accomplished assuming a single failure.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.4, Sharing of Systems; insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5]; and
- FSAR Section 1.4.23, Protection Against Multiple Disability For Protection Systems, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects [GDC-4].

Additionally, FSAR Sections 10.2.1.1 and 10.2.1.2 provide requirements for the capability of condensate and feedwater to transfer heat loads to a heat sink under normal operating conditions, and suitable isolation to assure that the safety functions can be accomplished assuming a single failure [GDC-44].

2.5.5.4.2 Technical Evaluation

Introduction

The Condensate and Feedwater Systems are described in FSAR Sections 10.2.1.1 and 10.2.1.2, respectively. As currently configured, the Condensate and Feedwater Systems function to collect steam condensed in the main condenser, exhausted from the low pressure turbines, and heat this condensate, then send it back to the steam generators at the temperature and pressure required for heat removal from the Reactor Coolant System. Safety-related components and piping within the Feedwater System are used for feedwater isolation during accidents and transients as well as being the main feedwater flow paths to each steam generator during normal operation.

Description of Analyses and Evaluations

The Condensate and Feedwater Systems and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. The evaluation considered the effects of the EPU on the following system / component design aspects:

- Design pressures / temperatures of piping, valves and components versus the EPU operating pressures / temperatures
- Flow velocities
- Feedwater isolation valve closure within the required time period at the EPU hydraulic conditions of flow and pressure drop.
- Capacity and control capability of the feedwater control valves at low loads
- Feedwater heaters design parameters and operating characteristics listed below

  a. Thermal performance

  b. Shell side and tube side velocities

  c. Steam and water nozzle velocities

  d. Shell and tube side pressure drops

  e. Shell and tube side relief valve capacities and setpoints

  f. Shell side venting capacity

  g. Steam impingement and tube vibration

  h. Shell side and tube side design pressure / temperature

- Pump and pump supporting subsystem design capabilities, including Net Positive Suction Head (NPSH), flow, head, brake horsepower, and minimum flow protection
- Process setpoints for protective functions

The Condensate and Feedwater systems were evaluated by utilizing a hydraulic model (Fathom) of the system components and piping and the EPU heat balance (PEPSE). Physical plant data for the installed components and piping were utilized in the hydraulic model. The physical changes to condensate and feedwater major components, valves and piping which resulted from the EPU evaluations are described in Appendix E, Major Plant Modifications, incorporated into the hydraulic model and verified as acceptable.

The pre-EPU plant operating data were gathered and included in the operating heat balances to reflect the performance of the existing components. The operating heat balances were then scaled to the EPU operating conditions and issued as the EPU heat balances. The EPU heat balances were used to establish the flow, temperatures and heat transfer requirements at the EPU power level.

Other evaluations of Condensate and Feedwater systems and components are addressed in the following Sections:

- Effects of increased flow and velocity on erosion / corrosion concerns - Section 2.1.8, Flow Accelerated Corrosion
- Piping / component supports and water hammer effects - Section 2.2.2.2, Balance of Plant Piping, Components and Supports (Non-Class 1)
- Protection against dynamic effects, missiles, pipe whip and discharging fluids - Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects
- Feedwater isolation valve testing and valve closure requirements - Section 2.2.4, Safety Related Valves and Pumps
- Operation of the Feedwater System, including isolation features during postulated abnormal and accident scenarios - Section 2.8.5, Accident and Transient Analyses
- Protection against turbine missiles and internal missiles - Section 2.5.1.2, Missile Protection
- Emergency Feedwater System - Section 2.5.4.5, Emergency Feedwater System
- Pipe Failures, for discussion of plant design for protection from piping failures outside containment - Section 2.5.1.3, Protection Against Postulated Piping Failures in Fluid Systems Outside Containment.

Results

The evaluation of the Condensate and Feedwater Systems capabilities at the EPU conditions demonstrates that CR-3 will continue to meet the current licensing basis with respect to the requirements of the design criteria described in CR-3 FSAR Sections 10.2.1.1, and 10.2.1.2. The feedwater system provides an essential isolation function of feedwater flow to the steam generators. The Feedwater

System provides this capability during accident conditions and is capable of achieving this function assuming a single failure. The implementation of the EPU does not affect the capability of these systems to perform these functions as demonstrated by the system and component evaluation results described below and by the results of the analyses of postulated abnormal operating conditions.

Accident analysis for the EPU Main Steam Line Break (MSLB) requires a change to the feedwater isolation valve (FWV-28, 29, 30) closure time requirement from the current 34 seconds to 31 seconds. No physical changes to the valve(s) are required since the change simply uses margin available in the surveillance testing acceptance criteria.

To support the EPU, the condensate pumps will be upgraded with a higher capacity impeller and larger motors. The booster feedwater pumps will also be replaced with higher capacity pump and motor combinations that will provide increased volume and net positive suction head (NPSH) to the existing turbine driven main feedwater pumps.

Feedwater isolation valves (FWV-14 and 15) will be replaced with new valves capable of closing within 20 seconds consistent with assumptions in safety analyses. (See Section 2.2.4 Safety-Related Valves and Pumps)

System Operating Conditions - Current versus the EPU Conditions

The Condensate and Feedwater System operating conditions, flow, temperature and pressure were determined from hydraulic modeling (Fathom) of the piping systems and from the current operating (benchmark) and the EPU heat balance (PEPSE).

Design Pressures / Temperatures - Components and Piping

The design pressures and temperatures of condensate and feedwater components and piping, as modified (refer to Appendix E), bound the EPU operating conditions.

Feedwater Heaters

Condensate system heaters CDHE-1A/B, 2A/B were evaluated as acceptable for the EPU operation based on their current design, materials, construction, and performance. Heaters CDHE-3A/B were determined to be undersized and were replaced with larger heat exchangers to support the EPU conditions.

Feedwater heaters FWHE-3A/B will be replaced to address excessive fluid velocities under EPU conditions.. The Intermediate Feedwater Heater (FWHE-2 A/B) will be replaced to accept the higher discharge pressure of the replacement Booster Feedwater pumps.

Current plant operating and inspection data and the predicted EPU heat balance conditions have been reviewed to reach these conclusions.

The feedwater heaters shell and tube side relief valves were evaluated. The CDHE-3A/B replacement heaters were supplied with new shell and tube side relief valves rated for the new design conditions. The FWHE-2A/B and FWHE-3A/B replacement heaters will be supplied with new shell and tube side relief valves rated for the new design conditions.

The feedwater heater shell side vents are acceptable for the EPU operation. The only change required for the feedwater heater shell air removal vent due to the EPU is to increase the vent valve setting for all feedwater heaters to provide sufficient vent flow.

Flow Velocities Piping

Flow velocities through the Condensate and Feedwater System were calculated at the current and EPU conditions. Velocities generally remain below the industry standard guidelines for these services although there are some pipes whose velocities exceed the guidelines. These individual pipes are evaluated as part of the erosion / corrosion program as described in Section 2.1.8, Flow Accelerated Corrosion.

Feedwater Low-Load Control Valves

The existing feedwater low-load control valves are adequate to provide the required flow at the required pressure drop up to 3 million lb/hr of main feedwater flow per line or 6 million lb/hr total. With main feedwater demand below 3 million lb/hr per line, the main feedwater pumps are controlled to maintain 80 psi across the flow control valves. The 80 psi setting will not be changed for EPU.. As flow is increased above 3 million lb/hr feedwater demand, the main block valves are automatically opened and the main feedwater flow is controlled by the main feedwater pump speed. In addition, once the main block valves go open, the start-up and low-load control valves lock in at their existing positions. These valves lock in and remain in the position required to resume flow control at the correct differential pressure when, later in the cycle, main feedwater flow demand is reduced below 2.7 million lb/hr and the main block valves automatically close.

After EPU, the transition to main feedwater pump speed control will occur at the same feedwater flow rate of 3 million lb/hr, but this will be a lower percentage of full power. The transition point will correspond to approximately 46% full main feedwater flow.

Condensate and Feedwater Pumps and Supporting Subsystems

The condensate pumps, booster feedwater pumps, main feedwater pumps and their supporting subsystems will continue to operate successfully during the EPU conditions based on the evaluation results, modifications and the post-EPU inspections described below:

- The condensate pumps will be modified to provide the necessary EPU flows at the required total dynamic head. The modifications include a new impeller and new constant speed motors. Modifications also include new condensate flow control valves in the condensate pump discharge piping, which are replacing the variable speed pump control. The condensate pump recirculation subsystem is also being modified to provide sufficient flow to meet the pump minimum flow requirements at the EPU conditions. In addition, condensate reject flow design will be improved to facilitate startup operations at low condensate flow rates.
- The replacement booster feedwater pumps will provide the required discharge pressure and flow to the main Feedwater pump suction at the EPU with adequate NPSH margin.
- The existing main feedwater pump turbine drivers are adequate at the EPU conditions, however the pumps will be replaced to provide increased flow.

Feedwater Isolation Valves

Feedwater isolation is required for a variety of postulated transients and accident events. The current plant design provides for feedwater isolation using the main feedwater isolation valves, associated bypass valves, and the feedwater pump suction isolation valves. In order mitigate a design basis steam line break in containment at the EPU conditions, faster isolation of feedwater addition to the faulted steam generators needs to occur so as to minimize the mass and energy released to containment.

Note that Improved Technical Specifications (ITS) 3.7.3, Main Feedwater Isolation Valves, Associated Bypass Valves, and Main Feedwater Pump Suction Valves, provide automatic feedwater isolation requirements. Containment isolation is accomplished by check valves on the feedwater headers outside containment and by normally closed manual valves on branch lines from the headers penetrating containment. The containment isolation requirements are unaffected by the EPU and the current plant design features remain acceptable.

2.5.5.4.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the Condensate and Feedwater System and concludes that its assessment has adequately accounted for the effects of changes in plant conditions on the design of the Condensate and Feedwater Systems. CR-3 concludes that the Condensate and Feedwater Systems, with the implementation of the modifications summarized in Appendix E, Major Plant Modifications, will continue to maintain their ability to satisfy feedwater requirements for normal operation and shutdown, maintain isolation capability in order to preserve the system's safety function, and not degrade safety-related structures, systems, and components. CR-3 further concludes that the Condensate and Feedwater Systems will continue to meet the requirements of FSAR Sections 1.4.4, 1.4.23, 10.2.1.1, and 10.2.1.2. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Condensate and Feedwater Systems.

2.5.5.4.4 References

None

2.5.6 Waste Management Systems

2.5.6.1 Gaseous Waste Management System

2.5.6.1.1 Regulatory Evaluation

Gaseous Waste Management Systems involve the Gaseous Radwaste System, which deals with the management of radioactive gases collected in the offgas system or the waste gas storage and decay tanks. In addition, it involves the management of the Condenser Air Removal System, the steam generator blowdown flash tank, and the containment purge exhausts; the Building Ventilation System exhausts. The CR-3 review of the Gas Waste Disposal System (GWDS) focused on the effects that the proposed CR-3 EPU may have on (1) the design criteria of the Gaseous Waste Management Systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exist.

The NRC’s acceptance criteria for the Gas Waste Disposal System are based on:

- 10 CFR Part 20.1302 insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values;
- GDC-3, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat-resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety;
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents;
- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and
- 10 CFR Part 50, Appendix I Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" (ALARA) criterion.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable design, construction, and operation of CR-3.

The following are applicable criteria for the GWDS:

- FSAR Section 1.4.3, Fire Protection, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat-resistant materials be used, and (c) fire detection and fighting systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety, [GDC-3];

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include a means to control the release of radioactive effluents, [GDC-60]; and

- FSAR Section 1.4.69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement, [GDC-61].

Additionally, FSAR Section 11.2.4, provides criteria insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values in conformance with 10 CFR 20.1302. Likewise, FSAR Section 11.1.1 and 11B provide criteria and sets numerical guides for design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" (ALARA) criterion in conformance with 10 CFR 50, Appendix I.

2.5.6.1.2 Technical Evaluation

The GWDS provides for the safe collection and storage of gases evolved from primary coolant in tanks and equipment. The GWDS is designed to collect, process, store, and provide for a controlled release of potentially radioactive gases to the environment within 10 CFR Part 50, Appendix I and 10 CFR 20.1302 criteria.

The GWDS and design functions are described in FSAR Section 11.2.2, Radioactive Gas Waste Disposal System. The GWDS and components were evaluated to ensure they are capable of performing their intended functions at the EPU conditions. The evaluation compared the existing design parameters of the systems / components with the EPU conditions.

<u>Description of Analyses and Evaluations</u>

The CR-3 GWDS flow rates, gaseous inventory, and process conditions are not changed by the EPU and are within the original design parameters of the system. There are no modifications or additions to system components as the result of the EPU that would introduce any new functions or change the functions of existing components.

The evaluations are based on the use of the PWR GALE code (as described in NUREG-0017) at the EPU levels (Reference 1). The gaseous waste source term for the reference plant is contained within PWR GALE. NUREG-0017 provides guidance to adjust the source term based on site specific data. The normal operations source term for the primary and secondary coolant is provided in Section 2.9.1, Source Terms for Radwaste Systems Analyses. Section 2.10.1, Occupational and Public Radiation Doses, addresses the dose impact from the higher gaseous waste activity concentrations.

Results

The implementation of the EPU does not significantly increase the inventory of gas normally processed by the GWDS since the plant system functions are not changing and the assumptions related to volume inputs remain the same. The proposed EPU does not add or change any of the sources of potentially explosive mixtures. The activity and explosive mixture are controlled by CR-3 procedures.

The proposed EPU results in an increase in the equilibrium radioactivity in the reactor coolant. This change in radioactivity of the reactor coolant impacts the concentrations of radioactive nuclides in the Waste Disposal Systems, as evaluated with PWR GALE code. The radiological impact of the increased activity in the GWDS is detailed in Section 2.10.1, Occupational and Public Radiation Doses**.** The activity level of the GWDS inputs and subsequent environmental releases proportionately increase; however, the GWDS design has adequate holdup and processing capacity to ensure design criteria are met.

Potentially radioactive gas is collected from selected systems and components and is directed to the GWDS. Gases resulting from process operations, gases used for tank cover gas, gases collected during venting, and gases generated in the radio-chemistry laboratory enter the waste gas decay tanks during all plant operating modes. The implementation of the EPU does not add any new sources of potentially contaminated gases, nor does it create any new flow paths or routes that would allow the contamination of uncontaminated gases.

The evaluation of the GWDS at the EPU conditions shows concurrence with 10 CFR 20.1302, insofar as the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This is demonstrated by the continued compliance post-EPU with the annual dose objective of 10 CFR 50 Appendix I as discussed in Section 2.10.1, Occupational and Public Radiation Doses. Discharge streams remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases, in accordance with the Offsite Dose Calculation Manual.

The evaluation of the GWDS at the EPU conditions demonstrates that CR-3 will continue to meet the current licensing basis with respect to the requirements of FSAR Design Criterion 1.4.3 insofar as it requires that the plant design includes fire detection and fighting systems of appropriate capacity and capability for the protection of structures, systems and components important to safety. There is no impact to the fire detection and fighting systems due to the EPU. There are no new gaseous waste components added as a result of the EPU and the gaseous waste flow rates, gaseous inventory, and process conditions are not changed by the EPU. Thus the existing systems retain their compliance to FSAR Design Criterion 1.4.3.

The evaluation of the GWDS at the EPU conditions demonstrates that CR-3 will continue to meet the current licensing basis with respect to the requirements of FSAR Design Criterion 1.4.70, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I, and are described in the Offsite Dose Calculation Manual. Since the design objectives of Appendix I have been demonstrated at pre-EPU conditions, the increased radioactive source term due to the EPU operation is minimal, and no changes to GWDS design or operation result from EPU, the GWDS will continue to meet the current licensing basis with respect to 10

CFR 50, Appendix I. As presented in Section 2.10.1, Occupational and Public Radiation Doses, the maximum dose due to gaseous effluents following the EPU is below the 10 CFR 50, Appendix I limits.

The evaluation of the GWDS at the EPU conditions demonstrates that CR-3 will continue to meet the current licensing basis with respect to the requirements of FSAR Design Criterion 1.4.69, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the EPU.

Since conformance to the design objectives of 10 CFR Part 50, Appendix I has previously been demonstrated, the increased source term due to the proposed EPU operation is minimal, and no changes to GWDS design or operation result from the EPU, the GWDS will continue to meet the current licensing basis with respect to 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D. These criteria set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" criterion as defined in the technical specifications requirements for the radioactive effluent controls program and the Offsite Dose Calculation Manual. As presented in Section 2.10.1, Occupational and Public Radiation Doses, the maximum dose due to gaseous effluents following EPU is below the 10 CFR 50, Appendix I limits.

2.5.6.1.4 Conclusion

CR-3 has reviewed the assessment related to the GWDS. CR-3 concludes that the assessment has adequately accounted for the effects of the increase in fission product and amount of gaseous waste on the ability of the GWDS to control releases of radioactive materials and preclude the possibility of an explosion if the potential for explosive mixtures exists. CR-3 finds that the GWDS will continue to meet its design functions following implementation of the proposed EPU. CR-3 further concludes that the GWDS will continue to meet the requirements of 10 CFR 20.1302, FSAR Sections 1.4.3, 1.4.69, and 1.4.70, 11.1.1, 11.2.4, and 11B, and 10 CFR 50, Appendix I, Sections II.B, II.C, and II.D. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Gas Waste Disposal System.

2.5.6.1.5 References

1. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR GALE code),” Revision 1, April, 1985.

2.5.6.2 Liquid Waste Management Systems

2.5.6.2.1 Regulatory Evaluation

The CR-3 review of the Liquid Waste Management System (LWDS) focused on the effects that the proposed EPU may have on the previous analyses and considerations related to the LWDS design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents.

The NRC's acceptance criteria for the Liquid Waste Management System are based on:

- 10 CFR Part 20.1302 insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values;
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents;
- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement; and
- 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" (ALARA) criterion.
- 10 CFR Part 50, Appendix I Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" (ALARA) criterion.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are applicable criteria for the LWDS:

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include means to control the release of radioactive effluents [GDC-60];
- FSAR Section 1.4.69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement [GDC-61];

Additionally FSAR Section 11.2.1.3 and FSAR Section 11.2.4 provide criteria insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values in conformance with 10 CFR 20.1302. Likewise, FSAR Section 11.1.1 and 11B provide criteria and sets numerical guides for design objectives and limiting conditions for operation to meet the "as-low-as-is-reasonably-achievable" (ALARA) criterion in conformance with 10 CFR 50, Appendix I.

2.5.6.2.2 Technical Evaluation

The LWDS provides for the safe collection, processing, and storage of potentially radioactive liquids. The LWDS is designed to collect, process, store, and provide for a controlled release of potentially radioactive liquids to the environment within 10 CFR 50, Appendix I and 10 CFR 20.1302 criteria.

The LWDS and design functions are described in FSAR Section 11.2.1, Radioactive Liquid Waste Disposal System. The LWDS and components were evaluated to ensure they are capable of performing their intended functions at the proposed EPU conditions. The evaluation compared the existing design parameters of the systems / components with the EPU conditions.

Description of Analyses and Evaluations

The CR-3 LWDS flow rates, water inventory, and process conditions are not changed by the EPU and are within the original design parameters of the system. There are no modifications or additions to LWDS components as the result of the EPU that would introduce any new functions or change the functions of existing components.

The evaluations are based on the use of the PWR GALE code (as described in NUREG-0017) at the EPU levels (Reference 1). The liquid waste source term for the reference plant is contained within the PWR GALE code. NUREG-0017 provides guidance to adjust the source term based on site specific data. The normal operations source term for the primary and secondary coolant is provided in Section 2.9.1, Source Terms for Radwaste Systems Analyses. Section 2.10.1, Occupational and Public Radiation Doses, addresses the dose impact from liquid waste at the EPU conditions.

Results

The implementation of the EPU does not significantly increase the inventory of liquid normally processed by the LWDS since the system functions are not changing and the assumptions related to volume inputs remain the same.

The EPU results in an increase in the equilibrium radioactivity in the reactor coolant. This change in radioactivity of the reactor coolant impacts the concentrations of radioactive nuclides in the LWDS, as evaluated with the PWR GALE code. The radiological impact of the activity in the Waste Disposal Systems is detailed in Section 2.10.1, Occupational and Public Radiation Doses. The activity level of the LWDS inputs and subsequent environmental releases proportionately increase; however, the LWDS design has adequate processing capacity to ensure design criteria are met.

Potentially radioactive drainage is collected in tanks and drain sumps from selected systems and components and is directed to the LWDS. Liquids leaking from process systems, liquids used during cleaning activities, liquid spills from maintenance activities, and liquids generated in the radio-chemistry laboratory enter the Equipment and Floor Drain System during all plant operating modes. The implementation of the EPU does not add any new sources of potentially contaminated leakage, nor does it create any new flow paths or routes that would allow the contamination of drainage systems designed for uncontaminated fluids.

The evaluation of the LWDS at the EPU conditions shows conformance with 10 CFR Part 20.1302, insofar as the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This is demonstrated by the continued compliance post-EPU with the annual dose objective of 10 CFR 50, Appendix I as discussed in Section 2.10.1, Occupational and Public Radiation Doses. Discharge streams remain appropriately monitored and adequate safety features remain incorporated to preclude excessive releases, in accordance with the Offsite Dose Calculation Manual.

The evaluation of the LWDS at the EPU conditions demonstrates that CR-3 will continue to meet the current licensing basis with respect to the requirements of FSAR Section 1.4.70, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU.

The evaluation of the LWDS at the EPU conditions demonstrates that CR-3 continues to meet the current licensing basis with respect to the requirements of FSAR Design Criterion 1.4.69, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement to ensure adequate safety under normal and postulated accident conditions. This design capability remains unchanged by the EPU.

The handling, control, and release of radioactive materials are in compliance with 10 CFR 50, Appendix I, and are described in the Offsite Dose Calculation Manual. Since the design objectives of Appendix I have been demonstrated at pre-EPU conditions, the increased radioactive source term due to the EPU operation is minimal, and no changes to LWDS design or operation result from the EPU, the LWDS continues to meet the current licensing basis with respect to 10 CFR 50, Appendix I. The evaluation of the LWDS at the EPU conditions demonstrates conformance with the requirements of 10 CFR 50, Appendix I, Sections II.A and II.D. These criteria set numerical guides for dose design objectives and limiting conditions for operation to meet the ALARA criterion as defined in the technical specifications requirements for the radioactive effluent controls program and the Offsite Dose Calculation Manual. As presented in Section 2.10.1, Occupational and Public Radiation Doses, the maximum dose due to liquid effluents following the EPU is below the 10 CFR 50, Appendix I limits.

2.5.6.2.3 Conclusion

CR-3 has reviewed the assessment related to the LWDS. CR-3 concludes that the assessment has adequately accounted for the effects of the increase in fission product and amount of liquid waste on the ability of the LWDS to control releases of radioactive materials. CR-3 finds that the LWDS will continue to meet its design functions following implementation of the proposed EPU. CR-3 further concludes that the assessment has demonstrated that the LWDS will continue to meet the requirements in FSAR Sections 1.4.69, 1.4.70, and 11.1.1, 11.2.1.3, 11.2.4, and 11B. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Liquid Waste Management System.

2.5.6.2.4 References

1. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR GALE code),” Revision 1, April 1985.

2.5.6.3 Solid Waste Management Systems

2.5.6.3.1 Regulatory Evaluation

The CR-3 review of the Radioactive Solid Waste Packaging System (RSWPS) focused on the effects that the proposed CR-3 EPU on the previous analyses and considerations related to the design objectives in terms of expected volumes of waste to be processed and handled, the wet and dry types of waste to be processed, the activity and expected radionuclide distribution contained in the waste, equipment design capacities, and the principal parameters employed in the design of the RSWPS.

The NRC's acceptance criteria for the Radioactive Solid Waste Packaging System are based on:

- 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values,
- GDC-60, insofar as it requires that the plant design include a means to control the release of radioactive effluents,
- GDC-63, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels,
- GDC-64, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents,
- 10 CFR 71, which states requirements for radioactive material packaging.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable design, construction, and operation of CR-3.

The following are applicable criteria:

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include a means to control the release of radioactive effluents [GDC-60],
- FSAR Section 1.4.18, Monitoring Fuel and Waste Storage, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels [GDC-63], and
- FSAR Section 1.4.17, Monitoring Radioactivity Release, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity

that may be released from normal operations, including anticipated operational occurrences, and postulated accidents. [GDC-64].

Additionally FSAR Section 11.1.1 and 11.2.4 provide criteria insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values in conformance with 10 CFR 20.1302. Likewise, FSAR Section 11.2.5 provides criteria insofar as it states requirements for radioactive material packaging in conformance with 10 CFR 71.

2.5.6.3.2 Technical Evaluation

Introduction

The RSWPS design functions are described in FSAR Section 11.2.5, Radioactive Solid Waste Packaging System. The routine types of waste that are generated at CR-3 are dry active waste (DAW), aqueous filter media (resin, filters, sludge), and contaminated oil. The RSWPS is designed to safely package, store and transport radioactive waste, while minimizing radiation exposure to personnel. Wastes are packaged for storage, shipment to offsite waste processors, or shipment to burial facilities. Waste can also be returned to CR-3 from offsite waste processors for long-term storage. Solid waste packaging and transportation is performed in accordance with both Department of Transportation (DOT) and NRC regulations. Waste containers are surveyed for radiological conditions and stored in designated storage areas. Storage locations include outside storage areas, the D Radioactive Material Storage Warehouse, modular hazardous material storage buildings, and the replaced reactor head containment building discussed in FSAR Section 11.2.5.1, Process Evaluation.

Solid radioactive wastes include solids recovered from the Reactor Coolant Systems (to include aqueous filter media), solids in contact with the reactor process system liquids or gases (to include DAW, sludge, and contaminated oil), and solids used in the Reactor Coolant System operation. Table 2.5.6.3-1 presents the historical annual volume and activity of solid low-level radioactive waste generated at CR-3 for years 2004 through 2008 (Reference 1). The operational data summarized in Table 2.5.6.3-1 reflect the estimated aqueous filter media data recorded in FSAR Table 11-4, Radioactive Waste Quantities. Fluctuations in annual DAW generation are based on outage versus non-outage years. Solid radioactive waste volumes are not expected to significantly change due to operating at the EPU conditions.

The EPU would result in a small increase in the equilibrium radioactivity in the reactor coolant which in turn would impact the concentrations of radioactive nuclides in the RSWPS. Section 2.5.6.2, Liquid Waste Management Systems, addresses the increase in the concentration of activity in the liquid wastes streams based on the increase in activity in the Reactor Coolant System due to the higher uranium enrichment in the fuel. The evaluations are based on the use of the PWR GALE code (as described in NUREG-0017) at the EPU levels (Reference 2). Section 2.10.1, Occupational and Public Radiation Doses address the dose impact from the slightly higher liquid waste concentrations.

The solid waste activity is based on the accumulation of source term of the process flows that are purified by demineralizer and filter packages. Activity accumulates on the resin and in the filter media. Therefore, a small increase in the equilibrium primary and secondary coolant concentrations results in a small increase in the solid waste activity. PWR GALE contains a primary and secondary coolant source term for a reference plant. These source terms were adjusted based on site specific parameters such as rated thermal power, demineralizer decontamination factors and process flow rates.

Description of Analyses and Evaluations

The RSWPS provides for the safe packaging, storage, and transport of radioactive waste while minimizing radiation exposure to personnel. Solid waste includes DAW, spent resins, tank and sump sludge, spent filters, and contaminated oil.

The evaluations are based on trending of historical data from 2003 to 2007 in order to project the volume of the solid waste for operation at the EPU conditions. The activity of the solid waste is projected based on increase primary and secondary coolant equilibrium radioactivity due to the EPU.

The source term for the reference plant is contained within PWR GALE. NUREG-0017 provides guidance to adjust the source term based on site specific data. The normal operations source term for the primary and secondary coolant is provided in Section 2.9.1, Source Terms for Radwaste Systems Analyses. The activity accumulated in the solid waste is dependent upon a small increase in the primary and secondary coolant source terms.

Results

The EPU has no significant effect on the generation of solid waste volume from the primary and secondary systems since the system functions are not changing and the assumptions related to volume inputs remain the same. The EPU will result in slight increases in the equilibrium radioactivity in the reactor coolant. This change in radioactivity of the reactor coolant impacts the concentrations of radioactive nuclides in the waste disposal systems as discussed in Section 2.5.6.2, Liquid Waste Management System. The impact of the activity in the Waste Disposal Systems at the EPU conditions is provided in Section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the RSWPS at the EPU conditions demonstrates compliance with 10 CFR Part 20.1302, since the annual average concentrations of radioactive materials released at the boundary of the unrestricted area will not exceed specified values. This is demonstrated by the continued compliance post-EPU with the annual dose objective of 10 CFR Part 50, Appendix I as discussed in Section 2.10.1, Occupational and Public Radiation Doses and FSAR Section 11B, Appendix I Analyses for CR-3. Discharge streams remain appropriately monitored as discussed in FSAR Section 11.2, Radioactive Waste Disposal System Summary, and FSAR Section 11.4.2.1.2, Atmospheric Monitoring System, and adequate safety features remain incorporated to preclude excessive releases in accordance with the site's Offsite Dose Calculation Manual (ODCM). No solid waste volumes are expected to leave the site except as properly packaged and shipped by an authorized carrier to a licensed burial site in accordance with the site's Process Control Program (PCP), NRC, DOT, and state regulations.

The evaluation of the RSWPS at the EPU conditions demonstrates that CR-3 continues to meet the current licensing basis with respect to the requirements of FSAR Design Criterion 1.4.70, insofar as it requires that the plant design include means to control the release of radioactive effluents. This design capability remains unchanged by the EPU. The processing, control, and release of radioactive materials are in compliance with 10 CFR Part 50, Appendix I, and are described in the CR-3 ODCM and PCP.

The evaluation of the RSWPS at the EPU conditions demonstrates that CR-3 continues to meet the current licensing basis with respect to the requirements of FSAR Design Criteria 1.4.18, 1.4.17, and 1.4.70, insofar as it requires that systems be provided in waste handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. This design capability remains unchanged by the EPU. Radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity. Refer to Section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the RSWPS at the EPU conditions demonstrates that CR-3 continues to meet the current licensing basis with respect to the requirements of CR-3 FSAR Design Criterion 1.4.17, insofar as it requires that a means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and postulated accidents. This design capability remains unchanged by the EPU. Radioactivity levels contained in the effluent discharge paths in the environs are continually monitored during normal and accident conditions by the station radiation monitoring system and by the radiation protection program for CR-3, as described in Section 2.10.1, Occupational and Public Radiation Doses.

The evaluation of the RSWPS at the EPU conditions demonstrates conformance with the requirements of 10 CFR Part 71, insofar as the radioactive material packaging accounts for the maximum dose rate allowed on the surface of the container by shielding of the package in which the container is shipped. Packaging, shielding and handling of radioactive material are not changed by the EPU; thus, compliance with 10 CFR Part 71 is not affected in accordance with the CR-3 PCP.

2.5.6.3.3 Conclusion

CR-3 has reviewed the assessment related to the RSWPS. CR-3 concludes that the assessment has adequately accounted for the effects of the increase in fission product and amount of solid waste on the ability of the RSWPS to process the waste. CR-3 finds that the RSWPS will continue to meet its design functions following implementation of the proposed EPU. CR-3 further concludes that the RSWPS will continue to meet requirements of 10 CFR 20.1302, 10 CFR 71, and FSAR Sections 1.4.17, 1.4.18, and 1.4.70. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Radioactive Solid Waste Packaging System.

2.5.6.3.4 References

1. Supplemental Environmental Report Extended Power Uprate, Crystal River Unit 3 Nuclear Power Plant, June 2009.

2. NUREG-0017, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR GALE code),” Revision 1, April, 1985.

Table 2.5.6.3-1: Solid Low-Level Radioactive Waste Generated at CR-3, 2004 - 2008

	Spent resins, filter sludges, evaporator bottoms, etc.		Dry active waste, equipment, etc.		Other		Total	
Year	**ft^3**	**Ci**	**ft^3**	**Ci**	**ft^3**	**Ci**	**ft^3**	**Ci**
2004	2,497	193	11,050	0.175	1,487	2.39	15,040	2.58
2005	480	506	13,000	82.9	600	1.59	14,080	591
2006	1,536	211	3,920	0.114	206	4.18	5,662	4.32
2007	306	476	12,010	1.38	604	2.65	12,920	480
2008	2,133	343	7,063	0.36	1,282	8.86	10,480	352

2.5.7 Additional Considerations

2.5.7.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

2.5.7.1.1 Regulatory Evaluation

Nuclear power plants are required to have redundant onsite emergency power supplies of sufficient capacity to perform their safety functions (e.g., power diesel engine-driven generator sets), assuming a single failure. The CR-3 review focused on increases in emergency diesel generator electrical demand and the resulting increase in the amount of fuel oil necessary for the system to perform its safety function.

The NRC's acceptance criteria for the Emergency Diesel Engine Fuel Oil Storage and Transfer System are based on:

- GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks;
- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and
- GDC-17, insofar as it requires onsite power supplies to have significant independence and redundancy to perform their safety functions, assuming a single failure.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, Missile Protection, insofar as it requires that SSCs important to safety be protected against dynamic effects, including missiles, pipe whip, and jet impingement forces associated with pipe breaks [GDC-4];
- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC-5]; and
- FSAR Section 1.4.24, Emergency Power For Protection Systems, and 1.4.39, Emergency Power For Engineered Safety Features, insofar as it requires onsite power supplies to have significant independence and redundancy to perform their safety functions, assuming a single failure [GDC-17].

2.5.7.1.2 Technical Evaluation

Introduction

The Emergency Diesel Generators (EDG) Fuel Oil Storage System provides sufficient fuel to permit the EDGs to perform their safety function. Each EDG has an independent underground storage tank which can be cross-tied to supply a single EDG for seven days at the upper limit of the 200-hour rating. Additionally, each EDG has a day tank to supply the EDG for one hour at all actual event-specific profiles. These systems will not be modified as part of the CR-3 EPU.

Description of Analyses and Evaluations

The EDG Fuel Oil Storage System is designed for a 7-day capacity assuming the upper limit of the EDG 200-hour rating. FSAR Section 8.2.3.1.3 states that the worst case EDG steady state auto-connected load, including momentary short duration loading, is less than the upper limit of the 200-hour rating of 3400kW and the worst case EDG steady state auto-connected load plus essential manual loads is less than 3300 kW. Since there are no load additions or modifications that will challenge the current 200-hour rating, no changes are necessary to the EDG Fuel Oil Storage and Transfer System and the current EDG fuel oil storage requirements remain bounding for EPU conditions.

Results

Because the EDG fuel oil storage requirements for CR-3 are based upon the amount of fuel oil that is consumed by the EDGs when they are operating at their 200-hour design rating, and the EDG electrical loads for EPU operation will not exceed the EDG 200-hour rating, the fuel oil storage requirements for CR-3 are not affected by the proposed EPU. Therefore, an evaluation of the EDG fuel oil storage requirements is not required.

2.5.7.1.3 Conclusion

The Emergency Diesel Engine Fuel Oil Storage and Transfer System has been evaluated with regards to the amount of required fuel oil for the emergency diesel generators, and has adequately accounted for the effects of the increased electrical demand on fuel oil consumption. CR-3 concludes that the Emergency Diesel Engine Fuel Oil Storage and Transfer System will continue to provide an adequate amount of fuel oil to allow the diesel generators to meet the onsite power requirements of FSAR Sections 1.4.4, 1.4.24, 1.4.39, and 1.4.40. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Emergency Diesel Engine Fuel Oil Storage and Transfer System.

2.5.7.1.4 References

None

2.5.7.2 Light Load Handling System (Related to Refueling)

2.5.7.2.1 Regulatory Evaluation

The Light Load Handling System (LLHS) includes components and equipment used in handling new fuel at the receiving station and the loading of spent fuel into shipping casks. The review covered the avoidance of criticality accidents, radioactivity releases resulting from damage to irradiated fuel, and unacceptable personnel radiation exposures. The CR-3 review focused on the effects of the new fuel on system performance and related analyses.

The NRC's acceptance criteria for the Light Load Handling System are based on:

- GDC-61, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection; and

- GDC-62, insofar as it requires that criticality be prevented.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.68, Fuel & Waste Storage Shielding, and FSAR Section 1.4.69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage, insofar as they require that systems that contain radioactivity be designed with appropriate confinement and with suitable shielding for radiation protection [GDC-61]; and

- FSAR Section 1.4.66, Prevention of Fuel Storage Criticality, insofar as it requires that criticality be prevented. [GDC-62]

2.5.7.2.2 Technical Evaluation

<u>Introduction</u>

The function of the LLHS is to permit the movement of new and spent fuel at the receiving stations. It consists of the Auxiliary Building Overhead Crane and associated equipment.

<u>Description of Analyses and Evaluations</u>

CR-3 will continue to utilize the same fuel design to achieve EPU conditions (Section 2.8.1, Fuel System Design). As such, the weight of the fuel and overall assembly design remains the same post-EPU. Therefore, there will be no modification to the LLHS to support the EPU.

Additionally, there will be no modifications made to the spent fuel cask handling equipment as part of the CR-3 EPU. In accordance with FSAR Section 9.6.4.7, CR-3 has committed to not use the overhead crane for spent fuel cask handling until the associated overhead crane is upgraded to meet singe-failure criteria.

Results

Based on the above, the Fuel Handling System does not change post-EPU. Therefore, the EPU will not affect the capability of the LLHS to perform its specified functions and an evaluation of the LLHS for the proposed power uprate is not required.

2.5.7.2.3 Conclusion

The LLHS has been evaluated to determine the effects of the new fuel on the ability of the LLHS to avoid criticality accidents including the effects of the new fuel in the analyses. Based on this review, the LLHS will continue to meet the requirements of FSAR Sections 1.4.66, 1.4.68, and 1.4.69 for radioactivity releases and prevention of criticality accidents. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Light Load Handling System .

2.5.7.2.4 References

None

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

2.6.1.1 Regulatory Evaluation

The containment encloses the Reactor System and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident.

The CR-3 review covered the pressure and temperature conditions in the containment due to a spectrum of postulated loss-of-coolant accidents (LOCAs) and main steam line breaks (MSLB).

The NRC's acceptance criteria for Primary Containment Functional Design are based on:

- GDC-13, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions;
- GDC-16, insofar as it requires that the reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment;
- GDC-38, insofar as it requires that the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels;
- GDC-50, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA; and
- GDC-64, insofar as it requires that means be provided for monitoring the plant environs for radioactivity that may be released from normal operations and postulated accidents.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the general design criteria (GDC) provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are the applicable CR-3 specific criteria.

- FSAR Section 1.4.10, Containment, insofar as it requires that the reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment [GDC-16];
- FSAR Section 1.4.12, Instrumentation and Control Systems, and FSAR Section 1.4.16, Monitoring Reactor Coolant Pressure Boundary, insofar as it requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and accident conditions [GDC-13];

- FSAR Section 1.4.17, Monitoring Radioactivity Release, insofar as it requires that means be provided for monitoring the plant environs for radioactivity that may be released from normal operations and postulated accidents [GDC-64];

- FSAR Section 1.4.49, Containment Design Basis, insofar as it requires that the containment and its internal components be able to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA [GDC-50]; and

- FSAR Section 1.4.52, Containment Heat Removal Systems, insofar as it requires that the containment heat removal system(s) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels [GDC-38].

2.6.1.2 Technical Evaluation

Introduction

Primary and secondary system piping breaks inside the reactor building may result in significant releases of high-energy fluid to the reactor building that could produce high pressure and temperature conditions for extended periods of time. Pressure and temperature conditions within the reactor building under accident conditions are evaluated to ensure the reactor building can prevent a significant release of radioactive fission products. The mass and energy (M&E) release to the reactor building and the subsequent containment response from limiting line breaks were reanalyzed for EPU conditions (refer to Sections 2.6.3.1, Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents, and 2.6.3.2, Mass and Energy Release Analysis for Secondary System Pipe Ruptures).

A spectrum of LOCA cases and a bounding secondary side MSLB case are addressed in this evaluation. The spectrum of LOCA cases considered break location, break type, and postulated single failures. The spectrum of cases for the containment response analyses are listed in Table 2.6.1-1. Short-term containment pressure and temperature responses (LOCA and MSLB) were evaluated to demonstrate compliance to peak pressure and temperature acceptance criteria. Long-term LOCA analyses were evaluated to demonstrate compliance to containment peak pressure and temperature limits for extended periods. Only the results for the limiting cases are presented.

The containment analyses described herein for both the LOCA and MSLB utilized the same NRC-approved methodology (Reference 1) as used in the current licensing bases that support a core power level of 2609 MWt. The CR-3 current licensing basis for a LOCA is presented in FSAR Section 14.2.2.5.9, Reactor Building (RB) Design Basis Accident. FSAR Tables 14-45 and 14-46 present the major assumptions used to support the FSAR reactor building design basis analyses. The CR-3 current licensing basis for a steam line break is presented in FSAR Section 14.2.2.1, Steam Line Failure Accident. FSAR Tables 14-27, 14-27a, and 14-45 present the major assumptions used to support the FSAR reactor building pressure analyses.

Inputs to the reactor building containment analyses for the EPU evaluation are consistent with those of the current licensing analysis that supported the core power level of 2609 MWt, except for those containment inputs identified in Table 2.6.1-7. The updated inputs are discussed below.

- The initial power level and Reactor Coolant System (RCS) average temperature are increased (see Sections 2.6.3.1, Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents, and 2.6.3.2, Mass and Energy Release Analysis for Secondary System Pipe Ruptures).

- The stainless steel thermal conductivity for heat structure components in containment was conservatively decreased for consistency to published data. Since a small amount of stainless steel is used in containment in comparison to other heat sink materials, the update to the thermal conductivity has an insignificant impact on the reactor building pressure during design bases events.

- The replacement once-through steam generator design parameters have been incorporated in the containment response analyses, and are of similar design to the original components. Therefore, the replacement steam generators do not significantly impact the containment response.

- The initial containment pressure assumed for the short-term containment pressure response was changed from 17.7 psia (3 psig) to 16.2 psia (1.5 psig) for a subset of cases determined to be the most limiting. This change was necessary to ensure adequate margin to peak pressure and temperature acceptance criteria at EPU conditions. The Improved Technical Specification (ITS) Limiting Condition for Operation (LCO) 3.6.4 maximum value for containment pressure under normal operation is being revised from 17.7 psia (3 psig) to 16.2 psia (1.5 psig) in association with the EPU, as described in Attachment 2, ITS Markups. The LOCA short-term results reported herein are based on the assumption of initial containment pressure of 1.5 psig, and the other reactor building analyses presented herein conservatively assume a higher initial reactor building pressure with respect to the planned ITS changes for the EPU.

  The long-term LOCA cases utilized an initial containment pressure of 16.7 psia (2 psig) while 17.7 psia (3 psig) was modeled for the MSLB case. These results represent bounding conditions with respect to the planned ITS changes for the EPU.

- Indicated pressurizer level assumed for the short-term containment pressure response was changed from 220 inches to 290 inches for a subset of cases determined to be the most limiting.

The specific acceptance criteria for the LOCA and MSLB primary containment responses are as follows:

- For the MSLB and short-term LOCA events, the peak calculated containment pressure should be less than the containment design pressure of 69.7 psia (55 psig) and the peak calculated containment temperature should be less than the containment design temperature of 281°F; and

- For the long-term LOCA event, the calculated pressure at 24 hours should be less than 50% of the peak calculated value, adequate NPSH margin exists, and containment temperatures remain bounded by the EQ envelope.

Description of Analyses and Evaluations

Calculation of the containment response following a postulated LOCA and MSLB event was analyzed using the NRC-approved GOTHIC methodology described in Reference 1. GOTHIC is a general purpose thermal-hydraulic computer program for design, licensing, safety, and operating analysis of nuclear power plant containments and containment sub-compartment and other confined areas. GOTHIC is a state-of-the-art program that solves the conservation equations, for mass, momentum, and energy for multi-component, multi-phase flow. GOTHIC version 7.2a was used for the LOCA and the MSLB containment response analyses.

The GOTHIC containment modeling for the CR-3 EPU evaluation is consistent with the NRC-approved evaluation model (Reference 1), which was approved for application to large dry containment designs that have active heat removal capabilities. The evaluation model (Reference 1) has been approved to evaluate the peak pressure and temperature of a large, dry, PWR containment atmosphere to large pipe breaks in high energy piping systems, and to evaluate long-term containment response following a design-basis LOCA. The NRC-approved containment response from GOTHIC has been qualified by comparison to experimental data for containment facilities on various scales (Reference 1).

The major modeling input parameters and assumptions used in the CR-3 containment evaluation model for the containment response events are summarized below.

- **Noding Structure** – The CR-3 GOTHIC containment evaluation model consists of one lumped volume. Additional boundary conditions, volumes, flow paths, and components are used to model accumulator nitrogen release and sump recirculation effects. The sump recirculation system and accumulator nitrogen release are modeled using boundary conditions.

- **Volume Input** – GOTHIC requires the volume, height, diameter, and elevation input values for each node. The containment is modeled as a single control volume in the containment model. The minimum free volume of 2,000,000 ft^3 is used. The height is calculated based on the cross-sectional area of the containment cylinder and the hydraulic diameter is determined by considering the volume along with the total surface area of the containment heat structures. The area of the liquid/vapor interface is calculated by the GOTHIC code.

- **Flow Paths** – Flow paths connect the boundary conditions to the containment volume. Standard values are used for the area, hydraulic diameter, friction length, and inertia length of the flow path, since the flow rate is specified by the boundary condition. For this single volume model, the elevation of the break flow path is set to 60 feet, and the elevation of the spray flow path is set conservatively to 96 feet to minimize the amount of time the spray can absorb heat from the atmosphere before entering the pool. Break flow is modeled as either a Double Ended Rupture (DER) or split break. For the M&E releases, flow from both sides of the break (i.e., two paths) is modeled for a DER. Flow boundary conditions are used to model the LOCA break flow to the containment. The boundary conditions are linked to functions that define the M&E of the break flow. The boundary conditions are connected to the containment control volume via flow paths.

- **Heat Sinks** – The heat sinks in the containment are modeled as GOTHIC thermal conductors. The heat sink data is based on conservatively low surface areas and is summarized in Table 2.6.1-5. The volumetric heat capacity and thermal conductivity for the heat sink materials is summarized in Table 2.6.1-6. The specific heat value was calculated based on the volumetric heat capacity.

- **Heat and Mass Transfer Correlations** – GOTHIC has a number of heat transfer coefficient options that can be used for containment analyses. The Tagami and Uchida heat transfer correlations are utilized for the LOCA cases, while the MSLB case only utilizes Uchida. These correlations are applied to the exposed portion of all the heat sinks inside containment, except for the RB floor, which uses liquid natural convection. This heat transfer methodology was reviewed and approved for the analysis of large, dry containment buildings (Reference 1). The revaporization fraction assumed with the Uchida correlation is determined by the default GOTHIC condensate revaporization model for the LOCA cases and limited to 8% for the MSLB case, consistent with the methodology approved in Reference 1.

- **LOCA Long-Term Sump Recirculation** – For long-term containment response, sump recirculation is modeled in GOTHIC following the sump swap-over. The post-sump swap-over Emergency Core Cooling System (ECCS) flow represents the ECCS flow from the containment sump to the reactor vessel, including any containment liquid spillover flow as dictated by the accident conditions. A one-pass shell and tube Decay Heat Heat Exchanger (DHHE) is modeled to cool the Low Pressure Injection (LPI) prior to injection into the RCS using a GOTHIC heat exchanger component and the manufacturer's as-built specifications. In addition, the source water for the reactor building spray in the long-term containment model is switched from a Borated Water Storage Tank (BWST) boundary condition to a containment sump control volume.

- **Reactor Building Fan Cooler** – The Reactor Building Cooling Unit (RBCU), or a reactor building fan cooler, is modeled in the containment evaluation model as a GOTHIC cooler component with a specified heat removal rate. The assumed initial conditions and input assumptions associated with the RBCU are listed in Table 2.6.1-3.

  The number of RBCUs credited is dependent on the break type and acceptance criteria. For the LOCA short-term and MSLB analyses, only one of the available three RBCUs, operating in slow speed, is credited. The limiting long-term LOCA case for half the peak pressure in 24 hours credits one RBCU operating at slow speed. The limiting long-term LOCA case for sump temperature does not credit any RBCUs.

  The RBCU is modeled to actuate after a specified 25-second delay. The heat removal rate for the RBCU is given as a function of containment steam saturation temperature and is presented in Table 2.6.1-4. The heat removal rate is read into a GOTHIC function to calculate the heat removal rate from containment. The heat removal rate is based on the minimum design value for the fan cooler of 80×10^6 Btu/hr with a fan capacity of 50,000 cfm, a RB temperature of 281°F, and a cooling water inlet temperature of 110°F.

- **Reactor Building Spray** – The RB spray is modeled in GOTHIC with a boundary condition. The assumed initial conditions and input assumptions associated with the RB sprays are listed in Table 2.6.1-3. The RB spray is modeled to actuate on the containment high-high pressure engineered safeguards signal and to begin injecting 120°F water initially drawn from the BWST after a specified 90 second delay. Since there are no RB spray heat exchangers, the temperature of the RB spray flow is set conservatively high relative to the BWST temperature of 100°F to minimize the ability of the sprays to cool the containment environment. The RB spray flow is conservatively biased based on the assumed equipment functionality. For the long-term containment response, the post-sump swap-over RB spray is connected to the containment control volume, and the flow rate is also modeled as a GOTHIC boundary condition.

- **Core Flood Tank Nitrogen Gas Modeling during LOCA** – Nitrogen cover gas in the core flood tanks (CFT) is modeled within GOTHIC as a non-condensable gas that contributes to the overall pressure in containment when it is injected after the CFTs empty. A mass conservation equation is calculated explicitly for nitrogen gas when introduced into the system, and the gas properties are evaluated by an equation of state. For conservatism in the LOCA containment model, it is assumed that all the nitrogen gas is injected into the containment volume by a direct connection from the CFT volume to the containment control volume.

  The nitrogen gas release rate is modeled with a GOTHIC boundary condition. The release rate is conservatively calculated by maximizing the mass available to be injected. The nitrogen gas release rate was used as input for the GOTHIC function, as a specified rate over a fixed time period. Nitrogen gas was released at a rate of 193.4 lbm/seconds, beginning at 37.51 seconds (earliest core flood tank water volume empty time) and ending at 50.65 seconds.

- **Blowdown Drop Diameter** – The liquid portion of the break flow for a LOCA containment analysis is released as drops with an assumed diameter of 100 microns (0.00394 in). For a MSLB analysis, the blowdown drop diameter is removed as the MSLB blowdown is considered to be entirely steam. This is consistent with the approved methodology and is based on data presented in Reference 1.

- **Initial Conditions** – The containment initial conditions, listed below, are conservatively set to maximize the containment pressure and temperature response. The containment pressure is initialized to the maximum allowable, consistent with proposed ITS changes (refer to Attachment 2). The initial relative humidity is assumed to be conservatively low in order to increase the partial pressure of the non-condensables, thereby reducing the condensation and overall effectiveness of the heat conducting surfaces. Finally, the containment temperature is initialized to the maximum allowable per Technical Specifications to elevate the initial heat sink temperatures, and further degrade their effectiveness. The containment initial conditions are listed below.

Pressure:	16.2 psia
Relative Humidity:	0%
Temperature:	130°F

It should be noted that the limiting short-term LOCA containment pressure analyses were analyzed with an initial containment pressure of 16.2 psia (1.5 psig). The long-term LOCA analyses were all analyzed at an initial pressure of 16.7 psia (2 psig). The MSLB analysis is analyzed at the same above conditions, except for the initial pressure, which was set to 17.7 psia (3 psig). The ITS LCO 3.6.4 maximum value for containment pressure under normal operation is being revised from 17.7 psia (3 psig) to 16.2 psia (1.5 psig) in association with the EPU to ensure adequate margin to the specific acceptance criteria. The initial containment temperature and relative humidity, assumed for the EPU, are consistent with the current licensing analysis.

The CR-3 containment analysis considered a spectrum of cases as listed in Table 2.6.1-1. The cases address break location, break type, availability of offsite power, ECCS flow rate, and reactor building (RB) back pressure. In each case, the conditions defined in Table 2.6.1-1 consider one of three postulated single failure scenarios. The first is a failure of one of the emergency diesel generators (EDGs) to start (minimum ECCS). In this scenario, it is assumed that there is loss of offsite power (LOOP) and a subsequent failure of one EDG causing the loss of one containment safeguards train. Each safeguards train powers one RB spray pump and one RBCU (fan cooler). With one safeguards train lost, only one RB spray pump and one RBCU is available for heat removal. The minimum ECCS failure scenario results in the minimum ECCS available for heat removal, and the longest delay in ECCS response.

The second scenario is a failure of one RB spray pump to start with or without offsite power available. This scenario results in the maximum ECCS injection available for heat removal and the shortest delay in ECCS response.

A third single failure scenario is the failure of all RBCUs. This scenario was evaluated for long-term containment response. In this scenario, both trains of ECCS and RB spray are available, depleting the BWST faster and resulting in an earlier switchover to the containment sump. After sump switchover, RB spray will be less effective at cooling the containment atmosphere because of the warmer sump water when compared to the cooler BWST water. This scenario is expected to challenge EQ concerns.

The LOCA and MSLB containment analyses are performed in two steps. First, the mass and energy release (M&E) rates to the containment are calculated. For the M&E release rates, the NRC-approved Topical (Reference 1) is used in conjunction with the RELAP5/MOD2-B&W computer code (Reference 2) to simulate M&E releases through a primary system pipe break. The LOCA M&E release analysis is discussed in Section 2.6.3.1, and Table 2.6.3.1-1 lists the key analysis parameters and initial conditions. The MSLB M&E release analysis is discussed in Section 2.6.3.2, Mass and Energy Release Analysis for Secondary System Pipe Ruptures.

The second step in the containment analyses is to evaluate the containment response following a LOCA or MSLB using the NRC-approved GOTHIC computer code (Reference 1). The methodology requires the use of conservative setpoints, available spray flow rates, and heat removal rates to demonstrate adequate margin to the applicable acceptance criteria. The M&E releases from RELAP5/MOD2-B&W simulations are input to the GOTHIC containment model through forcing functions on the flow boundary conditions. Table 2.6.1-3, Table 2.6.1-4, and Table 2.6.1-5 list the important analysis parameters and initial conditions for the GOTHIC containment analyses.

LOCA containment analyses consist of both short-term and long-term containment responses as indicated in Table 2.6.1-1. The short-term containment response extends to the time of at least core quench. Short-term containment analyses are based on RELAP5/MOD2-B&W M&E releases input into GOTHIC as boundary conditions. A full spectrum of short-term cases, as listed in Table 2.6.1-1, was analyzed to confirm the most limiting initial conditions and assumptions. Long-term containment responses are defined by the conditions following core quench, where the vessel level has recovered to the RCS loop nozzle elevations, and the ECCS injection ensures the core remains covered so that core decay heat removal and sensible heat removal is assured at all times.

Long-term GOTHIC cases, listed in Table 2.6.1-2, were analyzed at the hot leg break location, with additional cases chosen to validate the break location and demonstrate the effectiveness of offsite power and ECCS flow rates. Included in this spectrum are configurations with a complete loss of the RBCUs as the single failure. Two trains of RB Spray are available, along with full ECCS capacity. After sump recirculation is initiated, the RB Spray draws directly from the sump, which limits the effectiveness of the spray, due to the warmer sump water.

The long-term analyses considered all break locations, however the cold leg pump discharge break (CLPD) location was not explicitly analyzed for the long term because the break location was determined to be non-limiting. The CLPD break would allow the RCS to refill only up to the elevation of the cold leg nozzles. Continued decay heat generation would lead to steam production in the core, with the steam generated passing from the upper plenum into the upper downcomer via the reactor vessel vent valves. The reactor vessel vent valves open at a very low delta-P to allow steam flow into the downcomer. The LPI nozzles deliver injection flow near the top of the downcomer, such that the steam passed through the vent valves interacts with abundant, cold LPI prior to reaching the break. The condensation potential of the LPI is much greater than the steam generated due to decay heat. Because the steam passes through the vent valves, the steam flow through the RCS loops is small, especially in the intact loop. Therefore, the dissipation of stored energy from the RCS metal and the steam generators is slower than that from other break locations. Therefore, the CLPD break is not limiting for elevated temperatures in the long term.

The long-term analyses include the time period after core quench and include transfer to sump recirculation. The long-term response is analyzed with GOTHIC using the methodology described in Reference 1. The EPU is the first application of the long-term containment response methodology to CR-3, however the methodology is approved by the NRC for application to large, dry containments. The LOCA long-term M&E releases are modeled in GOTHIC by adding a node to represent the RCS as well as including heat sources to represent sensible heat and decay heat. With these additions, the GOTHIC model is capable of calculating the long-term M&E releases from the RCS to containment. Sensible heat from primary fluid stored energy is accounted for by transferring the fluid conditions at the time of transition into a single control volume representing the RCS. Primary system passive metal stored energy is summed at the end of the RELAP5 calculation, and is apportioned to the RCS liquid for metal in contact with liquid at the time of transition, or to the containment vapor space for metal in contact with steam at the time of transition. Secondary system stored energy (fluid plus metal) is conservatively dissipated into the containment vapor space. The rate of dissipation of the primary passive metal and secondary stored energy is based on the dissipation rate near the end of the short-term M&E release period. Decay heat is calculated by use of the ANS 1971 decay heat standard (plus heavy actinides), and is modeled in GOTHIC as a forcing function to a heater component assigned to the vessel node.

This decay heat bounds the ANS Standard 5.1 (+2-sigma uncertainty) for the determination of long-term boil-off from core, consistent with the long-term M&E methodology documented in Reference 1.

The short-term containment response for the MSLB event was also analyzed with GOTHIC methodology (Reference 1). Plant input assumptions for the MSLB analysis are the same or similar to those in the current licensing basis analysis as described in FSAR Section 14.2.2.1. These assumptions include a bounding Double Ended Rupture (DER) downstream of the steam generator outlet nozzle, a single failure such that the main feedwater pumps do not trip, and an initial full EPU power to maximize the steam generator inventory.

Results

The results of the limiting LOCA (both short-term and long-term) and MSLB analyses are presented below. A summary of the limiting LOCA and MSLB results discussed below are captured in Table 2.6.1-11 and Figure 2.6.1-1 through Figure 2.6.1-8.

Short-Term Containment Results (Peak Containment Pressure)

The most limiting event for the containment peak pressure response is a double-ended rupture on the hot leg nearest the steam generator with a single failure of an EDG to start. This analysis assumes a LOOP coincident with the associated single-failure assumption of an EDG to start. This results in one train of ECCS and containment safeguards equipment being available for heat removal. Under these conditions, one RB spray pump and one RBCU are assumed operable. Further, the LOOP delays the actuation times of the safeguards equipment due to the time required for the EDG startup after receipt of the safety injection signal.

The postulated RCS break results in a rapid release of M&E to the containment. The containment pressure continues to rise rapidly in response to the release of M&E, reaching the peak pressure of 68.66 psia (53.96 psig) at 25.0 seconds, and then decreases throughout the remainder of the short-term transient. The end of blowdown marks a time when the initial inventory in the RCS has been exhausted, and it is about this timeframe that the peak containment pressure occurs. From that point forward, filling the RCS downcomer in preparation for reflood has begun. Since the M&E release during this period is low, pressure continues to decrease. At approximately 37.5 seconds it is conservatively assumed that the core flood tanks have emptied and the introduction of nitrogen cover gas begins. The impact is not enough to result in a second pressure peak, and in fact is minimal, especially for the hot leg breaks that are limiting for peak pressure.

During this period the RBCUs (25.0 seconds) and RB spray (90.0 seconds after the high-high containment signal) have also started and are removing heat. Reflood continues at a reduced flooding rate due to the buildup of mass in the RCS core, which offsets the downcomer head. This reduction in flooding rate and the continued action of the RBCUs and RB spray leads to a slowly decreasing containment pressure. As the end of reflood approaches, the analysis transitions into the long-term phase.

The sequence of events for this LOCA case is listed in Table 2.6.1-8. The containment pressure and vapor temperature profiles for this LOCA case are shown in Figure 2.6.1-1 and Figure 2.6.1-2.

Short-Term Containment Results (Peak Vapor Temperature)

The most limiting event for the peak containment vapor temperature is a double-ended rupture of the main steam line immediately downstream of the SG exit nozzles, with a failure of the affected main feedwater pump to trip. This case resulted in a peak containment vapor temperature of 477.1°F occurring at 108 seconds. Although this temperature exceeds the design temperature of 281°F, it is not of concern for EQ purposes as this peak occurs for a short period of time (260 seconds, or just over 4 minutes). For EQ concerns, it is the operation at high temperatures for relatively long periods of time that increases the possibility of component failure.

The impact of the MSLB containment vapor temperature on equipment has been qualified previously for more extreme LOCA conditions. In the EPU MSLB containment analyses, the containment vapor temperature is calculated to be above the 281°F containment design temperature for approximately 260 seconds. However, the containment equipment has been previously qualified to a LOCA containment analysis temperature profile that is above 281°F for approximately 3500 seconds, and above 250°F for nearly 40,000 seconds. Therefore, the equipment qualification conclusions previously reported and accepted remain valid under the presently predicted containment conditions.

The containment analyses performed to support environment qualification of equipment are biased to achieve high containment pressures and vapor temperatures. The result of this bias is to minimize the heat transfer to the containment structure itself. The MSLB containment analysis was re-performed to bias the results for maximum impact on the structure temperature per guidance provided in Appendix B of NUREG-0588 (Reference 3). The results of this analysis show that the structure wall temperature does not exceed the 281°F containment design temperature at anytime for the MSLB event. The maximum structure temperature predicted for this event is 260.5°F. Therefore, there is no impact to the containment structure integrity for the predicted MSLB vapor temperature exceeding the 281°F containment design temperature for approximately 260 seconds.

The sequence of events for the MSLB case is listed in Table 2.6.1-9. The containment pressure and vapor temperature profiles for the MSLB case are shown in Figure 2.6.1-3 and Figure 2.6.1-4.

LOCA Long-Term Containment Results

Following the transition from short-term to long-term, the stored energy in the RCS and steam generators is conservatively dissipated in the timeframe following the transition. The inventory of the Borated Water Storage Tank is depleted and ECCS is realigned for sump recirculation resulting in an increased injection temperature (due to the delivery from the hot sump and a reduction in steam condensation). The timing of sump recirculation is dependent on whether minimum or maximum ECCS is modeled. For minimum ECCS cases, sump recirculation occurs at about 3900 seconds. For the maximum ECCS cases, sump recirculation occurs at about 2300 seconds with one RB spray pump, and at about 2000 seconds with both RB spray pumps operating.

The sump switchover results in an increase in the containment pressure and temperature, but not to values that exceed the earlier peaks. Eventually, the energy removal from the operating fan cooler (for those cases that model a RBCU) exceeds the energy release and the pressure and temperature turn around. For those cases without RBCUs, the second spray pump aids in the reduction of containment pressure and temperature. This trend continues to the end of the transient at 24 hours (86,400 seconds), which is of sufficient duration to demonstrate the effectiveness of the heat removal systems as the containment pressure and temperature continue to decrease.

A 3.0 ft^2 double-ended rupture of the cold leg pump suction piping, with no LOOP and minimum ECCS is the most limiting event for the criterion that containment pressure reach half of its peak pressure within 24 hours. For this event, containment pressure reaches half of its peak in just over 10 hours, well within the 24 hour criterion.

A double-ended rupture on the hot leg nearest the steam generator with maximum ECCS, a loss of offsite power, both RB sprays operational, and no RBCUs produces the highest sump temperature after sump switchover. For this event, the sump reaches a peak temperature of 262.9°F at 6340 seconds (just over 1 hour, 45 minutes). This temperature increase has insignificant impact on pump NPSH (see Section 2.6.5).

Although the MSLB event produced the highest containment vapor temperature, it was for a relatively short period of time. For EQ concerns, however, the scenario with the longest sustained elevated vapor temperature is typically most limiting, which would be a LOCA. From the matrix of cases (identified in Table 2.6.1-1) that were run for long-term consequences, almost all of the cases converged into one temperature profile after 10,000 seconds. From this matrix, two hot leg LOCA cases were selected based on the post-recirculation temperature profile.

The first case is a double-ended rupture of a hot leg nearest the steam generator with minimum ECCS, offsite power available, and one RBCU and one RB spray train available is selected as this case produces the highest vapor temperature at 10,000 seconds, which coincides with a region under the EQ curve with relatively little margin. In addition, this case maintains a higher containment vapor temperature profile throughout the rest of the transient.

The second case identified is a double-ended rupture of a hot leg nearest the steam generator with maximum ECCS, a loss of offsite power, both RB sprays operational, and no RBCUs. This case produces the highest post-sump switchover vapor temperature overall, but declines at a faster rate than the other case identified above (minimum ECCS case with one RBCU and one RB spray available).

The sequence of events for these LOCA long-term cases is listed in Table 2.6.1-10. The containment pressure, vapor temperature, and water (sump) temperature profiles for these LOCA cases are shown in Figure 2.6.1-5 through Figure 2.6.1-8.

Results Summary

A review of the results presented in Table 2.6.1-11 shows that the EPU LOCA analysis peak pressure and MSLB temperature demonstrate that the current margins are maintained. The current licensing containment response basis (FSAR 14.2.2.5.9) results for containment peak pressure and temperature for a LOCA event is 68.74 psia and 276.6°F, respectively. For the containment response analysis performed in support of the CR-3 EPU, the EPU containment peak pressure and temperature is 68.66 psia and 278.9°F. Based on the containment results provided in Table 2.6.1-11, along with the MSLB vapor temperature disposition provided above, the applicable acceptance criteria are met. For the purposes of 10 CFR 50 Appendix J containment leakage testing, the value of Pa (defined as the calculated peak containment internal pressure related to the design basis LOCA) is conservatively selected to be 68.9 psia (54.2 psig) as shown in ITS 5.6.2.20.

The containment pressure at 24 hours is less than half of the peak pressure. The peak post-recirculation vapor temperature does not challenge the peak temperature. The peak post-recirculation vapor temperature is reduced by either RBCUs in conjunction with RB spray (for those cases that credit RBCUs) or two trains of RB spray (for those cases that do not credit RBCUs). For the scenario where no RBCUs are credited, the temperature rise after sump recirculation is more severe than that observed with one RB Spray train and one RBCU active, and results in the peak sump temperature.

As discussed in Section 2.3.1, Environmental Qualification of Electrical Equipment, the reactor building is evaluated to ensure that it will continue to perform its safety-related function in the post-LOCA containment environment. For purposes of equipment qualification, the evaluations cover a time period of up to 30 days post-LOCA. In support of the evaluation of safety-related equipment in the reactor building, the vapor temperature results for the limiting hot leg break are conservatively extended to 30 days, at which point reactor building conditions are assumed to have returned to pre-accident levels. The conservative extension of the reactor building temperatures beyond 24 hours is depicted in Figure 2.6.1-8.

In summary, a spectrum of LBLOCA cases were analyzed, along with a limiting MSLB case, with the acceptance criteria defined above. The limiting case for peak pressure was determined to be a double-ended rupture on the hot leg nearest the steam generator with minimum ECCS. The limiting scenario assumes a LOOP coincident with the associated single-failure assumption of an EDG to start. The peak pressure of 68.66 psia is less than the allowable peak pressure of 69.7 psia. The limiting case for peak temperature is a MSLB, although it exceeds the design temperature of 281°F, is still acceptable due to the length of time the containment is at the elevated temperature. All of the LOCA scenarios modeled remain well below the design temperature.

For long-term containment building pressure response, the cold leg pump suction case with minimum ECCS and offsite power available reaches a pressure that is half the peak in just over 10 hours, which is well within the acceptance criterion of 24 hours. A double-ended rupture on the hot leg nearest the steam generator with maximum ECCS, a loss of offsite power, both RB sprays operational, and no RBCUs produces the highest sump temperature after sump switchover. For this event, a peak sump temperature of 262.9°F is reached at 6340 seconds. Two hot leg rupture cases are identified for the limiting peak containment vapor temperature profile for EQ concerns. The first is a double-ended rupture of a hot leg nearest the steam generator with minimum ECCS, offsite power available and one RBCU and RB spray train available. The second limiting containment vapor temperature scenario is double-ended rupture of a hot leg nearest the steam generator with maximum ECCS, a loss of offsite power, both RB sprays operational, and no RBCUs.

2.6.1.3 Conclusion

CR-3 has reviewed the assessment of the containment pressure and temperature transient and concludes that it has adequately accounted for the increase of M&E that would result from the proposed EPU. CR-3 further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. CR-3 also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and will continue to meet the requirements of FSAR Sections 1.4.10, 1.4.12, 1.4.16, 14.17, 1.4.49, and 1.4.52 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Containment Functional Design.

2.6.1.4 References

1. BAW-10252P-A, Rev.0, "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC".

2. BAW-10164PA-06, "RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis".

3. NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," US NRC, July 1981.

Table 2.6.1-1: Short-Term Containment Analyses Case Matrix

Break Location	Break Type [1]	LOOP [2]	ECCS Flow Rate	RB Back Pressure [3]
Short-term LOCA Cases				
Cold Leg Pump Discharge	2A DER	Yes	Minimum	Maximum
Cold Leg Pump Discharge	2A DER	Yes	Minimum	Minimum
Cold Leg Pump Discharge	2A DER	No	Minimum	Minimum
Cold Leg Pump Discharge	2A DER	Yes	Maximum	Minimum
Cold Leg Pump Suction	2A DER	Yes	Minimum	Maximum
Cold Leg Pump Suction	2A DER	Yes	Minimum	Minimum
Cold Leg Pump Suction	2A DER	No	Minimum	Minimum
Cold Leg Pump Suction	2A DER	Yes	Maximum	Minimum
Cold Leg Pump Suction	7.0 DER	No	Minimum	Minimum
Cold Leg Pump Suction	5.13 DER	No	Minimum	Minimum
Cold Leg Pump Suction	3.0 DER	No	Minimum	Minimum
Cold Leg Pump Suction	2.0 DER	No	Minimum	Minimum
Cold Leg Pump Suction	0.5 DER	No	Minimum	Minimum
Cold Leg Pump Suction	2A DER	No	Maximum	Minimum
Hot Leg RV Nozzle	2A DER	Yes	Minimum	Maximum
Hot Leg RV Nozzle	2A DER	Yes	Minimum	Minimum
Hot Leg RV Nozzle	2A DER	No	Minimum	Minimum
Hot Leg RV Nozzle	2A DER	Yes	Maximum	Minimum
Hot Leg Near SG	2A DER	Yes	Minimum	Maximum
Hot Leg Near SG	2A DER	Yes	Minimum	Minimum
Hot Leg Near SG	2A DER	No	Minimum	Minimum
Hot Leg Near SG	2A Split	Yes	Minimum	Minimum
Hot Leg Near SG	2A DER	Yes	Maximum	Minimum
Hot Leg Near SG	11.0 DER	No	Minimum	Minimum
Hot Leg Near SG	8.55 DER	No	Minimum	Minimum
Hot Leg Near SG	5.0 DER	No	Minimum	Minimum
Hot Leg Near SG	2A DER	No	Maximum	Minimum
Hot Leg Near SG [4]	2A DER	No	Minimum	Minimum
Hot Leg Near SG [4] [5]	2A DER	No	Maximum	Minimum
Main Steam Line Break Case				
SG Outlet Nozzle	2A DER	No	Minimum	Minimum

Notes

(1) DER represents a Double Ended Rupture break type. 2A refers to a break flow area of twice the system piping flow area. If a numerical value is listed, the value refers to the total DER break flow area in units of ft^2.

(2) LOOP represents a Loss of Offsite Power.

(3) The RB back pressure refers to the downstream pressure for the RELAP5 break flow M&E release.

(4) Reflects sensitivity case: The coastdown duration for main feedwater flow was extended from 12 seconds to 17 seconds.

(5) Reflects sensitivity case: No delay time for ECCS injection is assumed.

Table 2.6.1-2: Long-Term Containment Analyses Case Matrix

Break Location	Break Type (1)	LOOP (2)	ECCS Flow Rate	RB Back Pressure (3)	RB Spray	RBCUs	BWST Temperature
Cold Leg Pump Suction	3.0 DER	No	Minimum	Minimum(6)	1 train	1	120°F
Hot Leg Near SG	2A DER	No	Minimum	Minimum(6)	1 train	1	120°F
Hot Leg Near SG	Split	Yes	Minimum	Minimum(6)	1 train	1	120°F
Hot Leg Near SG	2A DER	Yes	Maximum	Minimum(6)	1 train	1	120°F
Hot Leg Near SG (4)	2A DER	No	Minimum	Minimum(6)	1 train	1	120°F
Hot Leg Near SG	2A DER	Yes	Maximum	Minimum(6)	2 trains	None	120°F
Hot Leg Near SG	2A DER	Yes	Modified Maximum(5)	Minimum(6)	2 trains	None	100°F
Hot Leg Near SG	2A DER	Yes	Maximum	Minimum(6)	2 trains	None	100°F

Notes

(1) DER represents a Double Ended Rupture break type. 2A refers to a break flow area of twice the system piping flow area. If a numerical value is listed, the value refers to the total DER break flow area in units of ft^2.

(2) LOOP represents a Loss of Offsite Power.

(3) The RB back pressure refers to the downstream pressure for the RELAP5 break flow M&E release.

(4) Reflects sensitivity case: The coastdown duration for main feedwater flow was extended from 12 seconds to 17 seconds.

(5) Modified ECCS flow rates (HPI, LPI, and RB Spray) are modeled. HPI is 600 gpm/pump, LPI is 3600 gpm/train prior to sump recirculation and 2000 gpm/train afterward, and RB spray is 1625 gpm/train from the BWST and 1375 gpm/train from the RB sump.

(6) The RB back pressure linearly increases from 14.7 psia to 40.0 psia during the first 100 sec of the transient.

Table 2.6.1-3: Limiting Containment Analyses Parameters

Parameter	LOCA Cases	MSLB Case
Containment Dimensions and Initial Conditions		
Initial Containment Temperature (°F)	130	130
Initial Containment Pressure (psia)	16.2 (1) 16.7 (2)	17.7
Initial Relative Humidity (%)	0.0	0.0
Initial Volume Fraction	0.0	0.0
Initial Sump Liquid Level (ft)	0.0	0.0
Hydraulic Diameter (ft)	19.92	19.92
Height (ft)	150.7	150.7
Net Free Volume (ft^3)	2,000,000	2,000,000
Heat Structures and Material Properties		
Heat Structure Geometry and Material Type	Table 2.6.1-5 Table 2.6.1-6	Table 2.6.1-5 Table 2.6.1-6
Heat Transfer Correlations		
Heat sinks (except for RB floor)	Tagami-Uchida	Uchida
Reactor Building Floor	Liquid Natural Circulation	Liquid Natural Circulation
Reactor Building Fan Cooling Units		
Total	3	3
Credited in Analysis	0 – 1 (3)	1
Fan Capacity (cfm)	50,000	50,000
Heat Removal Capability	Table 2.6.1-4	Table 2.6.1-4
Containment High Setpoint (psia)	18.7	18.7
Delay Time (sec)	25.0	25.0
Reactor Building Spray Pumps		
Total	2	2
Credited in Analysis	1-2 (3)	1
Flow rate (gpm) (per pump)	1,000 pre-sump swap-over 1,200 post-sump swap-over	1,000 pre-sump swap-over 1,200 post-sump swap-over
Spray Temperature (°F)	120.0	120.0
Spray Drop Diameter (m)	0.0425	N/A
Spray Elevation (ft)	96.0	96.0
Containment High High Setpoint (psia)	44.7	44.7
Delay Time (sec)	90	90
ECCS Recirculation Switchover, sec Min ECCS Max ECCS, 2 RB spray trains	 3880 1994	N/A

1. An initial pressure of 16.2 psia was used for the most limiting peak pressure case
2. An initial pressure of 16.7 psia was used for the long-term LOCA cases
3. Equipment availability depends upon single failure assumptions

Table 2.6.1-4: Reactor Building Cooling Unit Heat Removal Capability

Temperature (°F)[1]	1 RBCU Removal Rate (Btu/sec)
110	0.0
120	1299.5
130	2599.1
140	3898.6
150	5198.2
160	6497.7
170	7797.3
180	9096.8
190	10396.4
200	11695.9
210	12995.5
220	14295.0
230	15594.5
240	16894.1
250	18193.6
260	19493.2
270	20792.7
280	22092.3
281	22222.2

[1] The RBCU removal capability provided above is a function of containment steam saturation temperature. The RBCU heat removal capability is based upon a minimum removal rate of 80 MBtu/hr with a fan capacity of 50,000 cfm, a RB temperature of 281°F, and a cooling water temperature of 110°F. The heat removal rate is input into GOTHIC as table for a range of inlet air temperatures, which is constructed by assuming a linear relationship between the conditions at the minimum removal rate and the conditions at a zero heat transfer point assumed to occur at the cooling water temperature of 110°F.

Table 2.6.1-5: Containment Structural Heat Sink Input

Segment	Material Type	Surface Area, ft^2	Thickness	Description
1	Paint	81,700	15 mils	Reactor Building Walls and Dome
	Steel	81,700	3/8 in	
	Air Gap	81,700	1/32 in	
	Concrete	81,700	3.4 ft	
2	S. Steel	6,000	1/4 in	Refueling Canal Stainless Steel Liner on Inside
	Air Gap	6,000	1/32 in	
	Concrete	6,000	3.73 ft	
	Paint	6,000	15 mils	
3	Paint	11,000	15 mils	Reactor Building Floor
	Concrete	11,000	2.0 ft	
4	Paint	196,900	15 mils	Misc Internal Steel
	Steel	196,900	0.268 in	
5	Paint	105,941	15 mils	Misc Internal Concrete
	Concrete	105,941	1.435 ft	

Table 2.6.1-6: Material Properties for Containment Structural Heat Sink

Material	Density (lbm/ft^3)	Thermal Conductivity (Btu/hr-ft^2-°F/ft)	Heat Capacity (Btu/°F–lbm)
Paint	103.0	0.20	0.35
Steel	490.0	26.0	0.12
Air	0.0721	0.0184	0.17
Concrete	145.0	0.45	0.156
S. Steel	503.0	10.0	0.12

Table 2.6.1-7: Updates to Containment Analysis Inputs for EPU

Parameter	Pre-EPU	EPU
Rated Core Thermal Power (MWt)	2609	3014
Analyzed Core Thermal Power (MWt)	2619.36	3026.1
Heat Structure Thermal Properties, Thermal Conductivity for Stainless Steel (Btu/hr-ft^2-°F/ft)	26.0	10.0
Steam Generator Design	OTSG	Replacement OTSG
Initial Pressurizer level (inches)	220	220 (1) 290 (1)
Initial Reactor Building Pressure (psia)	17.7	16.2 (2) 16.7 (2) 17.7 (2)

(1) LOCA limiting short-term containment analyses utilized an indicated pressurizer level of 290 inches. LOCA limiting long-term and MSLB analyses utilized a pressurizer level of 220 inches.

(2) LOCA limiting peak pressure analyses utilized an initial reactor building pressure of 1.5 psig, MSLB analysis utilized initial reactor building pressure of 3.0 psig, and LOCA long-term containment analyses utilized an initial reactor building pressure of 2.0 psig.

Table 2.6.1-8: Limiting LOCA Short-Term Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs
25.0	Peak Vapor Pressure (68.66 psia)
25.0	Fan Coolers Initiated
37.51	Nitrogen Cover Gas Injection
50.66	Nitrogen Cover Gas Injection Ends
90.0	Reactor Building Spray Flow Initiated
600.0	Short-Term Transient Ends

Table 2.6.1-9: Limiting MSLB Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs
25.0	Fan Coolers Initiated
90.0	Reactor Building Spray Flow Initiated
108.0	Peak Vapor Temperature (477.1°F)
600.0	Short-Term Transient Ends

Table 2.6.1-10: Limiting LOCA Long-Term Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs
 25.0 N/A	Fan Coolers Initiated 1 RBCU No RBCUs
37.51	Nitrogen Cover Gas Injection
50.66	Nitrogen Cover Gas Injection Ends
  93.6 100.6	Reactor Building Spray Flow Initiated (high containment pressure + 90 sec time delay) Hot leg LOCA (max EQ vapor and sump temperature cases) CLPS LOCA (max half pressure peak case)
 1994 3880	Switchover to sump recirculation Maximum ECCS, 2 RB spray trains Minimum ECCS, 1 RB spray train
6340	Maximum Sump Temperature (262.9°F)
86400 (24 hours)	Transient Modeling terminated

Table 2.6.1-11: LOCA Containment Response Results Comparison
Peak Pressure Allowed = 69.7 psia
Peak Temperature Allowed = 281°F
Peak Pressure Allowed at 24 hours = 50% of Calculated Peak

Parameter	Pre-EPU	EPU
Peak Containment Pressure Case		
Break Location	Hot Leg Near Steam Generator	Hot Leg Near Steam Generator
Break Type	Double Ended Rupture	Double Ended Rupture
Loss of Offsite Power (LOOP)	No	No
ECCS Flow Rate	Minimum	Minimum
Reactor Building Back Pressure	Minimum	Minimum
Peak Pressure (psia)	68.9 (21 sec)	68.66 (25.0 sec)
Peak Containment Vapor Temperature Case		
Break Location	Main Steam Line Break	Main Steam Line Break
Break Type	Double Ended Rupture	Double Ended Rupture
Loss of Offsite Power (LOOP)	No	No
ECCS Flow Rate	Minimum	Minimum
Reactor Building Back Pressure	Minimum	Minimum
Peak Vapor Temperature (°F)	491.26 (117 sec)	477.1 (108 sec)
Long-Term Containment Pressure Case		
Break Location	Cold Leg Pump Suction	Cold Leg Pump Suction
Break Type	Double Ended Rupture	Double Ended Rupture
Loss of Offsite Power (LOOP)	No	No
ECCS Flow Rate	Maximum	Minimum
Reactor Building Back Pressure	Minimum	Minimum
Containment Pressure (psia)	~26 (24 hours)	26.5 (24 hours)
Long-Term Containment Sump Temperature Case		
Break Location	Cold Leg Pump Suction	Hot Leg Near Steam Generator
Break Type	Double Ended Rupture	Double Ended Rupture
Loss of Offsite Power (LOOP)	No	No
ECCS Flow Rate	Maximum	Maximum
Reactor Building Back Pressure	Minimum	Minimum
Peak Sump Temperature (°F)	243	262.9 (6340 sec)

Figure 2.6.1-1: Limiting LOCA Short-Term Containment Pressure Response

(Double-ended hot leg break, offsite power available, min ECCS, extended MFW, min RB backpressure)

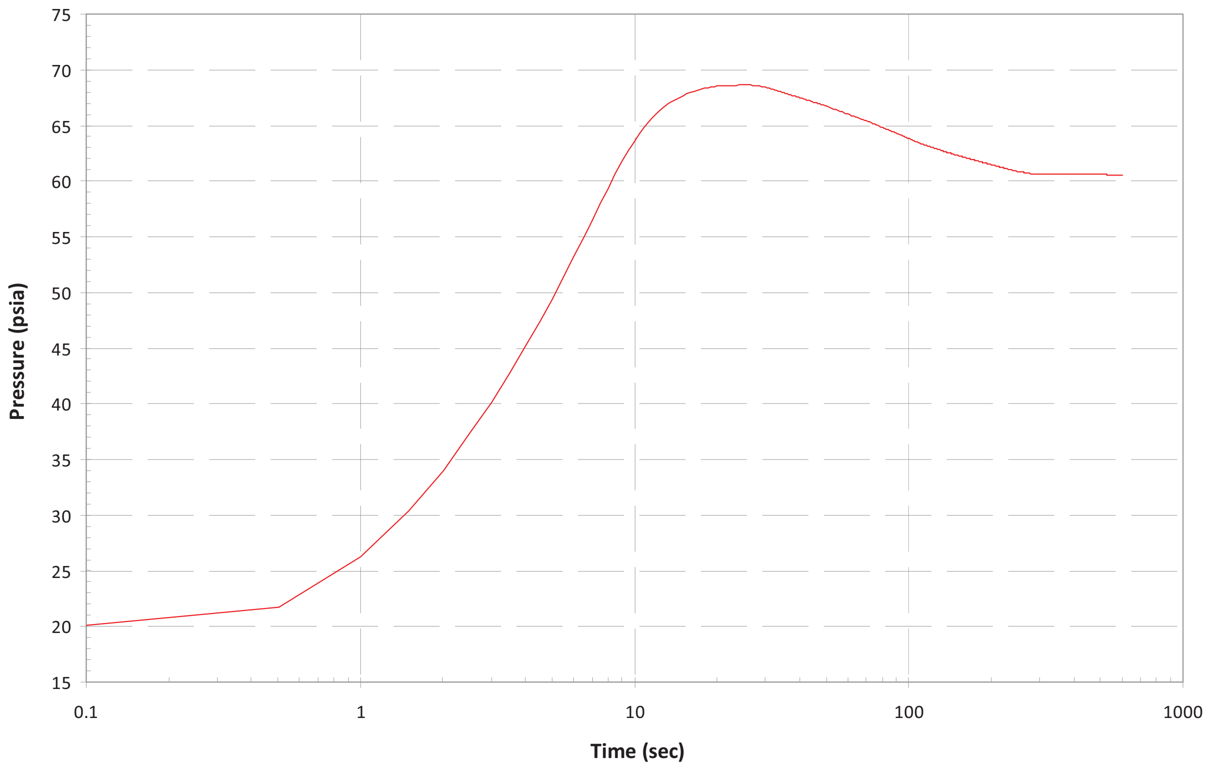


Figure 2.6.1-2: Limiting LOCA Short-Term Containment Vapor Temperature Response

(Double-ended hot leg break, offsite power available, min ECCS, extended MFW, min RB backpressure)

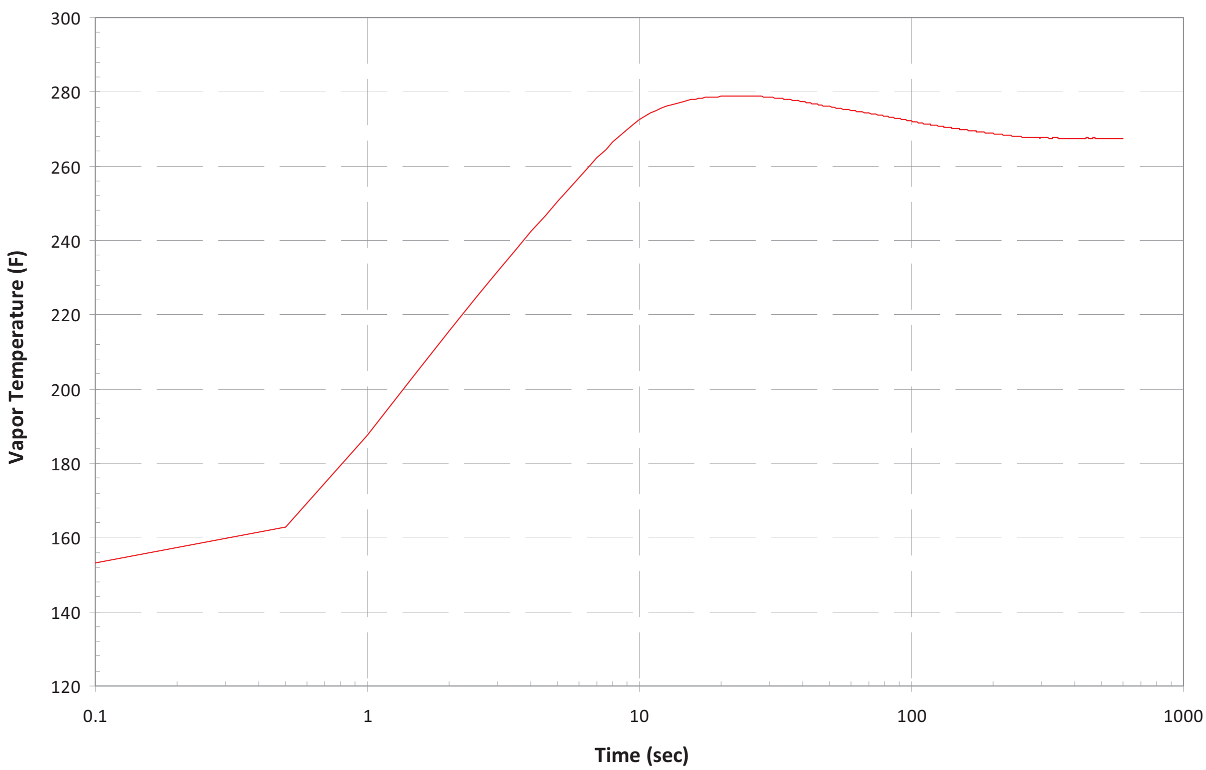


Figure 2.6.1-3: Limiting MSLB Containment Pressure Response

(DER downstream of SG outlet nozzle, failure to trip MFP, min ECCS, offsite power available)

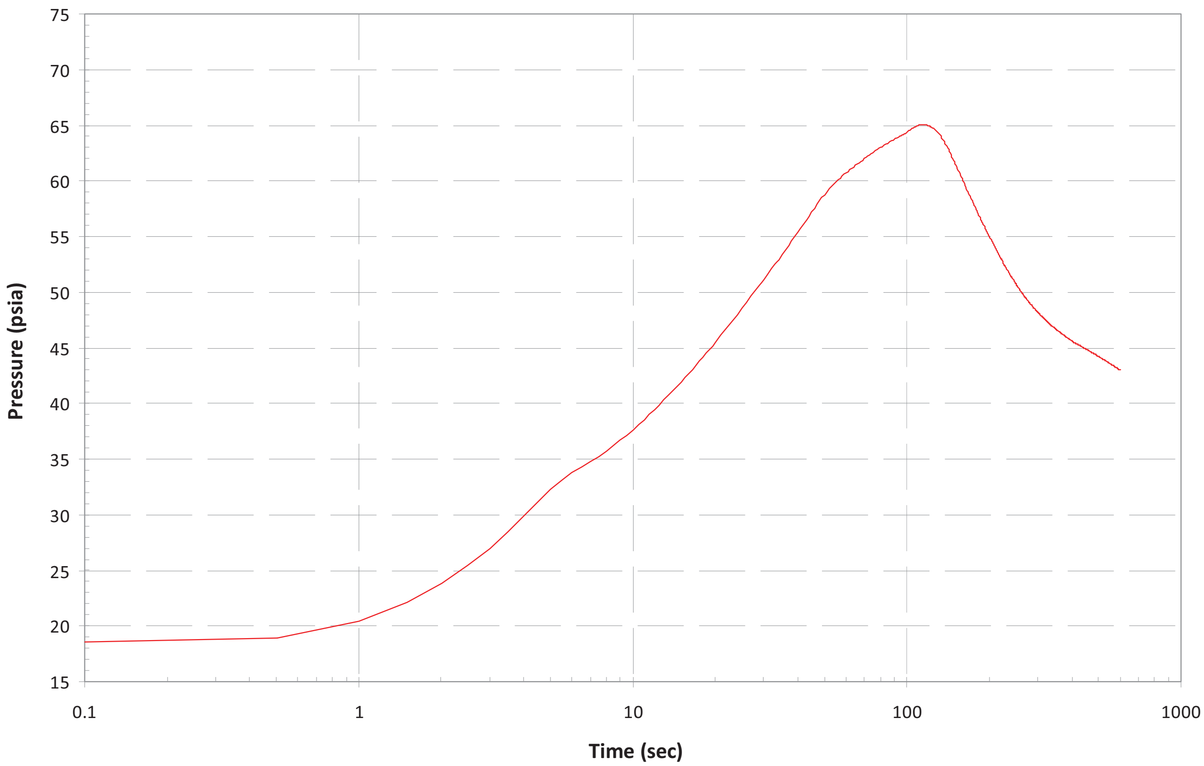


Figure 2.6.1-4: Limiting MSLB Containment Vapor Temperature Response

(DER downstream of SG outlet nozzle, failure to trip MFP, min ECCS, offsite power available)

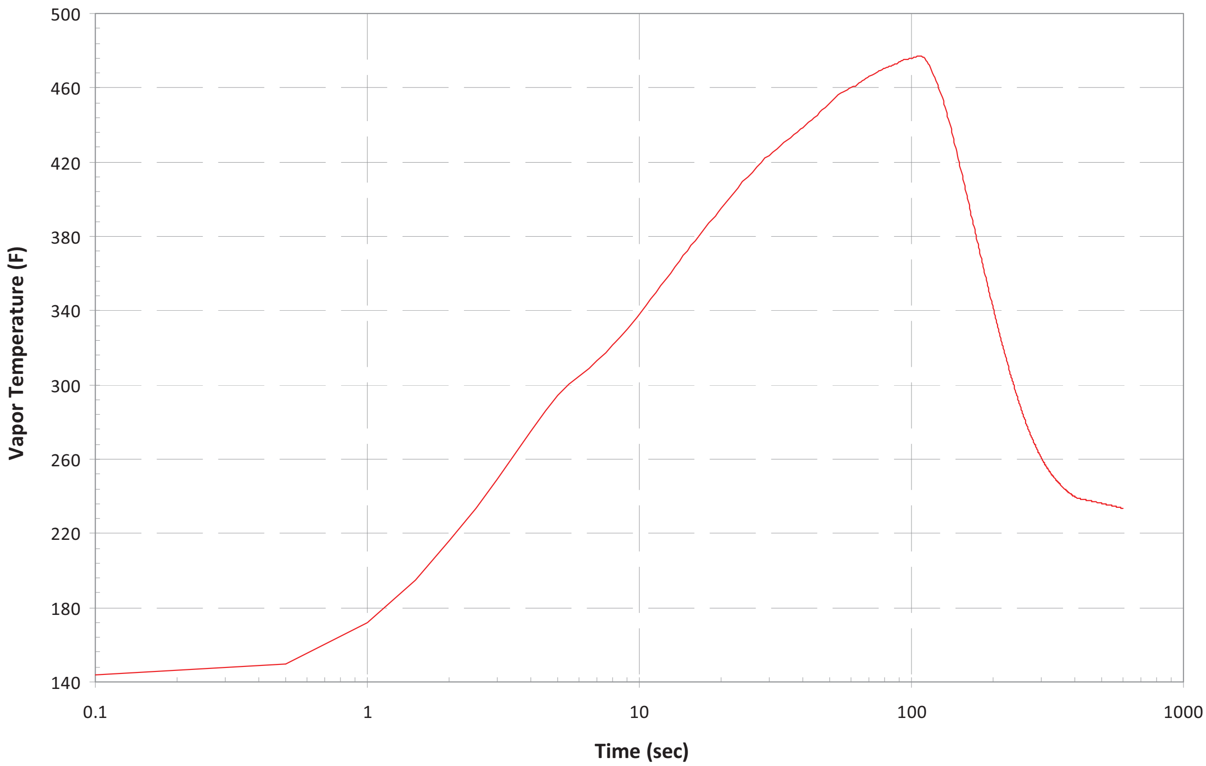


Figure 2.6.1-5: Limiting LOCA Long-Term Containment Pressure Response
(3.0 ft^2 CLPS DER, offsite power available, minimum ECCS)

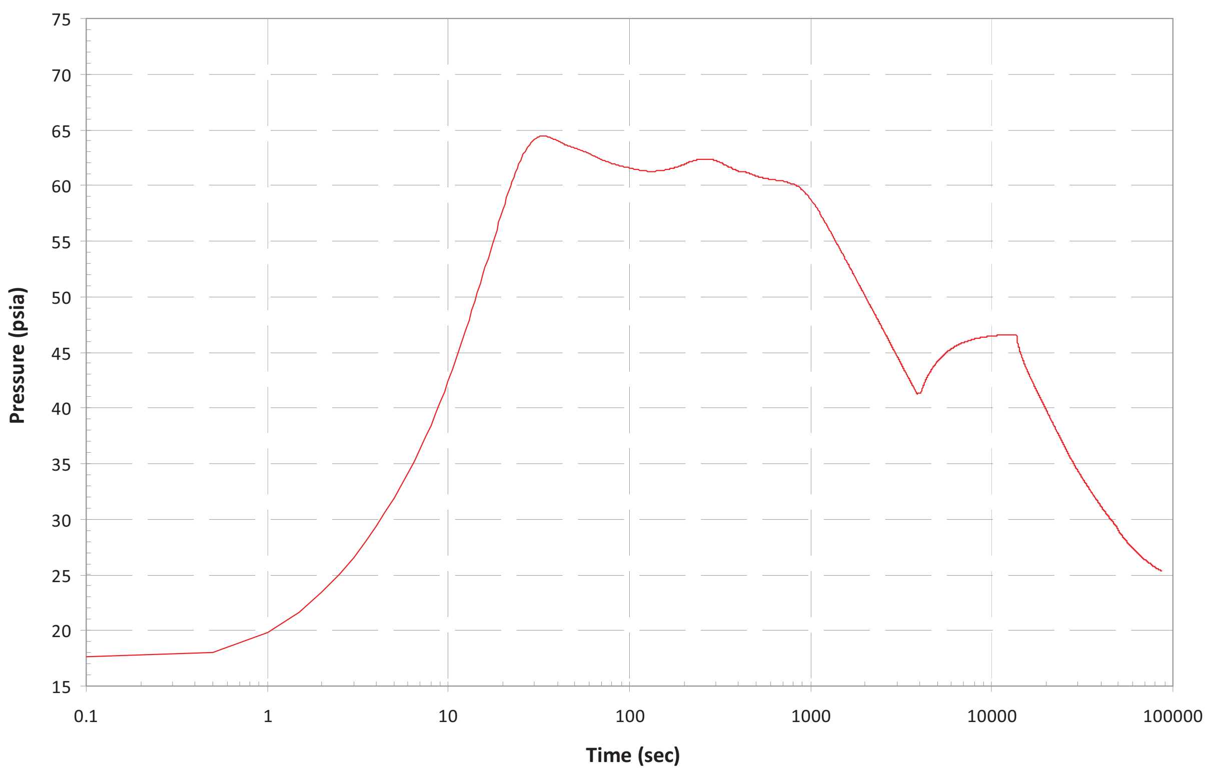


Figure 2.6.1-6: Limiting LOCA Long-Term Containment Vapor Temperature Response

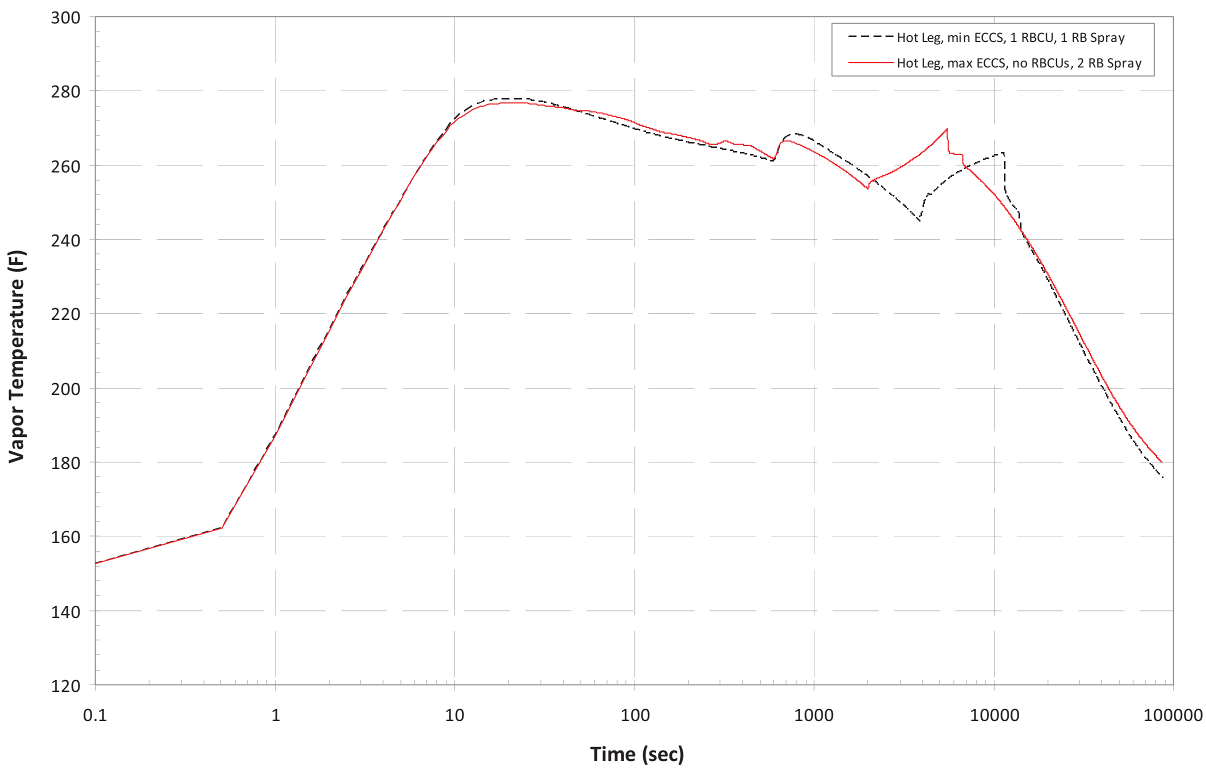


Figure 2.6.1-7: Limiting LOCA Long-Term Containment Sump Temperature Response
(Double-Ended Hot Leg Break, LOOP, max ECCS, no RBCUs, 2 RB Spray, min RB backpressure)

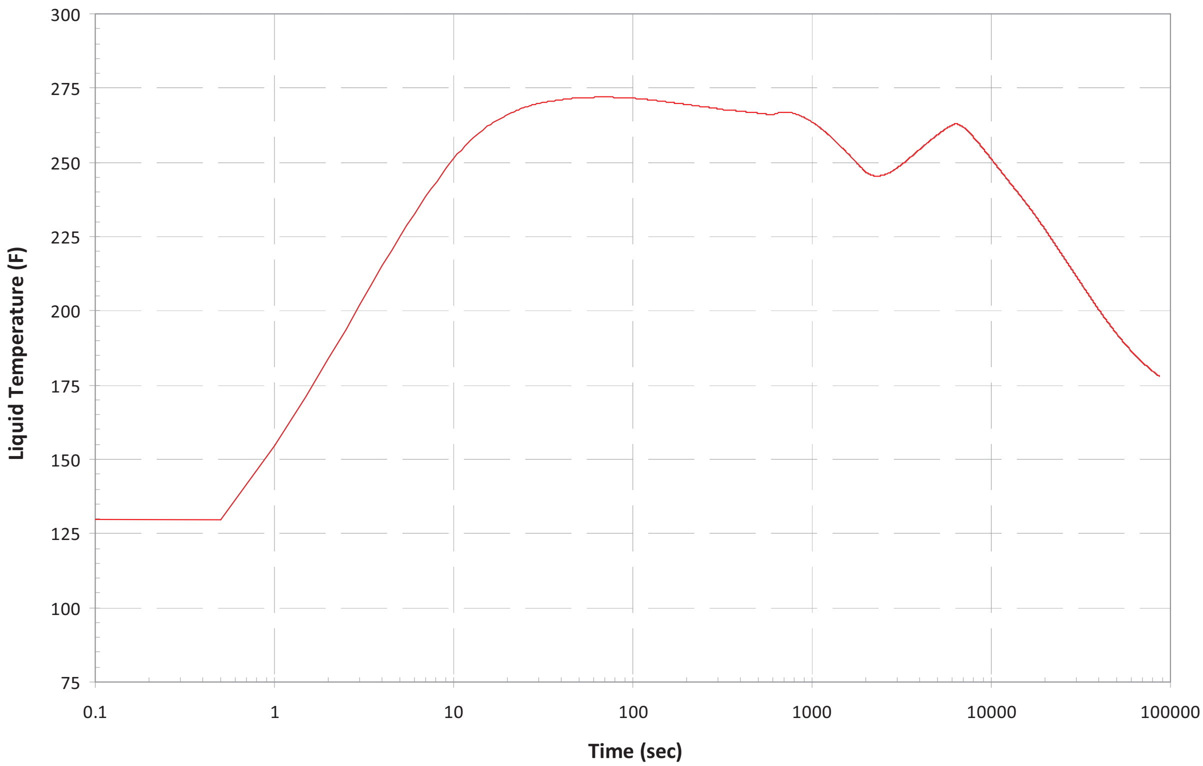
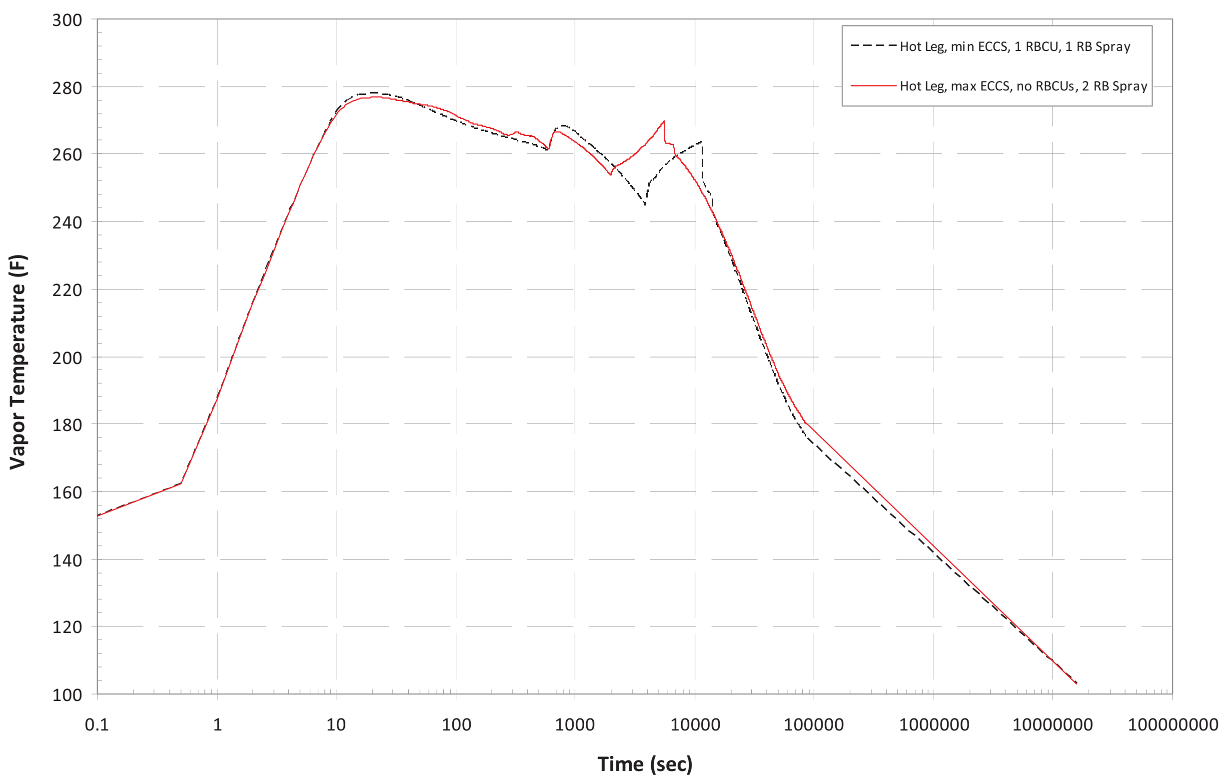


Figure 2.6.1-8: Limiting LOCA Long-Term Containment Vapor Temperature Response

Temperature Response Extended to 30 Days



2.6.2 Subcompartment Analyses

2.6.2.1 Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The CR-3 review for Subcompartment Analyses covered the determination of the design differential pressure values for containment subcompartments. The CR-3 review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization.

The NRC's acceptance criteria for Subcompartment Analyses are based on:

- GDC-4, insofar as it requires that SSCs important-to-safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects; and
- GDC-50, insofar as it requires that the containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, Missile Protection, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such SSCs be protected against dynamic effects. [GDC-4]; and
- FSAR Section 1.4.49, Containment Design Basis, insofar as it requires that the containment subcompartments be designed with sufficient margin to prevent fracture of the structure due to the calculated pressure differential conditions across the walls of the subcompartments. [GDC-50]

Leak-Before-Break (LBB) Criteria: In 1986, an exemption was granted to CR-3 based on “Leak-Before-Break” methodology described in B&W topical report BAW-1847, Rev. 1, “Leak-Before-Break Evaluation of Margin Against Full Break for RCS Primary Piping of B&W Designed NSS” (References 1, 2, and 3). As discussed in FSAR Section 14.2.2.5.11, topical report BAW-1847 modifies the structural design basis by eliminating dynamic effects of ruptures of large primary loop piping. Dynamic effects include missile generation, pipe whip, pipe break reaction forces, jet impingement forces, decompression waves within the ruptured pipe, and dynamic or non-static pressurization in cavities, subcompartments, and compartments that are not necessary for containment function.

2.6.2.2 Technical Evaluation

Introduction

In the original design analysis (i.e., prior to the approval of LBB methodology), the reactor was assumed to be operating at 102% of 2568 MWt at the time of the accident. Dynamic pressurization of the following subcompartments were analyzed: (a) Primary Cavity; (b) two Steam Generator Compartments; and (c) Primary Shield Pipe Penetration. In each case, an instantaneous conservative reactor coolant system (RCS) hot leg rupture with a discharge coefficient of 1.0 was considered. In each case, the analysis results confirmed that the acceptance criteria were met.

Description of Analyses and Evaluations

The current licensing basis no longer requires postulation of breaks in the RCS primary piping since CR-3 has been approved by the NRC for application of LBB methodology (References 1, 2, and 3). With the elimination of the large RCS primary pipe breaks for calculating dynamic effects, the only break locations that are considered are the largest remaining branch lines off of the primary loop piping. These branch lines are: (a) Pressurizer Surge Line, 10” Schedule 140 pipe; (b) Decay Heat Drop Line, 12” Schedule 160 pipe; and (c) Core Flood Line, 14” Schedule 140 pipe. The next largest line that is relevant for subcompartment analysis belongs to the Core Flood Line. The proposed EPU conditions increases RCS hot leg temperature (and hence, fluid energy content since pressure does not increase) only slightly (<1%) when compared to the original design analysis conditions. However, the cross-sectional area of the 14” pipe in the Core Flood Line is only a fraction (by a factor of 1/6) of the cross-sectional area of the 36” RCS pipeline that is considered for the original design analysis. Therefore, any dynamic effects of a Core Flood Line rupture associated with EPU conditions are judged not to exceed the previous results of the RCS hot leg rupture. As a result, the dynamic effects of the next largest pipeline rupture at EPU conditions are bounded by the dynamic effects of the RCS hot leg rupture at original power.

Results

This qualitative technical analysis demonstrates that the original Subcompartment Analyses remain bounding for the proposed EPU conditions.

2.6.2.3 Conclusion

CR-3 has reviewed the results from the original Subcompartment Analyses that remain bounding for the proposed EPU conditions. CR-3 concludes that the containment SSCs important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent failure of the structure, due to pressure difference across the walls following implementation of the proposed EPU. Based on this, CR-3 concludes that the plant will continue to meet the CR-3 current licensing basis with respect to the requirements of FSAR Sections 1.4.40 and 1.4.49 following implementation of the proposed EPU: Therefore, CR-3 finds the proposed EPU acceptable with respect to Subcompartment Analyses.

2.6.2.4 References

1. BAW-1847, Rev. 1, “Leak-Before-Break Evaluation of Margin Against Full Break for RCS Primary Piping of B&W Designed NSS.”
2. Safety Evaluation Report: Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, February 18, 1986.
3. License Amendment No. 89: NRC to CR-3 letter, dated May 23, 1986, “Crystal River Unit 3 - Amendment to Facility Operating License.”

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

2.6.3.1.1 Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment system, including sub-compartments and systems within containment. The CR-3 review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident.

The NRC's acceptance criteria for Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs) are based on:

- GDC-50, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained, and
- 10 CFR Part 50, Appendix K, insofar as it identifies energy sources during a LOCA.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates those general design criteria (GDC) provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are applicable CR-3 criteria:

- FSAR Section 1.4.49 Containment Design Basis, insofar as it requires that sufficient conservatism be provided in the mass and energy release analysis to assure that containment design margin is maintained, [GDC-50].

Additionally, FSAR Section 14.2.2.5 provides criteria to meet Appendix K to 10 CFR Part 50, insofar as it identifies energy sources during a LOCA.

2.6.3.1.2 Technical Evaluation

<u>Introduction</u>

The evaluation/generation of the design basis LOCA Mass & Energy (M&E) release data was completed to support the EPU operation. The M&E release rates described in this Section form the basis of further computations to evaluate the containment response following the postulated LOCA and to ensure that containment design margin is maintained.

The uncontrolled release of pressurized high-temperature reactor coolant, termed a LOCA, will result in the release of steam and water into the containment. This, in turn, will result in increases in the local sub-compartment pressures and an increase in the global containment pressure and temperature. Both short-term (peak pressure and temperature) and long-term (pressure and temperature at 24 hours) effects on the containment resulting from a postulated LOCA are supported by the M&E calculations summarized within this Section.

The LOCA M&E releases were analyzed for CR-3 until approximately the time at which the reactor vessel level has recovered to the RCS loop nozzle elevation. The M&E releases are inputs to the containment integrity analysis discussed in Section 2.6.1, Primary Containment Functional Design.

Input Parameters and Assumptions

Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. All input parameters are chosen to be consistent with accepted analysis methodology. Tables 2.6.3.1-1 through 2.6.3.1-3 present key data assumed in the analysis. Selected inputs and boundary conditions are also discussed below:

- The core-rated power of 3026.1 MWt, adjusted for calorimetric error (i.e., 100.4% of 3014 MWt), was used in the analysis.
- The analysis modeled the planned increase in Reactor Coolant System (RCS) operating temperatures that will accompany the EPU.
- The initial RCS pressure in this analysis was based on a nominal value of 2170 psia, as measured at the hot leg pressure tap. The RCS rapidly depressurizes from this value until the point where it equilibrates with containment pressure.
- The fuel characteristics for the long-term M&E release calculation were selected to maximize the energy stored in the fuel at the beginning of the postulated accident. Therefore, the analysis used a conservatively high initial fuel temperature.
- The RCS volume was increased by an appropriate allowance for thermal expansion. This assumption maximized the reactor coolant volume and fluid released.
- A steam generator tube plugging (SGTP) level of 0% was modeled. This assumption maximized the reactor coolant volume and fluid released by including the RCS fluid in all steam generator tubes. During the post-blowdown period, the steam generators are active heat sources since significant energy remains in the secondary metal and secondary fluid that has the potential to be transferred to the primary side. The 0% tube plugging assumption maximized heat transfer area and, therefore, the transfer of secondary heat across the steam generator tubes. Additionally, this assumption reduced the reactor coolant loop resistance, which reduced the ΔP across the steam generator primary side and increased break flow. Thus, the analysis conservatively modeled the effects related to SGTP.
- The core flood tanks (CFTs), which are passive means of Emergency Core Cooling System (ECCS) injection, are assumed to be available and injection flow begins when the RCS depressurizes below 653 psia. The CFT temperature was modeled with a conservatively high value of 130°F.
- With respect to pumped ECCS, the spectrum of cases run include consideration of both minimum safeguards (one high pressure injection (HPI) pump and one low pressure injection [(LPI) pump,

see Table 2.6.3.1-2), and maximum safeguards (two HPI pumps and two LPI pumps, see Table 2.6.3.1-3). Plant modifications for EPU impacted the HPI and LPI flow rate assumptions used in the analyses. The EPU modifications resulted in increased HPI flow rates as a function of RCS pressure, throughout the range of pressures considered. The LPI flow rates increased at lower RCS pressures that exist throughout the LBLOCA analysis. The higher ECCS flow rates do not impact the short term (peak containment pressure and vapor temperature) results, since ECCS flow is not established until after peak containment pressure and vapor temperature have been observed. In the long term, the time to sump transition is affected by the increase in ECCS flow. The limiting long-term containment analysis, documented in Section 2.6.1, models ECCS flow rates that bound ECCS flow rates in Tables 2.6.3.1-2 and 2.6.3.1-3. Higher ECCS flow is conservative for the long term analysis since it results in earlier transition to sump recirculation. The ECCS contributions to mass and energy releases for the long-term case presented in Section 2.6.1 are:

  - HPI flow is assumed to be 600 gpm per train, with two trains assumed available.
  - LPI flow is assumed to be 3600 gpm per train prior to sump recirculation and 2000 gpm per train after sump recirculation begins, with two trains assumed available.
  - Reactor building spray flow is assumed to be 1625 gpm prior to sump recirculation and 1375 gpm per train after sump recirculation begins, with two trains assumed available.
  - The ECCS flow rates above are applicable only to the long-term analysis, with maximum ECCS assumptions.

- The M&E analyses accounted for the impact of loss of offsite power.
- Minimum containment backpressure was determined via sensitivity studies to produce the most limiting M&E. A conservatively low backpressure relative to the containment response was thus applied to all cases.
- The decay heat levels assumed in the analyses are based on the American Nuclear Society (ANS) 1971 decay heat standard plus 20%. Conservatism was incorporated into the decay heat model by assuming infinite operating times, including actinides, and using uncertainties consistent with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling."

An analysis of the effects of the single-failure criterion has been included within the spectrum of breaks analyzed. The effects resulting from a loss of offsite power coincident with the pipe rupture are also evaluated. If offsite power is lost coincident with the pipe rupture, actuation of the emergency diesel generators (EDGs) is required to power the non-passive ECCS. The delay inherent with operating the EDGs is incorporated into the timing sequence for the delivery of the non-passive ECCS required to mitigate the transient. Should offsite power be available, it is assumed that the reactor coolant pumps (RCPs) are tripped within two minutes of receiving a loss of subcooling margin. The two-minute pump trip is conservative in comparison to the assumed values for the LOCA-PCT event.

The mass and energy release analyses model two ECCS configurations that allow for assessment of the effects of a single failure. The first configuration assumed minimum safeguards flow based on the postulated single failure of an EDG. This assumption results in the loss of one train of safeguards equipment. Thus the remaining train was conservatively modeled as: one HPI pump and one LPI pump. The other event scenario assumed maximum safeguards flow based on no postulated failures that could impact the amount of ECCS flow. The maximum safeguards flow was modeled as: two HPI pumps and two LPI pumps. The single failure assumption postulated is the failure of one containment spray pump, or loss of reactor building coolers. However, these single failures have no impact on the amount of ECCS flow and, therefore, no impact on the M&E release portion of the analysis. The analyses of the event scenarios described provided confidence that the effect of credible single failures is bounded.

Description of Analyses and Evaluations

The plant model used for the LOCA M&E release calculations is consistent with the guidelines described in BAW-10252-P-A (Reference 1). This methodology has been reviewed and approved by the NRC. The approval letter is included with (Reference 1). Consistent with the methodology of BAW-10252-P-A (Reference 1), the blowdown, refill, and reflood phases of the LOCA M&E analysis are performed using RELAP5/MOD2-B&W (Reference 3). Initial conditions and boundary conditions are chosen to maximize the stored energy in the primary and secondary systems, to maximize the energy removal from the fuel elements during the blowdown phase, and to minimize the refill period.

Discussed in this Section are the LOCA M&E releases for the hypothetical double-ended hot leg (DEHL) rupture near the steam generator with minimum safeguards and a delayed RCP trip. The scenario was determined to result in the highest peak containment and pressure and along with the remaining scenarios was well below half of the peak pressure within 24 hours (see Section 2.6.1, Primary Containment Functional Design, for a discussion of results).

The containment system receives M&E releases following a postulated rupture in the RCS. These releases continue over a time period, which, for the LOCA M&E analysis, is divided into four phases.

- Blowdown - the period of time from accident initiation (when the reactor is at steady-state operation) to the time that the RCS and containment reach an equilibrium state.
- Refill - the period of time when the lower plenum is being filled by the CFTs and ECCS water.
- Reflood - the period of time that begins when water from the lower plenum enters the core and ends when the core is completely quenched.
- Post-Reflood - the period of time following the reflood phase.

The RELAP5/MOD2-B&W code (Reference 3) was used for computing the blowdown, reflood, and refill phases of the LOCA transient. The code utilizes the control volume (element) approach with the capability for modeling a large variety of thermal fluid system configurations. The code also provides the means to incorporate user-specific calculations into the analysis. The methodology described in BAW-10192-P-A (Reference 2) was used to develop the LOCA M&E model for this code.

The LOCA M&E model integrates the details of the RCS, including the reactor vessel, steam generators, pressurizer, reactor coolant loops, and RCPs, with sufficient detail that additional transient phenomena such as a core flood tank (CFT) discharge, pumped ECCS fluid, RCP performance, and steam generator heat transfer are modeled explicitly. Control volumes are modeled with the nonequilibrium option, except for the core region and some steam generator secondary volumes, which are modeled as equilibrium volumes. A point kinetics solution is used with weighted feedback effects to model the fuel temperature coefficient and the moderator void coefficient. No credit for control rod insertion is taken until after the end of blowdown. Critical flow correlations for subcooled (Extended Henry-Fauske), two-phase (Moody), and superheated (Murdock-Bauman) break flow are incorporated into the analysis. Additional modifications are made to enhance the heat transfer from the fuel pin, to minimize the refill period, and to switch the core control volumes from equilibrium volumes to nonequilibrium volumes at the start of the reflood phase.

Break Size and Location

The analyses performed for the EPU included break size sensitivity studies that demonstrated the effect of postulated break size on the LOCA M&E releases. The double-ended guillotine (DEG) break was shown to be limiting due to larger mass flow rates during the blowdown phase of the transient. During the reflood and post-reflood phases, the break size has little effect on the releases.

Four distinct locations in the RCS loop are postulated for a pipe rupture for M&E release purposes:

- Hot leg (vessel outlet).
- Hot leg (near steam generator).
- Cold leg (between pump and vessel).
- Pump suction (between steam generator and pump).

All break locations were analyzed for the EPU. The most limiting break location with respect to the containment peak pressure response is in the hot leg near the steam generator (HLSG). The results for all analyzed cases included break M&E releases for the blowdown, refill, and reflood phases of the LOCA. A subset of these cases was chosen for further analysis of the long-term containment (24-hour) response, with the post-reflood phase addressed as discussed later in this section.

The DEG hot leg break results in the highest blowdown M&E release rates due to the size of the break. Although the core flooding rate would be the highest for breaks in the hot leg, the amount of energy released from the steam generator (SG) secondary side is minimal because the majority of the fluid that exits the core vents directly to containment, bypassing the steam generators. However, in the case of the DEG HLSG break, there is the potential for more rapid removal of the stored energy from the secondary side of the steam generator. For either hot leg break location, the reflood M&E releases are reduced as compared to either cold leg break location releases where a portion of the core exit mixture must pass through the steam generator before venting through the break.

The cold leg pump discharge (CLPD) break location was found to be less limiting in terms of the overall containment energy releases. The CLPD blowdown is faster than that of the cold leg pump suction (CLPS) break, and more mass is released into the containment. However, the core heat transfer is greatly reduced since liquid is lost to the break that would otherwise pass through the core, and the result is considerably lower energy release into containment. The analyses completed for the EPU determined that the transient for the CLPD is, in general, less limiting than that of the CLPS break. During reflood, the flooding rate is greatly reduced and the energy release rate into the containment is reduced. Therefore, the CLPD break is bounded by other breaks.

The CLPS break combines the effects of the relatively high core flooding rate, as in the hot-leg break, and the addition of the stored energy in the steam generators. For the containment peak pressure response the CLPS break location is not limiting, since the hot leg breaks produce higher blowdown M&Es. For the long-term containment (24-hour) response, the CLPS breaks may provide increased energy addition during the post-blowdown period, especially in the case where a liquid seal forms in the cold leg piping, resulting in steam generated in the core flowing through the steam generator, possibly becoming superheated before exiting the steam generator side of the break. Thus, a CLPS case was considered for the long-term analyses in addition to a subset of HLSG breaks.

Post-Reflood Calculations

Near the end of reflood, the analysis shifts to a long-term phase where the GOTHIC code is used for computing the post-reflood transient (Reference 1). The phenomena that are modeled include: 1) a nearly constant level in the core, 2) steam production from the core, and 3) the transfer of heat from the remaining heat sources in the primary and secondary systems. The primary fluid stored energy, primary system passive metal (including core metal) stored energy, the secondary system (fluid and metal) stored energy, and core decay heat are all dissipated conservatively. A single RCS node is placed in the model, and the initial conditions for that node are taken from the system model (the RELAP5/MOD2-B&W LOCA M&E model) including the liquid volume fraction and the averaged quantities of RCS liquid and vapor temperatures and RCS pressure. The stored energy of the RCS passive metal remaining at the transition time is conservatively modeled and released to the reactor vessel liquid or containment atmosphere based on the heat transfer rate at transition. The RCS metal energy is lumped based on exposure to either water or steam environments (for metal exposed to water, the energy is released to the reactor vessel and contributes to steaming rates, while for metal exposed to steam the energy is released directly to the containment atmosphere). For secondary stored energy, a conservative rate of heat transfer from the SG secondary to the primary side fluid is determined based on the change in calculated stored energies over a chosen time interval near the time of transition. The total sensible heat from the combined SG fluid and metal is assumed to be transferred until it is depleted. The final temperature of the secondary fluid and metal is ensured to be equal to or less than the saturation temperature at the 24-hour containment pressure. Finally, the long-term decay heat contribution is modeled in GOTHIC as a forcing function to a heater component assigned to the vessel node.

Sources of M&E

The sources of mass considered in the LOCA M&E release analysis are:

- RCS water.
- Core flood tank water (both inject).
- Pumped ECCS injection (HPI and LPI).

The energy inventories considered in the LOCA M&E release analysis are:

- RCS water.

- Core flood tank water (both inject).
- Pumped ECCS injection (HPI and LPI).
- Decay heat.
- Core-stored energy.
- RCS metal (includes steam generator tubes).
- Steam generator secondary energy (includes fluid mass and steam mass).

The energy release from the zirc-water reaction is considered as part of the BAW-10192-P-A (Reference 2) methodology. Based on the way that the energy in the fuel is conservatively released to the vessel fluid, the fuel cladding temperature does not increase to the point where the zirc-water reaction is significant. This is in contrast to the 10CFR50.46 analyses, which are biased to calculate high fuel rod cladding temperatures and therefore a significant zirc-water reaction. For the LOCA M&E calculation, the energy created by the zirc-water reaction value is small, none-the-less it is included in the LOCA M&E release.

The analysis used the following energy reference points to calculate the stored energy release from the RCS fluid and passive metal. The initial condition for the stored energy calculation is that which exists at the end of the reflood period.

- Available energy: 32°F, 14.7 psia (energy available that could be released).
- Total energy content: 32°F; 14.7 psia (total internal energy of the RCS).

A representative sequence of events for the LOCA transient is shown in Table 2.6.3.1-4. This sequence of events is typical of the double-ended guillotine break of the hot leg near the steam generator.

Results

The integrated M&E release data for a limiting case is shown in Figures 2.6.3.1-1 and 2.6.3.1-2. The LOCA M&E releases from accident initiation to near the time at which the reactor vessel is refilled (~1000 seconds) have been provided for a DEG HLSG break. These M&E results are the basis for the containment response analyses reported in Section 2.6.1, Primary Containment Functional Design.

The consideration of the various energy sources for the long-term M&E release analysis provides assurance that all available sources of energy have been included in this analysis.

2.6.3.1.3 Conclusion

CR-3 has reviewed the mass and energy release assessment and concludes that the effects of the proposed EPU have been adequately addressed and that the M&E release assessment appropriately accounts for the sources of energy identified in 10CFR50, Appendix K. Based on this, CR-3 finds that the M&E release analysis meets the requirements of CR-3 FSAR design criterion 1.4.49 and FSAR Section 14.2.2.5. Hence, CR-3 finds the proposed EPU acceptable with respect to Mass & Energy Release Analysis for Postulated Loss-Of-Coolant Accident.

2.6.3.1.4 References

1. AREVA NP Topical Report BAW-10252(P)-A, Revision 0, September 2005 (Proprietary) and BAW-10252(NP)-A, Revision 0, September 2005 (Nonproprietary), "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC."
2. AREVA NP Topical Report BAW-10192P-A, Revision 0, June 1998 (Proprietary) and BAW-10192-A, Revision 0, June 1998 (Nonproprietary), "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants."
3. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary) and BAW-10164NP-A, Revision 6, June 2007 (Nonproprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

Table 2.6.3.1-1 System Parameters Initial Conditions

Parameter	Value
RCS Initial Conditions	
Rated Core Power, MWt	3014
Core Power Uncertainty, %	0.4
RCP Power, MWt	16.16
RCS Average Temperature, °F	582 nominal
RCS Hot Leg Pressure, psia	2170
RCS Flow Rate, gpm	374,880
Indicated PZR Level, in	290 on 320 inch scale[1]
MFW Temperature, °F	460
Turbine Header Pressure, psia	930 nominal
SG Tube Plugging, %/SG	0
Decay Heat Parameters	
Decay Heat Standard	ANS 1971 + 20%
Actinides	Heavy
RCP Parameters	
Pump Manufacturer	Byron-Jackson
RCP Type, Single Phase Head Difference	RELAP5
Two-Phase Full-Degraded Head Difference	M3-Modified
RC Pump Trip, s	On LOOP (time of break) If OP available: 2 minutes after loss of subcooling margin [2]
ECCS Parameters	
BWST Temperature, °F	120
CFT Liquid Volume, ft^3/tank	1070 (8005 gallons)
CFT Cover Gas Pressure, psia	653
CFT Surge Line Area, ft^2	0.7213
CFT Surge Line Length, ft	Line A: 84.26 Line B: 97.18
CFT Liquid Temperature, °F	130
CFT Line Resistance (form+friction)	6.84
CFT Line Elevation Change, ft	-6.16

1. The most limiting subset of analyses considered initial pressurizer level at 290 inches, for evaluation of short term containment peak pressure and temperature.
2. The time assumed for operator action to trip the RCPs conservatively bounds the RCP trip time assumed for the LOCA-PCT analyses.

Table 2.6.3.1-2 HPI Flow Rates Minimum HPI Flow CLPD Break

Pressure (psia)	EPU Pre-HPI Mod		EPU Post-HPI Mod [1]	
	Broken Cold Leg Flow (gpm)	Intact Cold Leg Flow (gpm)	Broken Cold Leg Flow (gpm)	Intact Cold Leg Flow (gpm)
15	135.7	332.4	127.3	374.9
615	121.9	298.5		
915	114.1	279.5	108.0	318.0
1215	105.6	258.7		
1515	96.1	235.6		
1815	85.4	209.2	82.2	241.9
2115	72.8	178.1	70.5	207.5
2415	56.7	139.0		
2615			39.6	116.6

Maximum HPI Flow CLPD Break

Pressure (psia)	EPU Pre-HPI Mod		EPU Post-HPI Mod [1]	
	Broken Cold Leg Flow (gpm)	Intact Cold Leg Flow (gpm)	Broken Cold Leg Flow (gpm)	Intact Cold Leg Flow (gpm)
15	215.4	527.2	228.8	675.5
615	192.9	472.4	206.4	609.5
1215	166.9	408.5	180.2	531.9
2515	80.6	197.4	94.4	278.9

Table 2.6.3.1-2 (continued) HPI Flow Rates

Minimum HPI Flow For All Breaks Except CLPD

Pressure (psia)	EPU Pre-HPI Mod	EPU Post-HPI Mod [1]
	Total Flow to RCS (gpm)	Total Flow to RCS (gpm)
15	468.1	502.2
615	420.4	
915	393.6	426.0
1215	364.3	
1515	331.7	
1815	294.6	324.1
2115	250.9	278.0
2415	195.7	
2615		156.2

Maximum HPI Flow For All Breaks Except CLPD

Pressure (psia)	EPU Pre-HPI Mod	EPU Post-HPI Mod [1]
	Total Flow to RCS (gpm)	Total Flow to RCS (gpm)
15	742.6	904.3
615	665.3	815.9
1215	575.4	712.1
2515	278.0	373.3

1 Refer to Appendix E for discussion of the High Pressure Injection (HPI) System modification. The Pre-HPI Mod flows are utilized in the analyses. The Post-HPI Mod flows are provided for information, and reflect modifications in conjunction with EPU. Also, see discussion under "Input Parameters and Assumptions", which describes the conservative ECCS flow rate assumptions applied to the long-term containment response analysis.

Table 2.6.3.1-3 LPI Flow Rates

Minimum LPI Flow

Pressure (psia)	Total Flow to RCS Pre-LPI Mod (gpm)	Pressure (psia)	Total Flow to RCS Post-LPI Mod[1] (gpm)
0	2685		
		15.0	2886
		101.0	2886
		117.0	2684
124	2685	124.0	2581
180	1000		
190	675		
195	0		

Maximum LPI Flow

Pressure (psia)	Total Flow to RCS Assumed Pre-LPI Mod (gpm)	Pressure (psia)	Total Flow to RCS Post-LPI Mod[1] (gpm)
0	5370		
		15.0	6000
		101.0	4452
124	5370		
180	2000		
190	1350		
195	0		

1. Refer to Appendix E for discussion of the Low Pressure Injection (LPI) System modification. The Pre-LPI Mod flows are utilized in the analyses. The Post-LPI Mod flows are provided for information, and reflect modifications in conjunction with EPU. Also, see discussion under "Input Parameters and Assumptions", which describes the conservative ECCS flow rate assumptions applied to the long-term containment response analysis.

Table 2.6.3.1-4 DEG HLSG Break Sequence of Events

Time (sec)	Event Description
0.0	Break occurs, Reactor Trip assumed.
0.12	ESAS Low RCS Pressure Setpoint (1640 psia) reached
2.00	MFW Coastdown begins
8.41	CFT-1 Injection starts
8.58	CFT-2 Injection starts
9.62	ESAS Low-Low RCS Pressure Setpoint (515 psia) reached
14.00	MFW terminated
17.86	End of Blowdown Phase
35.12	LPI Flow begins
40.22	CFT-1 Empty
42.31	CFT-2 Empty
67.12	HPI Flow begins
125.71	RCPs tripped on Loss of Subcooling Margin
252.64	Core quenched
600.0	RELAP5/MOD2-B&W analysis terminated

Figure 2.6.3.1-1 Integrated Mass Release for Hot Leg LBLOCA

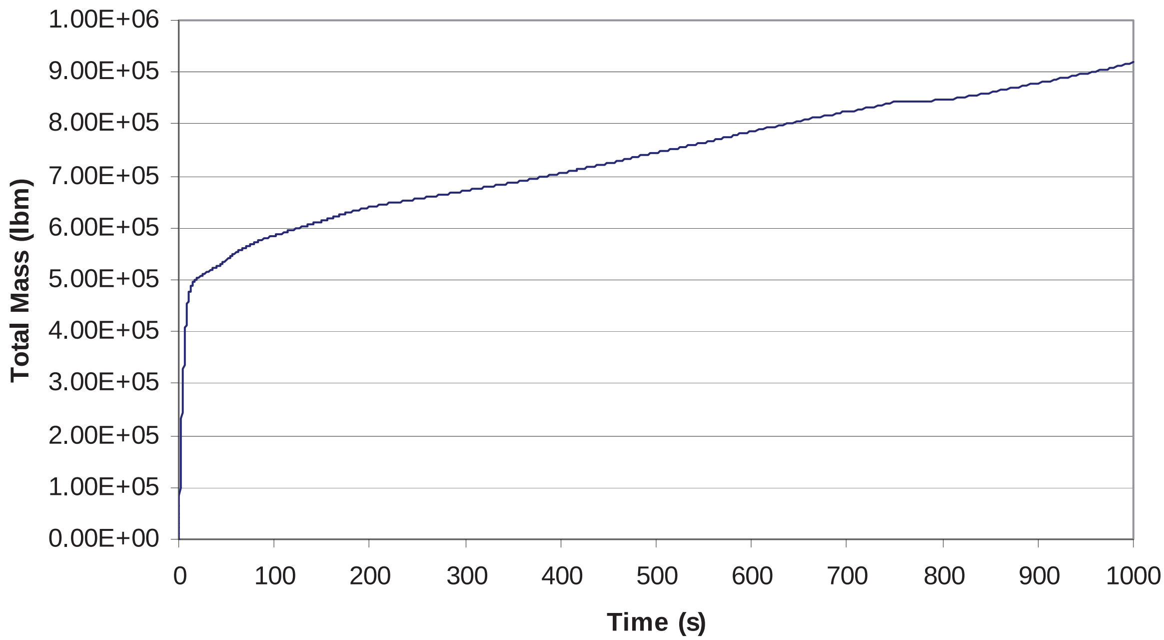
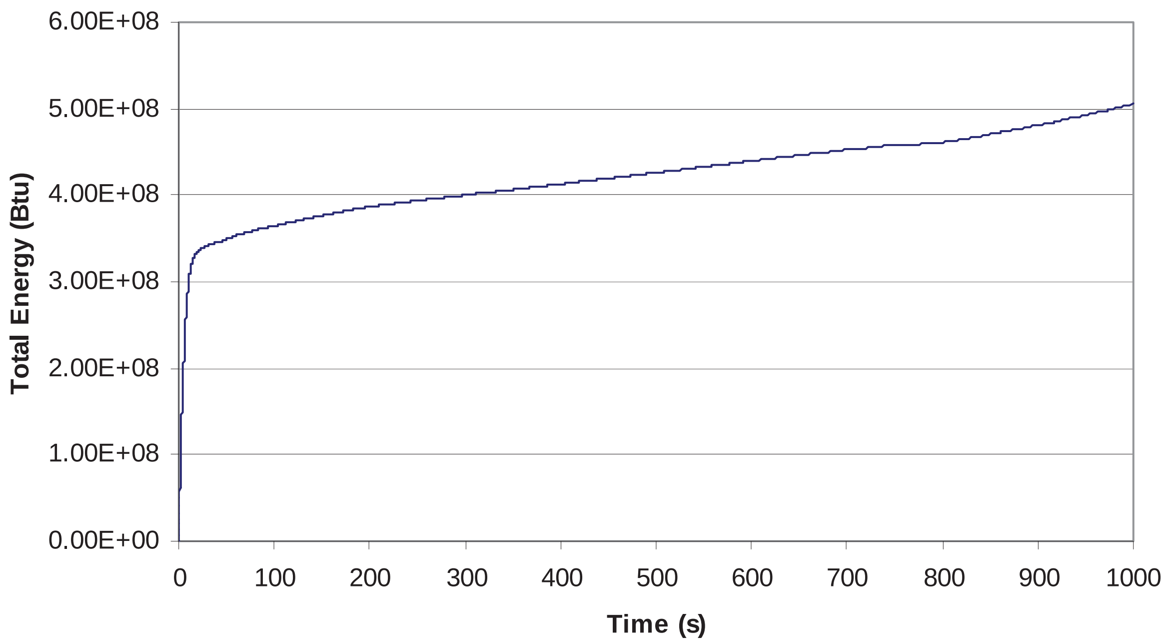


Figure 2.6.3.1-2 Integrated Energy Release for Hot Leg LBLOCA



2.6.3.2 Mass & Energy Release Analysis for Secondary System Pipe Ruptures

2.6.3.2.1 Regulatory Evaluation

The CR-3 review covered the energy sources that are available for release to the containment, the mass and energy release rate calculations, and the single-failure analyses performed for steam and feedwater line isolation provisions, which would limit the flow of steam or feedwater to the assumed pipe rupture.

The NRC's acceptance criteria for Mass and Energy Release Analysis for Secondary System Pipe Ruptures are based on:

- GDC-50, insofar as it requires that the margin in the design of the containment structure reflect consideration of the effects of potential energy sources that have not been included in the determination of peak conditions, the experience and experimental data available for defining accident phenomena and containment response, and the conservatism of the model and the values of input parameters.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates those GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are applicable CR-3 criteria:

- FSAR, Section 1.4.49, Containment Design Basis, insofar as it requires that the margin in the design of the containment structure reflects consideration of the effects of potential energy sources that have not been included in the determination of peak conditions, the experience and experimental data available for defining accident phenomena and containment response, and the conservatism of the model and the values of input parameters. [GDC-50]

2.6.3.2.2 Technical Evaluation

Introduction

Steamline ruptures occurring inside the reactor containment structure may result in significant releases of high-energy fluid to the containment environment, producing elevated containment temperatures and pressures. The magnitude of the releases following a steamline rupture is dependent upon the plant initial operating conditions and the size of the rupture as well as the configuration of the plant steam system and the containment design. The analysis considers credible single failures within the postulated accident scenario in order to determine the worst case for containment response following a steamline break. The limiting case is initiated from hot full power (HFP), since both power level and steam generator inventory are maximized for the B&W plants at HFP.

Input Assumptions, Parameters, and Acceptance Criteria

Where appropriate, bounding inputs are utilized and instrumentation uncertainties are included. All input parameters are chosen to be consistent with accepted analysis methodology. Some of the most critical items are related to Reactor Coolant System (RCS) initial conditions, Once-Through-Steam-Generator (OTSG) inventory, and Main Feedwater (MFW) delivery prior to isolation. Selected inputs and boundary conditions are discussed below:

- The nominal reactor power level is 3014 MWt. In addition, the initial power level assumed in the analysis includes consideration of heat-balance error, for a total core power of 3026.1 MWt. Initiating the event from HFP maximizes the energy addition to the RCS and also conservatively maximizes the initial inventory that ultimately is released through the break.
- The RCS average temperature is 582°F, reflecting an increase in RCS operating temperature in conjunction with the EPU.
- Continued operation of the reactor coolant pumps (RCPs) contributes heat to the RCS and maintains a high heat transfer rate to the steam generators.
- The model includes consideration of the heat that is stored in the RCS metal.
- The capability for reverse heat transfer from the intact steam generator to the RCS is modeled.
- Core residual heat generation is assumed based on the 1971 ANS Decay Heat Standard, and includes the contribution of heavy actinides. The decay heat model chosen bounds the 1979 ANS decay heat plus 2σ model (Reference 2).
- Conservative core reactivity coefficients corresponding to end-of-cycle conditions with the most reactive rod stuck out of the core are assumed. This maximizes the reactivity feedback effects as the RCS cools down as a result of the steamline break.
- The Reactor Protection System (RPS) initiates reactor trip on either high reactor flux or low RCS pressure. The modeling of the high flux trip includes an error term to account for the transient-induced flux error caused by decreasing downcomer temperatures. Initiation of RPS due to high containment building pressure is not credited for the analysis.
- A minimum shutdown margin of 1.3%Δk/k at Hot Zero Power (HZP) is assumed in accordance with requirements for power operation (Modes 1 and 2).
- The Engineered Safeguards Actuation System (ESAS) initiates high pressure injection flow on a low RCS pressure signal.
- Minimum high pressure injection flow rates corresponding to the failure of one safety injection train have been assumed for all cases in this analysis. The flow rates are modeled to conservatively minimize the amount of cooling provided by high pressure injection, and to minimize the amount of boron delivered to the RCS.
- No steam generator tube plugging is assumed to maximize the primary-to-secondary heat transfer rate.

- The Emergency Feedwater Initiation and Control (EFIC) low steamline pressure setpoint is modeled and initiates start of Emergency Feedwater (EFW) pumps and closure of all feedwater isolation valves.

- Emergency feedwater is actuated due to low steam line pressure. However, the CR-3 plant design includes logic that prohibits emergency feedwater delivery to the faulted steam generator. Therefore, emergency feedwater does not contribute to the mass and energy released to the containment building.

- The system model accounts for closure of MFW isolation valves within 31 seconds from receipt of the EFIC signal. This includes appropriate conservatisms for signal processing times. The valve closure time assumed bounds the EPU configuration of the CR-3 plant, since faster-acting isolation valves will be implemented in association with the EPU (refer to Appendix E, Major Plant Modifications). The isolation valves to be installed for the EPU will be capable of closure in less than 20 seconds after receipt of the EFIC signal. The change to the isolation time will result in significantly less feedwater added to the steam generator and will consequently result in a reduction in mass and energy releases. Other MFW System changes are being implemented for the EPU, including modifications to the MFW and MFW booster pumps to meet the increased flow requirement for the EPU while maintaining adequate design margin. Thus, the faster-acting isolation valves are being installed to offset the other MFW System changes and the increase in steady-state flow required for the EPU. Analyses confirmed that the reduction in feedwater addition due to faster isolation (within 20 seconds) more than offsets the other MFW changes. The analyses further confirmed that the change to faster acting isolation valves does not alter the limiting single failure considered for the event (failure of the affected loop MFW pump to trip). As a result, the mass and energy generated with the 31-second closure time is bounding and conservative with respect to the planned EPU configuration. The mass and energy results herein, as depicted by Figures 2.6.3.2-1 and 2.6.3.2-2 reflect the longer isolation time, since the results have been demonstrated to be conservative and bounding relative to the EPU plant configuration.

- A maximum initial steam generator mass in the faulted loop steam generator was assumed. The analysis is initiated at HFP which maximizes secondary inventory for the B&W plants and the OTSG. In addition, a degree of secondary fouling is postulated which results in inventory levels that approach the upper limit for operate-range level (96% OR). The use of a high faulted loop initial steam generator mass maximizes the steam generator inventory available for release to containment.

- The initial steam in the steamline between the break and the steamline non-return check valve is included in the mass and energy released from the break.

- The break effluent calculation accounts for any steam superheat generated during blowdown. If liquid is present in the break effluent due to swell of steam generator two phase mixture, the AREVA methodology for containment building response specifies that the mass and energy releases be adjusted to saturated conditions when input to the containment analysis.

- A conservatively low containment backpressure is assumed throughout the M&E analysis.

Upon receipt of the feedwater isolation signal on low steam line pressure, the main feedwater pumps are sent a signal to trip. The most limiting single failure that is analyzed for this event is a failure of the main feedwater pump to trip after the isolation signal has been generated. This results in continued delivery of forced main feedwater flow to the faulted steam generator until the main feedwater isolation valves are closed. This failure consideration is most limiting due to the relatively long isolation time of the feedwater isolation valves.

Acceptance Criteria

The main steamline break (MSLB) is classified as an ANS Condition IV event. The acceptance criteria associated with the steamline break event resulting in mass and energy releases inside containment is based on an analysis that provides sufficient conservatism to show that the containment design margin is maintained. The specific criteria applicable to this analysis are related to the assumptions regarding power level, stored energy, the break flow, main feedwater flow, steamline and feedwater isolation, and single failure assumptions that have been included in this steamline break mass and energy release analysis.

Description of Analyses and Evaluations

The MSLB mass and energy (M&E) release calculation is performed according to the AREVA methodology for generation of containment building releases and containment response (Reference 3). The RELAP5/MOD2 B&W computer code (Reference 4), which has received full certification, is used to calculate the system and core responses. The double ended guillotine (DEG) break was analyzed. The DEG break produces a maximum rate of release to the containment building, and also the most limiting reactivity response due to the overcooling, which ultimately contributes more energy to be removed via the steam generators. The code uses a control volume approach to solve the time-dependent conservation equations for mass, energy and momentum over the steam and liquid phases.

Following the initiation of the steamline break, the feedwater flow increases due to a lower backpressure on the feedwater pump as a result of the depressurizing steam generator. This maximizes the total mass addition prior to feedwater isolation. The feedwater isolation response time, following EFIC signal, is assumed to be a total of 31 seconds, accounting for delays associated with signal processing plus feedwater isolation valve stroke time. The limiting mass and energy release scenario considers a single failure of the MFW to trip upon receipt of an isolation signal, which allows additional feedwater inventory into the faulted steam generator prior to isolation valve closure.

Following feedwater isolation, as the steam generator pressure decreases, some of the fluid in the feedwater lines downstream of the isolation or regulator and bypass valves may flash to steam if the feedwater temperature exceeds the saturation temperature. This unisolable feedwater line volume is an additional source of fluid that can increase the mass discharged out of the break.

The limiting MSLB case considers the largest possible break, a DEG break in the 24-in section of the steam line inside containment. A DEG is defined as a rupture in which the steam pipe is completely severed and the ends of the break fully displace from each other. This break conservatively bounds the plant response to any smaller break size. The OTSGs installed at CR-3 do not contain flow-limiting restrictors in the steam outlet nozzles. There is a small degree of flow restriction at the outlet nozzles which is accounted for within the RELAP5/MOD2-B&W model. The forward break area corresponds to the cross-sectional area of the 24-in main steam line piping, as does the break area for flow in the reverse direction.

Results

The limiting steamline break case was analyzed with the assumed initial power level and single failure. The M&E release from the break was calculated using the RELAP5/MOD2-B&W code. The analysis included the effects of the extended power uprate to 3014 MWt, and an increase in the shutdown margin to 1.3%Δk/k at HZP for power operation.

The limiting MSLB containment pressure case for the EPU (see Section 2.6.1, Primary Containment Functional Design) is a large double-ended rupture steamline break initiated from full power with a single failure of the affected-side main feedwater pump to trip. The sequence of events for this case is given in Table 2.6.3.2-1, while the M&E releases are depicted in Figures 2.6.3.2-1 and 2.6.3.2-2.

The limiting MSLB containment pressure case for EPU causes the MFW, MSL and MSIV valves on Loop B to close and activates the control rods insertion. Shortly after, ESAS low RCS pressure setpoint is reached followed by MFW, MSL and MSIV valves on Loop A also close. The event is terminated after the minimum RCS pressure and maximum post-trip total core power are reached, complete feedwater / steam lines isolation is achieved, the HPI injection begins and the pressurizer indicated level is back on scale.

The steamline break event resulting in mass and energy releases inside containment has been performed with sufficient conservatism to show that the containment design margin is maintained. Therefore, the acceptance criterion for the event has been met.

2.6.3.2.3 Conclusion

CR-3 has reviewed the mass and energy release assessment for the postulated secondary system pipe ruptures and finds that the analysis adequately addresses the effects of the proposed EPU. Based on this, CR-3 concludes that the analysis meets the CR-3 current licensing basis requirements with respect to CR-3 FSAR Criterion 1.4.49 for ensuring that the analysis is conservative (i.e., that the analysis includes sufficient margin). Therefore, CR-3 finds the proposed EPU acceptable with respect to Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures.

2.6.3.2.4 References

1. AREVA NP Topical Report BAW-10193P-A, Revision 0, January 2000 (Proprietary) and BAW-10193NP-A, Revision 0, January 2000 (Nonproprietary), "RELAP5/MOD2-B&W for Safety Analysis of B&W Designed Pressurized Water Reactors."

2. AREVA NP Topical Report BAW-10192P-A, Revision 0, June 1998 (Proprietary) and BAW-10192-A, Revision 0, June 1998 (Nonproprietary), "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants."

3. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary) and BAW-10164NP-A, Revision 6, June 2007 (Nonproprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

Table 2.6.3.2-1: Sequence of Events and Key Results for MER Case

Event	Time (sec)
Break Initiated	0.00
EFIC Setpoint Reached for Low SG Pressure, Loop B	0.19
MFW and MSL Isolation Valves begin to close in Loop B	1.19
MSIVs Closed on Loop B	6.19
RPS High Flux Trip Setpoint Reached	6.44
Control Rods Begin to Insert	6.87
ESAS Low RCS Pressure Setpoint Reached	12.12
Indicated PZR Level Off-Scale Low	17.00
EFIC Setpoint Reached for Low SG Pressure, Loop A	19.91
MFW and MSL Isolation Valves begin to close in Loop A	20.91
Peak MFW Flow to Faulted SG (2446.2 lbm/sec)	22.50
MSIVs Closed on Loop A	25.91
Full Closure of MFPSV, MFIV, and SUBV, Loop B	31.23
Minimum Cold Leg Temperature Reached (454.94°F)	47.50
Minimum Post-Trip Reactivity (-61.6632×10^{-5} $\Delta k/k$)	50.50
Full Closure of MFPSV, MFIV, and SUBV, Loop A	50.91
Minimum RCS Pressure (at hot leg tap) (851.51 psia)	51.00
Maximum Post-Trip Total Core Power (804.2 MWt)	51.00

Event	Time (sec)
Full Closure of MFLLB, Loop B	67.20
HPI Flow Begins	79.12
Full Closure of MFLLB, Loop A	86.91
Indicated PZR Level Back On-Scale	270.00
End of Transient	600.00

Figure 2.6.3.2-1 Integrated Mass Release for Main Steam Line Break

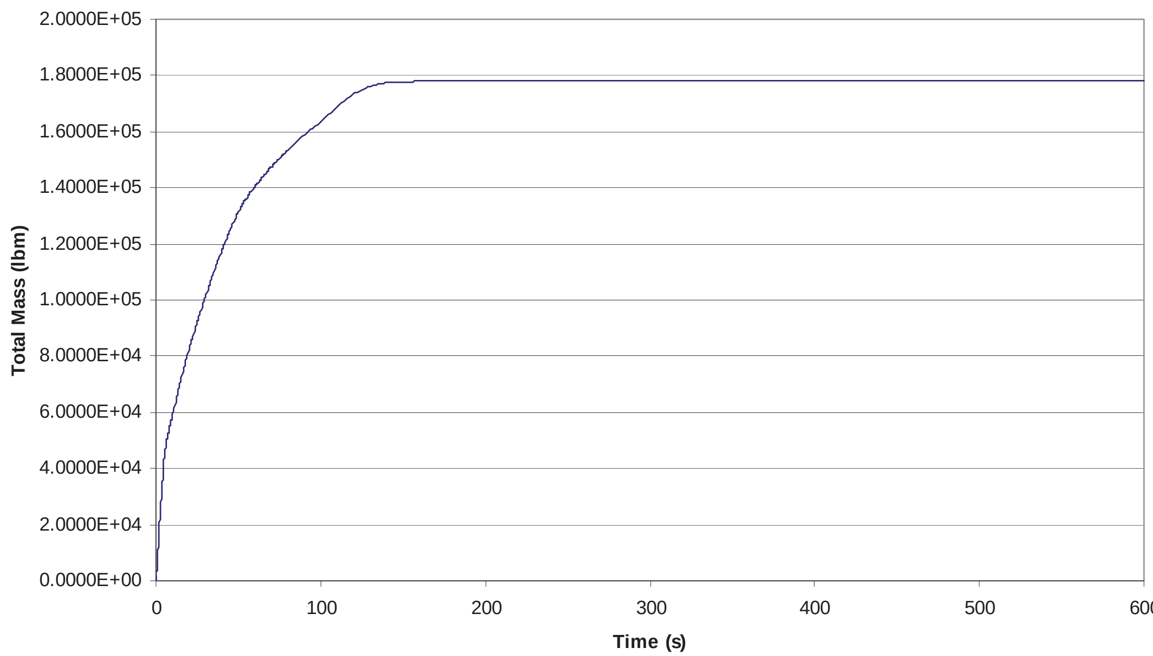
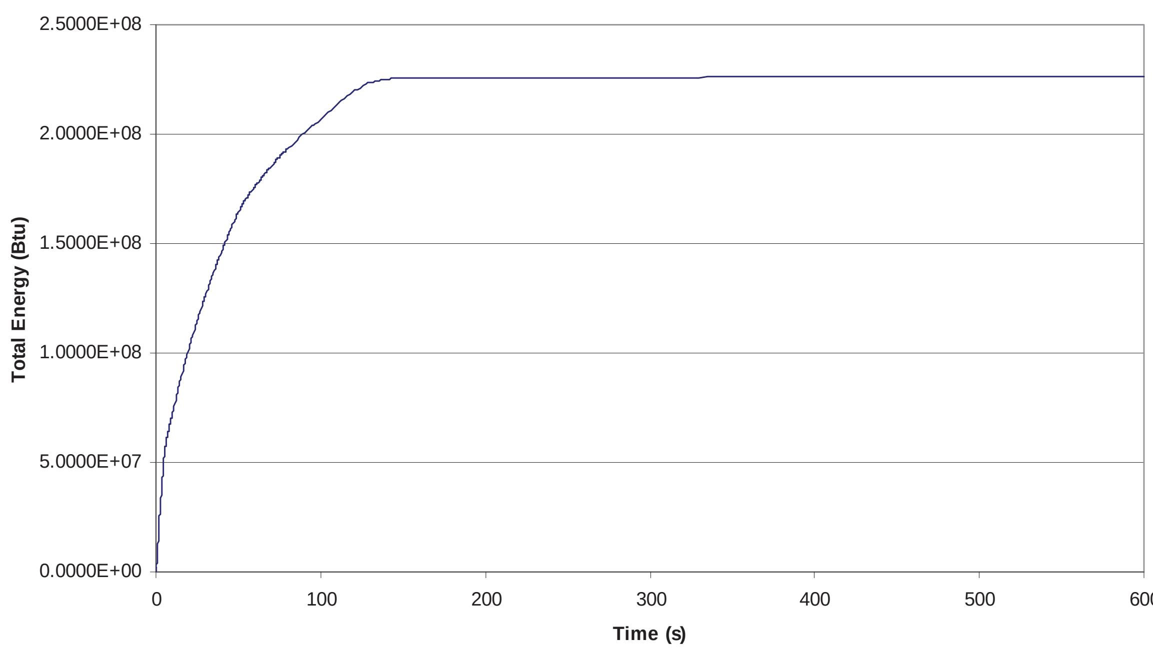


Figure 2.6.3.2-2 Integrated Energy Release for Main Steam Line Break



2.6.4 Combustible Gas Control in Containment

2.6.4.1 Regulatory Evaluation

Following a Loss-Of-Coolant Accident (LOCA), hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The CR-3 review covered (1) the production and accumulation of combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The CR-3 review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated.

The NRC's acceptance criteria for Combustible Gas Control in Containment are based on:

- GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions;
- GDC-41, insofar as it requires that systems be provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained;
- GDC-42, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic inspection;
- GDC-43, insofar as it requires that systems required by GDC-41 be designed to permit appropriate periodic testing; and
- 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The NRC amended 10 CFR 50.44 in September 2003 to eliminate certain requirements for hydrogen recombiners and hydrogen purge systems, and relaxed the requirements for hydrogen and oxygen monitoring equipment to make them commensurate with risk significance. CR-3 adopted the provisions of the amended rule with a license amendment (CR-3 License Amendment No. 216, issued on April 5, 2005) and made commitments to maintain the hydrogen monitoring systems capable of diagnosing beyond design basis accidents. The design basis of hydrogen monitoring system is presented in FSAR Sections 7.3.5 and 9.11. The instrumentation and displays comply with Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

2.6.4.2 Technical Evaluation

Introduction

10 CFR 50.44 was revised in September 2003 and no longer defines a design basis LOCA hydrogen release and eliminates the requirements for hydrogen control systems to mitigate such releases.

Description of Analyses and Evaluations

CR-3 has adopted the revised rule as addressed by License Amendment No. 216, issued on April 5, 2005, which eliminated the requirements for hydrogen monitors and made commitments to maintain the hydrogen monitoring systems capable of diagnosing beyond design basis accidents. CR-3 has determined that the proposed EPU has no effect on the design of these systems or on the ability of these systems to perform their intended functions.

Results

Based on the NRC-approved changes and the low safety significance of post-LOCA combustible gas generation in large, dry pressurized water reactor containment buildings, the proposed EPU has no effect on the regulatory requirements for Combustible Gas Control in Containment.

2.6.4.3 Conclusion

The CR-3 review primarily focused on the impact of the proposed EPU on the regulatory requirements for combustible gas control in containment and concludes that the plant will continue to have sufficient capabilities, consistent with the requirements in 10 CFR 50.44 as discussed above. Therefore, CR-3 finds the proposed EPU acceptable with respect to Combustible Gas Control in Containment.

2.6.4.4 References

None

2.6.5 Containment Heat Removal

2.6.5.1 Regulatory Evaluation

Fan cooler systems, spray systems, and Residual Heat Removal (RHR) Systems are provided to remove heat from the containment atmosphere and from the water in the containment sump. The CR-3 review in this area focused on (1) the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH) to the Containment Heat Removal System pumps and (2) the analyses of the heat removal capabilities of the spray water system and the fan cooler heat exchangers.

The NRC's acceptance criteria for the Containment Heat Removal System are based on:

- GDC-38, insofar as it requires that the Containment Heat Removal System be capable of rapidly reducing the containment pressure and temperature following a loss of coolant accident (LOCA), and maintaining them at acceptably low levels.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided today in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following is the applicable CR-3 specific criteria for the Containment System:

- FSAR Section 1.4.52, Containment Heat Removal Systems, insofar as it requires that the containment heat removal system be capable of rapidly reducing the containment pressure and temperature following a LOCA, and maintaining them at acceptably low levels [GDC-38].

FSAR Sections 6.2 and 6.3 provide criteria for the post-accident heat removal capabilities of the Reactor Building (RB) Spray System operating in conjunction with the Reactor Building Emergency Cooling System.

Additionally, FSAR Section 6.1.2.1 provides criteria for the available Net Positive Suction Head (NPSH) to the decay heat removal pumps.

2.6.5.2 Technical Evaluation

Introduction

This section discusses the Containment Heat Removal Systems modeled in the containment integrity analysis for a postulated LOCA event (as described in Section 2.6.1, Primary Containment Functional Design) in support of the EPU operation. The EPU conditions increase the heat available to be released into containment, and thus, subsequent heat loads on the Containment Heat Removal Systems.

The Containment Heat Removal Systems are described in FSAR Sections 6.2 and 6.3. Two means of removing heat from the containment atmosphere are provided: the RB Cooling Units (RBCUs) and the RB Spray System. FSAR Section 6.3 describes three combinations of RBCUs and RB Spray that are acceptable for limiting the Reactor Building below the design pressure of 55 psig and the design temperature of 281°F after a design basis LOCA. Each of the three combinations is required to provide sufficient steam-condensing capacity to avoid containment overstress, and to remove that portion of the decay heat and other heat sources released to the containment atmosphere. The Nuclear Services Closed Cycle Cooling System (SW) is the heat sink for the RBCUs.

During a large break LOCA that evolves to sump recirculation, the Decay Heat Removal System (i.e., the Low Pressure Injection (LPI) System) would function to remove heat energy from the core and Reactor Coolant System (RCS), and therefore the containment, via the containment sump. The LPI System is discussed in Section 2.8.5.6.3, Emergency Core Cooling System and Loss of Coolant Accidents.

Description of Analysis and Evaluations

The Containment Heat Removal Systems were not required to be modified for the EPU. The power level increase and the increase to the RCS average temperature (from 579°F to 582°F) result in increased demand on the Containment Heat Removal Systems. The impact of this is summarized below.

The NPSH requirements for the Containment Heat Removal System pumps were evaluated for the EPU. Prior to sump recirculation, the low pressure injection and building spray pumps take suction from the borated water storage tank (BWST). The conditions of the BWST (inventory and temperature) are unchanged for the EPU. There are no hardware modifications being made to the systems between the BWST and the building spray (BS) or decay heat (DH) pumps. The evaluations performed for the EPU determined that, based on expected flow rates for LPI and BS, adequate NSPH margin is maintained. The EPU evaluation also considered the scenario with pumps aligned to the containment sump. The flow rate requirements for these pumps during sump swap over and subsequent sump suction flow are unchanged for the EPU. The containment building conditions and sump temperature change slightly for the EPU (Section 2.6.1, Primary Containment Functional Design), but this slight sump temperature change has insignificant impact on NPSH results which are calculated at containment sump saturated conditions. As demonstrated in Section 2.6.1 the Containment Heat Removal Systems remain adequate for rapidly reducing the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. Since no modifications to the Containment Heat Removal Systems were postulated in order to achieve the results in Section 2.6.1, the design of these systems remains adequate with implementation of the EPU.

Results

It was determined that the Containment Heat Removal System pump has adequate NPSH post-EPU. The adequacy of the spray water system and the fan cooler heat exchangers is demonstrated in Section 2.6.1, Primary Containment Functional Design.

2.6.5.3 Conclusion

CR-3 has reviewed the Containment Heat Removal Systems assessment and concludes that it has adequately addressed the effects of the proposed EPU. CR-3 finds that the systems will continue to meet the requirements of FSAR Section 1.4.52 for provision of independent systems for reducing the containment pressure and temperature following a LOCA. Therefore, CR-3 finds the proposed EPU acceptable with respect to Containment Heat Removal Systems.

2.6.5.4 References

None.

2.6.6 Pressure Analysis for Emergency Core Cooling System Performance Capability

2.6.6.1 Regulatory Evaluation

Following a loss-of-coolant accident (LOCA), the Emergency Core Cooling System (ECCS) supplies water to the reactor vessel to reflood, and thereby cool the reactor core. The core flooding rate will increase with increasing containment pressure. CR-3 reviewed analyses of the minimum containment pressure that could exist during the period of time following a LOCA until the core is reflooded to confirm the validity of the containment pressure used in ECCS performance capability studies. The CR-3 review included assumptions made regarding heat removal systems, structural heat sinks, and other heat removal processes that have the potential to reduce the containment pressure.

The NRC's acceptance criteria for the pressure analysis for Emergency Core Cooling System Performance Capability are based on:

- 10 CFR 50.46, insofar as it requires the use of an acceptable Emergency Core Cooling System evaluation model that realistically describes the behavior of the reactor during LOCAs, or an Emergency Core Cooling System evaluation model developed in conformance with 10 CFR 50, Appendix K.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predate the GDC provided today in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3. Additionally, FSAR Section 14.2.2.5 describes the methodology used to analyze the large break LOCA insofar as it requires the use of an acceptable Emergency Core Cooling System evaluation model that realistically describes the behavior of the reactor during LOCAs, or an emergency core cooling system evaluation model developed in conformance with 10 CFR 50, Appendix K [10 CFR 50.46].

2.6.6.2 Technical Evaluation

<u>Introduction</u>

This section discusses the containment backpressure analysis used in the large break LOCA analysis to support the EPU. The ECCS containment backpressure analysis for a large break LOCA is performed as prescribed in the current large break LOCA evaluation model (EM) (Reference 1).

<u>Description of Analyses and Evaluations</u>

Large break LOCA can be treated analytically in three separate phases: blowdown, refill, and reflood. The blowdown phase is characterized by the rapid depressurization of the Reactor Coolant System (RCS) to a condition nearly in pressure equilibrium with its containment surroundings. The RELAP5/MOD2-B&W code (Reference 2) calculates system thermal hydraulics and cladding temperature responses. The thermal hydraulic transient calculations are continued by the REFLOD3B code

(Reference 3) to determine refill time and core reflooding rates. The BEACH code (Reference 4) determines cladding temperature responses during the refill and reflood phases of the transient.

The CONTEMPT code (Reference 5) calculates the containment pressure response with time and requires mass and energy release input from both RELAP5 and REFLOD3B codes. The containment pressure response is intended to be conservatively low and as such, includes the effects of operating all pressure-reducing systems and processes, e.g., maximized ECCS flows. The computer codes documented in References 2 through 5 are the latest code versions supporting the evaluation model.

As specified in 10 CFR 50, Appendix K, the containment backpressure boundary condition analysis is acceptable if the containment pressure used for evaluating the cooling effectiveness during reflood does not exceed a pressure calculated conservatively for this purpose. The calculation process, which is iterative to ensure a convergent solution, should include the effects of operation of all installed pressure-reducing systems and processes. CR-3 and AREVA NP have ongoing processes which ensure that the values and ranges used in the ECCS containment backpressure analysis for a large break LOCA conservatively bound the values and ranges of the plant as-operated for those parameters. The analysis included the power level increase, increase in the nominal RCS average temperature, changes in the steam generator and steam generator tube plugging (SGTP), and modification of the number of reactor building fan coolers available on ES actuation. The selection of inputs satisfies the regulatory requirement.

<u>Results</u>

Figure 2.6.6-1 provides a comparison of the calculated pre-EPU and EPU minimum containment pressure responses. The results reflect the increased mass and energy release for the EPU case. There is a slight divergence in the long-term trend between the two cases, with the pre-EPU case showing a slowly decreasing pressure, while the EPU case shows a slowly increasing pressure. The slowly increasing containment pressure suggests that some input changes made to better reflect the plant's actual configuration (e.g., only one Reactor Building Cooling fan starting on Engineered Safeguards (ES) are contributing to the increase. However, the rates of change are small, with the EPU case showing an increase of less than 1 psid from about 50 seconds to 300 seconds, and with the pre-EPU case showing a decrease of 1.6 psid in that same time period. This difference in trend is not significant for a plant with reactor internals vent valves (RVVVs). With RVVVs, the peak clad temperature occurs near the end of the adiabatic heat-up period, 25 to 30 seconds after the break opening, well before the actuation of the Reactor Building Cooling fans can influence the containment pressure. As such, this small difference has a negligible impact on the ECCS analysis.

An additional evaluation was also performed for partial power conditions. It shows that the containment pressure response calculated for the full power conditions is acceptable for use in all partial power analyses. A sensitivity study concluded that for partial powers, the variation in the containment pressure response up to the time of PCT was minimal, and thus the resulting effect on the calculated PCT would also be minimal. This is expected since the stored energy of the RCS fluid, which is primarily dependent upon Tave, remains unchanged at partial powers for OTSG plants. This conclusion for partial power is equally applicable to partial power operation with three RCPs, since the reduced flow rates with three-RCP operation would not significantly alter the mass and energy release rates to the containment during the LOCA.

The minimum containment pressure analysis used in the CR-3 EPU large break LOCA analyses has adequately accounted for plant operation at the EPU power level and was performed using acceptable analytical models. It is further concluded that the evaluation has demonstrated that the containment pressure curve used in the large break LOCA analysis is acceptable.

2.6.6.3 Conclusion

CR-3 has reviewed the impact that the proposed EPU would have on the minimum containment pressure analysis and concludes that this area of review has been adequately addressed to ensure that the requirements in 10 CFR 50.46 regarding ECCS Performance Capability will continue to be met following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to minimum containment pressure for Emergency Core Cooling System Performance Capability.

2.6.6.4 References

1. AREVA Topical Report BAW-10192-A, Revision 0, “BWNT LOCA – Loss of Coolant Accident Evaluation Model for Once Through Steam Generator Plants,” June 1998.

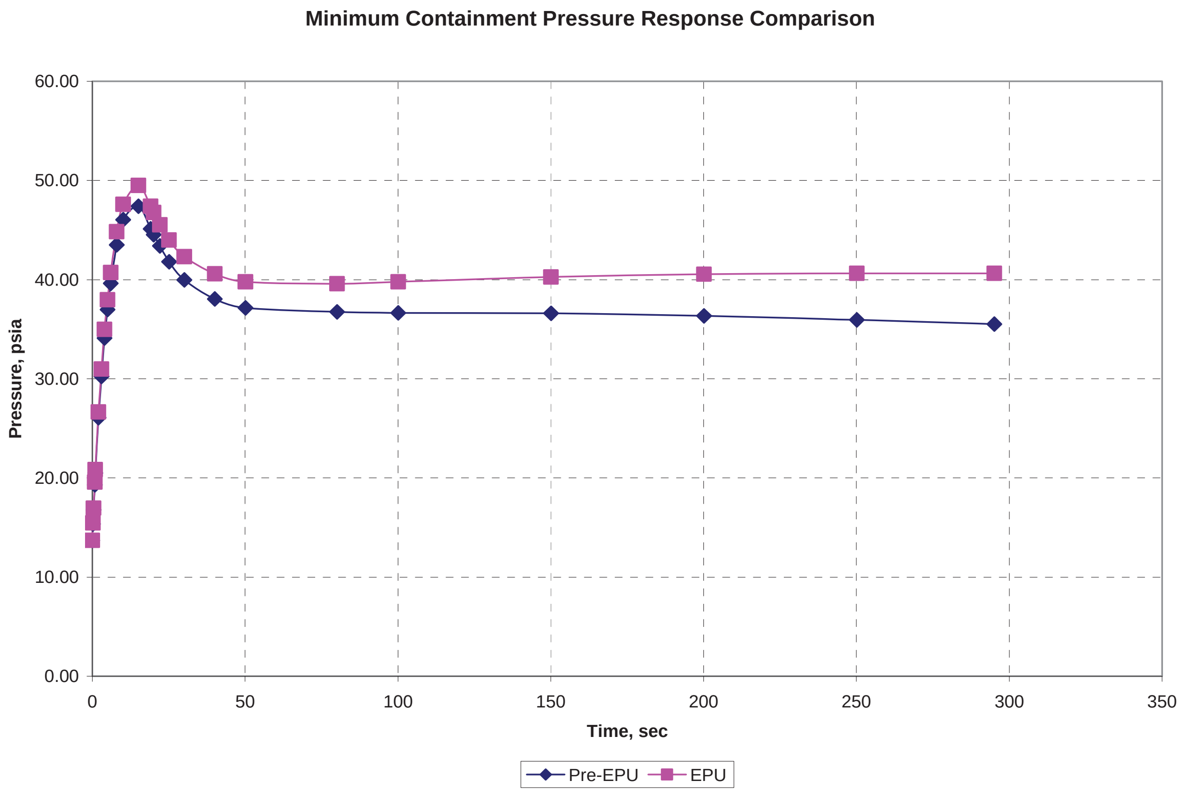
2. AREVA NP Proprietary Topical Report BAW-10164P-A, Rev. 6, “RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis,” June 2007.

3. AREVA NP Proprietary Topical Report BAW-10171P-A, Rev. 3, “REFLOD3B – Model for Multinode Core Reflooding Analysis,” December 1995.

4. AREVA NP Proprietary Topical Report BAW-10166P-A, Rev. 5, “BEACH – A Computer Program for Reflood Heat Transfer During LOCA,” November 2003.

5. AREVA NP Proprietary Topical Report BAW-10095-A, Rev. 1, “CONTEMPT – Computer Program for Predicting Containment Pressure-Temperature Response to a LOCA,” April 1978.

Figure 2.6.6-9


Minimum Containment Pressure Response Comparison
60.00
50.00
40.00
30.00
20.00
10.00
0.00
Pressure, psia
0
50
100
150
200
250
300
350
Time, sec
Pre-EPU
EPU

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

2.7.1.1 Regulatory Evaluation

CR-3 has reviewed the Control Room Habitability System and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of CR-3's review was to ensure that the control room can be maintained as the backup center from which technical support center (TSC) personnel can safely operate in the case of an accident. CR-3's review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination.

The NRC's acceptance criteria for the Control Room Habitability System are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents, including the effects of the release of toxic gases; and
- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.11, Control Room, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident. [GDC-19]

Additionally, FSAR Section 9.7.1 provides criteria for the Control Room Habitability System insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents. FSAR Section 7.4.5.1 provides criteria related to the protection from toxic gases. [GDC-4]

2.7.1.2 Technical Evaluation

Introduction

The Control Room Habitability System is comprised of the Control Complex Habitability Envelope (CCHE) and the emergency mode of operation of the Control Room Heating, Ventilation, and Air Conditioning (CRHVAC) System.

CCHE is comprised of the top five elevations of the Control Complex. The lower floor is isolated from the CCHE under accident conditions. The top floor of the CCHE contains the control complex ventilation equipment, thus it is all internal to the CCHE. The control room is one floor below the ventilation equipment room. The CCHE, along with Control Room Emergency Ventilation System (CREVS) are designed to protect the operator in case of a radiological or toxic gas release. The control complex is not pressurized to limit inleakage. Leak tightness and filtration capability provide the necessary level of protection for the control room operator to ensure that exposure limits associated with postulated accidents and toxic gas events are not exceeded.

The CRHVAC System has two modes of operation: normal control room area ventilation and control room emergency ventilation. The normal duty system mode is operated from the control room and runs continuously. Two 100% capacity trains of supply and return fans utilize a percentage of outside air to ventilate all areas of the Control Complex. Chilled Water is supplied from the Control Complex Chilled Water System which is a major subsystem of the Chilled Water (CH) System. In the emergency mode of operation the bubble tight outside air dampers close and the system switches to the recirculation mode. This establishes the boundary of the “Habitability” envelope (CCHE). Upon receipt of a high radiation signal or loss of power the normal duty supply and return units are tripped. The emergency supply and return fans must be manually started from the control room. Once started, the air flow is directed through the Control Complex emergency charcoal filters in the recirculation mode. The Control Complex Ventilation System is discussed in FSAR Section 9.7.2.1.g.1. Further discussion of the CRHVAC system can be found is Section 2.7.3.1. Discussion of the limiting accident for Control Room Habitability can be found in Section 2.9.2 Radiological Consequence Analysis.

Description of Analyses and Evaluations

The CCHE and CRHVAC are designed to protect the control room from the effects of external events including toxic gas and smoke. There are no new chemical or combustible materials stored near or on-site as a result of the extended power uprate which would increase the generation of toxic gases. Therefore, none of these considerations are affected by the EPU.

The radiological consequences to the control room are affected by the EPU. The effects on dose were evaluated and found to be within limits. Section 2.9.2 describes the changes and their acceptability.

The TSC is located in a different structure, and is not part of the CCHE at CR-3. Therefore, being separate structures and systems, the control room can act to provide key TSC personnel with a backup location from which to perform their function, should that become necessary.

Results

Since no new chemical or combustible materials which could increase the generation of toxic gases are stored near or on-site as a result of the EPU, none of these considerations are affected by the EPU.

The control room can act to provide key TSC personnel with a backup location from which to perform their function, should that become necessary since The TSC is located in a different structure, and is not part of the CCHE at CR-3.

2.7.1.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on the ability of the Control Room Habitability System to protect plant operators against the effects of accidental releases of toxic and radioactive gases. CR-3 has adequately accounted for the increase of toxic and radioactive gases that would result from the proposed EPU. CR-3 further concludes that the Control Room Habitability System will continue to provide the required protection following implementation of the proposed EPU. Based on this, CR-3 concludes that the control room habitability system will continue to meet the requirements of FSAR Sections 1.4.11, 7.4.5.1, and 9.7.1. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Control Room Habitability System.

2.7.1.4.1 References

None

2.7.2 Engineered Safety Feature Atmosphere Cleanup

2.7.2.1 Regulatory Evaluation

Engineered Safety Feature (ESF) Atmosphere Cleanup Systems are designed for fission product removal in post accident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., emergency or post accident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. The CR-3 review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber sections from exceeding design limits.

The NRC's acceptance criteria for the Engineered Safety Feature Atmosphere Cleanup Systems are based on:

- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident;
- GDC-41, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents;
- GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and
- GDC-64, insofar as it requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.11, Control Room, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident [GDC-19];
- FSAR Section 1.4.62, Inspection of Air Cleanup Systems, insofar as it requires that systems to control fission products released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents [GDC-41];

- FSAR Section 1.4.69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions [GDC-61]; and
- FSAR Section 1.4.17 Monitoring Radioactivity Release, insofar as it requires that means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including AOOs, and postulated accidents. [GDC-64]

2.7.2.2 Technical Evaluation

Introduction

Other than the Control Room Area Ventilation System (addressed in Section 2.7.3.1), CR-3 has no safety-related atmosphere cleanup systems. Furthermore, the non-safety related atmosphere cleanup systems are not credited for fission product removal in post-accident environments.

The Reactor Building (RB) Purge System consists of a supply system and an exhaust system. The supply system consists of two 50% capacity fans, roughing filter for intake of outside air, electric heating coils, and inside and outside containment isolation valves. The exhaust system consists of an inside and outside containment isolation valve, two 50% capacity charcoal exhaust filters, two 50% capacity fans, and a monitored discharge path to the unit vent.

The RB atmosphere can be purged or pressure equalized using the 6 inch containment mini-purge valves. Mini-purge flow joins the normal Reactor Building Purge Exhaust System downstream of the redundant leak rate throttled valves and is used for post-accident venting of the RB through the charcoal exhaust filters.

Description of Analyses and Evaluations

The radiological consequences of the design basis accidents (DBAs) are presented in Section 2.9.2 Radiological Consequence Analysis. No credit is assumed for any filtration systems following a DBA. Since the original design basis dose rates are not exceeded, the temperature of the carbon adsorbers is not an issue for the EPU.

As stated in FSAR 1.4.62, the containment Mini-Purge System is utilized intermittently during normal plant operation to replace the atmosphere within containment with fresh air. Under an accident condition, the Mini-Purge System is isolated. Containment air cleanup for post-accident iodine removal is accomplished by use of a Reactor Building Spray System which is addressed in Section 2.5.3.1 Fission Product Control Systems and Structures. Therefore, the criterion for the ESF Atmosphere Cleanup Systems is not considered applicable at CR-3.

Results

Since CR-3 does not credit any atmosphere cleanup systems for fission product removal in post-accident environments, the proposed EPU has no effect on the ESF Atmosphere Cleanup Systems to control the release of post-accident radioactivity to the environment within regulatory limits. Control of post-accident radiation in the control room is discussed in Section 2.7.3.1. Control of the monitoring and release of normal radioactivity is discussed in FSAR Section 11.4.

2.7.2.3 Conclusion

CR-3 has reviewed the assessment of the effects of the proposed EPU on the ESF Atmosphere Cleanup Systems. CR-3 concludes that since CR-3 does not credit any atmosphere cleanup systems for fission product removal in post accident environments, the proposed EPU has no effect on the ESF Atmosphere Cleanup Systems to control the release of post accident radioactivity to the environment within regulatory limits. Based on this, CR-3 concludes that the ESF Atmosphere Cleanup Systems will continue to meet the CR-3 requirements of FSAR Sections 1.4.11, 1.4.17, 1.4.62, and 1.4.69. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Engineered Safety Feature Atmosphere Cleanup Systems.

2.7.2.4.1 References

None

2.7.3 Ventilation Systems

2.7.3.1 Control Room Area Ventilation System

2.7.3.1.1 Regulatory Evaluation

The function of the Control Room Area Ventilation System (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, anticipated operational occurrences (AOOs), and design basis accidents (DBA) conditions. The CR-3 review of the Control Complex Air Handling Systems focused on the effects that the EPU will have on the functional performance of the safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the Control Complex Air Handling Systems.

The NRC's acceptance criteria for the Control Complex Air Handling Systems are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;
- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident; and
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.11, Control Room, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident [GDC-19]; and
- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include means to control the release of radioactive effluents. [GDC-60]

Additionally, FSAR Section 9.7.1 provides design basis criteria for Control Complex Air Handling Systems insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. [GDC-4]

2.7.3.1.2 Technical Evaluation

<u>Introduction</u>

The CRAVS for CR-3 is comprised of two safety-related subsystems collectively referred to as the Control Complex Air Handling Systems. These subsystems are the Control Complex Ventilation (AH-XK) System and the Emergency Feedwater Initiation and Control (EFIC) (AH-XS) System, which are supported by the Chilled Water (CH) System. These systems fulfill basic functional requirements.

Control Complex Ventilation System (AH-XK)

The safety functions of the Control Complex Ventilation System (AH-XK) are:

- Provide cooling and maintain the vital area temperature to within the design values
- Provide protection for the control room operators during a High Radiation Signal or an Engineered Safeguards Reactor Building Isolation Signal

The normal duty system mode is operated from the control room and runs continuously. Two 100% capacity trains of supply and return fans utilize a percentage of outside air to ventilate all areas of the Control Complex. In the emergency mode of operation, the outside air dampers close and the system switches to the recirculation mode. Upon receipt of a high radiation signal or loss of power, the normal duty supply and return units are tripped. The emergency supply and return fans must be manually started from the control room. Once started, the air flows through the Control Complex emergency charcoal filters in the recirculation mode. The Control Complex Ventilation System is discussed in FSAR Section 9.7.2.1.g.1.

EFIC System (AH-XS)

EFIC System (AH-XS) is a separate system from the AH-XK System. One of two 100% redundant air handling units is operated continuously from the main control board in all plant modes. The system design includes single failure criteria, seismic qualification, and emergency power. The safety function of the EFIC System (AH-XS) is to provide cooling and maintain the design temperature in the four cubicles of the EFIC Room during all modes of plant operation. The EFIC Room HVAC System is evaluated in FSAR Section 9.7.2.1.g.6. The EFIC System is discussed in FSAR Sections 9.7.2.d.5 and 9.7.2.1.g.6.

Chilled Water (CH) System

Chilled Water is supplied from the Control Complex chillers, which are major components of the Chilled Water (CH) System. The CH System is discussed in FSAR Section 9.7.2.d.

Description of Analyses and Evaluations

The changes in heat loads for ventilation subsystems that make up the Control Complex Ventilation Systems were evaluated to ensure that the ventilation systems are capable of performing their intended functions under the proposed EPU and emergency modes at the EPU conditions.

Impact to Control Complex Ventilation System (AH-XK)

The EPU conditions involve an increase in the Reactor Coolant hot-leg temperature of 6.6°F maximum, which will have no impact on the AH-XK System loads during normal operation. Since the AH-XK System serves areas containing instrumentation and control equipment, and areas intended for human occupancy, the various loads on the AH-XK System are not impacted by operation at the EPU conditions.

Accident conditions at the EPU conditions also have no impact on the AH-XK System loads. The thermal loads are based on the operation of equipment within the rooms. Differences occur as equipment energizes and de-energizes. Control Complex heat load calculations assume that the equipment heat load generated within each Control Complex room is constant for the duration of the analyzed DBAs. The same assumption applies for the EPU conditions. The heat load increases for EPU are small and the equipment operational thermal loads do not change under the EPU loss of coolant accident conditions. The EPU heat loads in the Control Complex were evaluated for LOCA/LOOP and LOCA/nonLOOP conditions and the maximum allowable Control Complex temperatures are not exceeded.

The Control Complex is not pressurized to limit inleakage. Leak tightness and filtration capability provide the necessary level of protection for the control room operators to ensure that exposure limits associated with DBAs and toxic gas events are not exceeded. Additionally, these features prevent entry of smoke into the control complex in the event of a fire. Since this function is independent of temperature, and that the EPU conditions create no change to system flow rates or flow paths, this safety function is not impacted by the EPU. Section 2.9.2 Radiological Consequence Analysis evaluated the control room dose consequences due to the EPU and has determined that the dose remains within regulatory limits.

The flow paths and flow rates associated with the AH-XK System do not change as a result of the EPU conditions. Also, the production and buildup of hydrogen in the Control Complex and Battery Rooms do not change as a result of the EPU conditions; therefore, this safety function is not impacted.

The normal operation flow paths and flow rates associated with the AH-XK System do not change as a result of the EPU conditions. Therefore, the normal system cooling in the non-vital areas of the building is not impacted by the EPU conditions. Similarly, normal system operation to provide fresh air and make-up air for personnel comfort for the fume hood operations in the Controlled Access Area is not impacted by the EPU conditions. And, proper system ventilation in the Controlled Access Area is such that normal operation airflow is in the direction of increasing radioactivity and is not impacted by the EPU conditions.

Impact to EFIC System (AH-XS)

The thermal load of the equipment contained in these rooms is essentially constant for varying conditions. These heat loads will only increase slightly as a result of the EPU power uprate, and the maximum allowable temperatures will not be exceeded.

Other evaluations related to the Control Room Area Ventilation System are addressed in Section 2.9.2.

Results

The proposed EPU has no effect on the ability of the Control Complex Ventilation Systems to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. CR-3 has adequately accounted for the increase of radioactive gases that would result from a DBA under the proposed EPU operating conditions, and any associated changes to parameters affecting environmental conditions such as toxic gases and smoke for control room personnel and equipment. The systems will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU.

Thus, following the EPU, CR-3 will continue to meet the current licensing basis with respect to the applicable Criteria for the Control Complex Ventilation Systems specified in FSAR Section 1.4. The effects of potential releases to the environment are evaluated in Section 2.10.1, Occupational and Public Radiation Doses, and remain within current limits following the EPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR Part 50, Appendix I, as described in Section 2.5.6.1, Gaseous Waste Management Systems.

2.7.3.1.3 Conclusion

CR-3 has reviewed the assessment of the effects of the proposed EPU on the ability of the Control Complex Ventilation Systems to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. CR-3 concludes that the system design has adequately accounted for the increase of toxic and radioactive gases that would result from a DBA under the conditions of the proposed EPU, and associated changes to parameters affecting environmental conditions for control room personnel and equipment. Accordingly, CR-3 concludes that the Control Complex Ventilation Systems will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. CR-3 also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, CR-3 concludes that the Control Complex Ventilation Systems will continue to meet the requirements of FSAR Sections 1.4.11, 1.4.70, and 9.7.1. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Control Complex Ventilation System.

2.7.3.1.4 References

None

2.7.4 Spent Fuel Pool Area Ventilation System

2.7.4.1 Regulatory Evaluation

The function of the Spent Fuel Pool Area Ventilation System is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, anticipated operational occurrences, and following postulated fuel-handling accidents. The CR-3 review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system.

The NRC's acceptance criteria for the Spent Fuel Pool Area Ventilation System are based on:

- GDC-60, insofar as it requires system design to include a means to control the release of radioactive effluents.
- GDC-61, insofar as it requires that systems which contain radioactivity to be designed with appropriate confinement and containment.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of the FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the facility design include those means necessary to control the release of plant radioactive effluents [GDC-60]; and
- FSAR Section 1.4.69, Protection Against Radioactivity Release from Spent Fuel and Waste Storage, insofar as it requires that that systems which may contain radioactivity to be designed with appropriate confinement and containment [GDC-61].

2.7.4.2 Technical Evaluation

Introduction

The function of the Spent Fuel Pool Area Ventilation System is described in FSAR Section 9.7.2. The Spent Fuel Pool Area Ventilation System includes the following systems:

- Spent Fuel Pit Supply System (AH-XH)
- Auxiliary and Fuel Building Exhaust System (AH-XJ)
- Fuel Handling Area Supply System (AH-XE)

These systems control airborne radioactivity in the spent fuel pool area during normal operating conditions. However, these systems are not safety related and not credited for controlling airborne radioactivity in the area during anticipated operational occurrences or following postulated fuel-handling accidents.

Description of Analyses and Evaluations

The Spent Fuel Pool Area Ventilation System was evaluated to ensure it is capable of performing its intended functions at the EPU conditions. The decay heat loads in the spent fuel pool increase due to the EPU conditions. The EPU decay heat loads and pool water temperatures have been evaluated to ensure that the system is capable of performing its intended functions for the EPU operation and refueling modes. The activities that occur in the spent fuel pit are unaffected by the EPU, therefore there are no impacts on that portion of the Spent Fuel Pool Area Ventilation System due to the EPU. Other evaluations, including dose effects following postulated fuel-handling accidents, are addressed in the following sections:

- Spent Fuel Pool Cooling System evaluation - Section 2.5.4.1, Spent Fuel Cooling and Cleanup System.
- Offsite dose consequences of a fuel handling accident - Section 2.9.2, Radiological Consequences Analysis.

There are no changes associated with operation of the Spent Fuel Pool Area Ventilation System at the EPU conditions and the EPU does not add any new or previously unevaluated materials to the system. System component internal and external environments remain within the parameters previously evaluated.

Results

The air temperature in the spent fuel pool area is affected by heat released from the spent fuel pool. Although the decay heat in the spent fuel is greater at the EPU conditions, the spent fuel pool water temperature during the normal and abnormal EPU operation does not exceed the assumed maximum value as discussed in Section 2.5.4.1, Spent Fuel Pool Cooling and Cleanup System. Therefore, the Spent Fuel Pool Area Ventilation System will maintain the required air temperature conditions for personnel and equipment during the EPU operation (see Section 2.5.4.1).

The design of the Spent Fuel Pool Area Ventilation System will not change following the implementation of the EPU. Airborne radioactivity released from the spent fuel in the pool will continue to be collected and exhausted by the auxiliary building ventilation system. Therefore, the control of airborne radioactivity in the spent fuel pool area is not affected following implementation of the EPU. The evaluation of the ability of the Spent Fuel Pool Area Ventilation System to maintain the required temperature conditions and to contain radioactivity to permit personnel access during the EPU demonstrates that there is no impact on the system design capability following the EPU implementation. This system was evaluated in Section 2.10.1, Occupational and Public Radiation Doses, and no changes are required as a result of the EPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR Part 50 Appendix I and is described in the Offsite Dose Calculation Manual.

The offsite dose consequences of a fuel handling accident are addressed in Section 2.9.2, Radiological Consequences Analysis.

2.7.4.3 Conclusion

CR-3 has assessed the effects of the proposed EPU on the Spent Fuel Pool Area Ventilation System. CR-3 concludes that the evaluation adequately accounts for the effects of the proposed EPU on the system's capability to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, CR-3 concludes that the Spent Fuel Pool Area Ventilation System will continue to meet the requirements in FSAR Sections 1.4.70 and 1.4.69. Therefore, CR-3 finds the proposed EPU is acceptable with respect to the Spent Fuel Pool Area Ventilation System.

2.7.4.4 References

None

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

2.7.5.1 Regulatory Evaluation

The function of the Auxiliary and Radwaste Area Ventilation System and the Turbine Areas Ventilation System is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during anticipated operational occurrences (AOOs), and after postulated accidents. The CR-3 review focused on the effects of the EPU on the functional performance of the safety-related systems.

The NRC's acceptance criteria for the Auxiliary and Radwaste Area Ventilation System and the Turbine Area Ventilation System are based on:

- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the facility design include those means necessary to maintain control over the plant radioactive effluents. [GDC-60]

2.7.5.2 Technical Evaluation

Introduction

The CR-3 ventilation systems that are enveloped by the classification of Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems are as follows:

- Auxiliary Building Supply System (AH-XD)
- Fuel Handling Area Supply System (AH-XE)
- Spent Fuel Pit Supply System (AH-XH)
- Auxiliary and Fuel Building Exhaust System (AH-XJ)
- Turbine Building Ventilation System (AH-XN)
- Turbine Building Sampling Room Cooling System (AH-XN)

- Turbine Building Switchgear Rooms Ventilation System (AH-XN)
- Turbine Building Non Class 1E Battery Room Ventilation System (AH-XN)
- Intermediate Building Air Handling System (AH-XM)

None of these systems perform a nuclear safety function. The Auxiliary Building contains other ventilation sub-systems that do perform a safety function and those systems are evaluated in Section 2.7.6, Engineered Safety Feature Ventilation System.

FSAR Section 9.7, Plant Ventilation Systems, provides a more detailed description of the ventilation systems, the functions of the systems and operation of the systems that comprise the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems listed above.

Evaluations of the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems for the impact of the EPU have been performed. The evaluations conclude that the EPU will have only minor impact on any of the addressed systems.

Description of Analyses and Evaluations

The changes in heat loads for the CR-3 ventilation systems that comprise the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems were evaluated to ensure that the ventilation systems are capable of performing their intended functions under the normal EPU modes of operation.

The function of the Auxiliary Building Supply System (AH-XD) is to provide filtered and conditioned air to the Auxiliary Building described in FSAR Section 9.7.2. Considering the large acceptable building temperature range (55°F to 122°F), any localized EPU-related temperature affects such as ambient air temperature increases in the fuel pool areas would be within the system's air removal capability. Therefore there is no EPU-related impact on this operational function of the AH-XD System.

One function of the Fuel Handling Area Supply System (AH-XE) is to provide filtered and conditioned air to the Fuel Handling area. As the EPU-related spent fuel begins to be handled and stored in the Fuel Handling Area, the overall ambient temperature of the air and pool water in the area will increase slightly. Since a very large temperature range (55°F to 122°F) is acceptable within the Fuel Handling area, minor effects of the EPU stored and transferred fuel can be accommodated within the AH-XE System's ventilation capability, which is comprised of outside air regulation and electric heaters - a non-nuclear safety-related function. This capability is not be impacted by the EPU conditions, so there is no impact to this system operational function.

A second operational function of the AH-XE System is to provide ventilation and, in conjunction with the Auxiliary and Fuel Building Exhaust System (AH-XJ), to provide a slight negative pressure to ensure that air flows in the direction of increasing radioactivity (FSAR Section 9.7.2). The EPU conditions will not change flow rates or flow paths in the AH-XE System, therefore the ability to maintain a slight negative building pressure will not be affected.

The operational function of the Spent Fuel Pit Supply System (AH-XH) is to maintain positive ventilation across the top of the spent fuel pits (FSAR Section 9.7.2). The system consists only of fans, dampers, and ductwork that function to transport stored fuel gases to the Auxiliary Building Exhaust System. The

EPU conditions will not affect the flow rate or flow path of the Spent Fuel Pit Supply System, therefore the ability to maintain positive ventilation across the top of the spent fuel pits will not be impacted.

The operational function of the Auxiliary and Fuel Building Exhaust System (AH-XJ) is to limit the release of radioactivity to the environment (FSAR Section 9.7.2). Radioactivity release to the environment is limited by maintaining a negative pressure within the Auxiliary Building by coordination of supply and exhaust air flow rates. AH-XJ exhaust release is to the unit vent after being filtered and monitored within the AH-XJ System. Radiological conditions, regardless of the EPU conditions, are monitored and processed accordingly. The ability to maintain negative building pressure by flow rate, damper position, and fan interlocks will not be affected by the EPU.

The operational function of the Turbine Building Ventilation System (AH-XN) is to provide air circulation through the Turbine Building to prevent excessive heat build-up (FSAR Section 9.7.2.1.h).

The EPU includes significant plant modifications that will increase the potential for heat load in the Turbine Building. The potential heat load will primarily be dissipated through a more efficient turbine generator resulting in an increased electrical output rather than waste heat. Any increase in the Turbine Building heat load from the EPU will be from newly added insulated vessels and piping and replacement pump motors of larger horsepower ratings.

Insulated vessels and piping operate at temperatures related to Feedwater and Main Steam temperatures, which will increase slightly at the EPU conditions. The new vessels and piping are insulated to existing plant specification to maintain a specific surface temperature. The additional heat load is a function of added insulated area which is insignificant compared to the heat transferred directly to the air by the replacement motors.

The Turbine Building motors being upgraded to support operation at the EPU conditions are as follows:

- The two Secondary Services Closed Cycle Cooling System (SC) pump (SCP-1A/1B) motors were upgraded from 350 hp to 600 hp during R16, Fall 2009. One pump is normally in service.
- The two Condensate System (CD) pump (CDP-1A/1B) motors will be upgraded from 2000 hp to 2500 hp. Both pumps are normally in service.
- The two Feedwater System (FW) Booster Pump (FWP-1A/1B) motors will be upgraded from 2500 hp to 3750 hp. Both pumps are normally in service.

The replacement motors being upgraded are located on the lowest level (95' elevation) of the Turbine Building and subsequently the coolest location. The heat load will increase in the vicinity of the replacement motors but the local ambient temperatures will not exceed the rated capability of the installed components adjacent to replacement motors. The existing air circulation will be sufficient to prevent excessive heat build-up in the Turbine Building. Refer to Appendix E, Major Plant Modifications for further description of these plant modifications.

The operational function of the Turbine Building Sampling Room Cooling System (AH-XN) is to maintain the Sampling Room temperature by supplying cool air from a roof-mounted, self-contained, packaged air conditioner. Samples drawn from the various secondary system locations are either pre-cooled or provide insignificant heat input to the sampling room. In either case, operation at the EPU conditions does not involve any significant temperature increases in either steam or feedwater. Normal temperature changes in the Sampling Room will be accommodated by the thermostatically-controlled self-contained air conditioner, AHU-1. The Sampling Room is enclosed and insulated from the overall Turbine Building (TB) atmosphere. This non-safety-related operational function will not be impacted by the EPU conditions.

The first operational function of the Turbine Building Switchgear Rooms Ventilation System (AH-XN) is to maintain a constant temperature and humidity in the Turbine Area Switchgear Rooms on 95' and 119' elevations and in the Instrumentation Calibration Room on 145' elevation to ensure proper operation of the electrical switchgear and calibration room equipment (FSAR Section 9.7.2.1.h). The Switchgear Rooms on 95' and 119' are separate enclosures and are insulated from the main Turbine Hall. Heating is controlled by electric heaters and cooling is provided by Appendix R Chilled Water. The impact on the switchgear rooms and their associated equipment will be minimal from a ventilation perspective. Therefore, this non-safety-related system function will not be impacted by operation at the EPU conditions.

The second operational function of the AH-XN system is to control the room dust content of the Turbine Area Switchgear Rooms and the Instrumentation Calibration Room (FSAR Section 9.7.2). Two filters in parallel are provided for filtering the overall system. An additional HEPA filter is provided in series to the parallel filter pair to filter the Instrumentation Calibration Room. Operation at the EPU conditions does not increase the dust content in the Switchgear and Calibration Room area. Therefore, this non-safety-related system function will not be impacted by operation at the EPU conditions.

The first operational function of the Turbine Building Non Class 1E Battery Room Ventilation System (AH-XN) is to maintain a constant room temperature of 77°F. Temperature is controlled by integral cooling and heating as part of the air handling unit. The EPU conditions will not change conditions within the Non Class 1E Battery Room; therefore this non-safety function is not impacted.

The second operational function of the Turbine Building Non Class 1E Battery Room Ventilation System is to provide continuous hydrogen removal. A roof-mounted exhaust fan is provided on independent power from the supply unit to provide continuous hydrogen removal. The EPU conditions will not change conditions within the Non Class 1E Battery Room; therefore this non-safety function is not impacted.

The first operational function of the Intermediate Building Air Handling System (AH-XM) is to remove internal heat from the Intermediate Building during normal operation. (FSAR Chapter 9, Figure 9-14) The major pieces of equipment located within the Intermediate Building contributing to the heat load are the 24-inch Main Steam piping and the 18-inch Feedwater piping, the motor and turbine-driven Emergency Feedwater Pumps, and various motor control centers. Since there will be only a minor change in the EPU temperatures for Feedwater and Main Steam , there will be no significant additional heat input from these sources. There are no EPU-related changes to the motor and turbine-driven Emergency Feedwater Pumps, therefore their operation will remain within the existing design parameters. Consequently, there will be no additional heat input to the Intermediate Building. Therefore, there is no impact to this non-safety-related function to remove heat from the Intermediate Building during normal operation.

The second operational function of the AH-XM System is to maintain the Intermediate Building temperature within the design temperature range. Temperature is controlled only by outside air temperature, fan operation, and damper positioning. There is no impact to this non-safety-related function to maintain a design temperature range in the Intermediate Building.

Results

The Auxiliary and Radwaste Area and Turbine Areas Ventilation System's ability to provide required temperature conditions for personnel and equipment during normal operation is unaffected by the changes proposed for the EPU. The increased heat loads in these areas are primarily due to changes in the main steam and feedwater system operating conditions and small increases in electrical loads. For plant areas where temperature is controlled by air conditioning units, the small increase in heat loads is well within the capacity of the units. For plant areas that use outside air exchange to provide cooling, outside air temperature changes dominate any potential temperature changes caused by the EPU.

The evaluation of the plant equipment changes for the proposed EPU did not identify any need to modify the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems. There are no equipment changes as a result of the EPU that could create a new potentially unmonitored airborne radioactive release path or affect the capability to maintain control over the plant radioactive effluents. The effects of potential releases to the environment are evaluated in Section, 2.10.1, Occupational and Public Radiation Doses, and remain within current limits following the EPU. The handling, control, and release of radioactive materials are in compliance with 10 CFR Part 50, Appendix I, as described in Section 2.5.6.1, Gaseous Waste Management System.

2.7.5.3 Conclusion

CR-3 has assessed the effects of the proposed EPU on the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems. CR-3 concludes that the evaluation has adequately accounted for the effects of the proposed EPU on the capability of these systems to; 1) maintain ventilation, 2) permit personnel access, and 3) control the concentration of airborne radioactive material in the following areas: auxiliary and radwaste equipment areas, turbine areas, and intermediate building areas. In addition, CR-3 concludes that the evaluation has adequately accounted for the effects of the proposed EPU on the capability of these systems to control release of gaseous radioactive effluents to the environment. Based on this, CR-3 concludes that the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems will continue to meet the requirements of FSAR Section 1.4.70. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems.

2.7.5.4 References

None

2.7.6 Engineered Safety Feature Ventilation System

2.7.6.1 Regulatory Evaluation

The function of the Engineered Safety Feature (ESF) Ventilation System is to provide a suitable and controlled environment for ESF components following certain anticipated transients and design basis accidents (DBAs). The CR-3 review of the ESF Ventilation System focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The CR-3 review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESF Ventilation System performance; (2) the capability of the ESF Ventilation System to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESF Ventilation System to control airborne particulate material (dust accumulation).

The NRC acceptance criteria for the Engineered Safety Feature Ventilation System are based on:

- GDC-4, insofar as it requires SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;
- GDC-17, insofar as it requires that onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety; and
- GDC-60, insofar as it requires that the plant design include means to control the release of radioactive effluents.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.24, Emergency Power for Protection Systems; and FSAR Section 1.4.39, Emergency Power for Engineered Safety Features, insofar as it requires onsite and offsite electric power systems be provided to permit functioning of SSCs important to safety [GDC-17]; and
- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include means to control the release of radioactive effluents [GDC-60].

FSAR Section 9.7.1 provides design criteria for ventilation systems insofar as it requires SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents [GDC-4].

2.7.6.2 Technical Evaluation

Introduction

The function of an ESF Ventilation System is to maintain temperatures within specified limits in areas containing safety-related equipment. At CR-3, this function is accomplished by multiple independent systems rather than a single system. These “essential“ ventilation systems are listed below, excluding the Control Complex Ventilation, which is addressed in Section 2.7.3.1, Control Room Area Ventilation System:

- Decay Heat Closed Cycle Pump Cooling System (AH-XF)
- Spent Fuel Coolant Pump Cooling System (AH-XG)
- Emergency Diesel Generator Air Handling System (AH-XL)
- Emergency Feedwater Initiation & Control (EFIC) Room HVAC (AH-XS)
- Emergency Feedwater Pump Building HVAC System (AH-XU)

Decay Heat Closed Cycle Pump Cooling System (AH-XF)

As a safety function, the AH-XF System cools and recirculates air in the room containing the Decay Heat Closed Cycle Cooling (DC) Pumps, to cool the DC Pump motors. The heat load source for AH-XF System is the operation of Decay Heat Closed Cycle Cooling Water (DC) Pumps and Motors. Air from the AH-XF fans blows over the DC Pumps/Motors and is recirculated back to the fan intake. DC water from the DC Pump discharge is routed to the fan coils, which are designed to cool the DC pump motors under ambient conditions of 150°F. Since the air cools the pump & motor combination, it is the cooling of the motors that can affect pump operability. Although the AH-XF System design does not include a filter section, potential airborne contamination as a result of normal operation is ultimately filtered by the Auxiliary Building Exhaust (AH-XJ) System.

Spent Fuel Coolant Pump Cooling System (AH-XG)

As a safety function, the AH-XG System cools and recirculates air in the room containing the Spent Fuel Coolant Pumps to cool the Spent Fuel pump motors. Cooling water from the Nuclear Services Closed Cycle Cooling (SW) is provided to each of two 100% capacity AH-XG fan coils, which have a design capacity sufficient for cooling under postulated ambient conditions of 140°F. Although the AH-XG System design does not include a filter section, potential airborne contamination as a result of normal operation is ultimately filtered by the Auxiliary Building Exhaust (AH-XJ) System.

Emergency Diesel Generator Air Handling System (AH-XL)

The AH-XL System provides continuous ventilation, and dissipates internal heat gains in each Emergency Diesel Generator (EDG) Room when the diesel engine is operating. Each EDG Room has a separate and identical ventilation system. Each system consists of two 50%-capacity cooling fans, one 100% capacity roughing filter for ventilation air, one 100% capacity filter for combustion air, one 100% capacity EDG control room exhaust fan, common ductwork, and controls. There is no provision for refrigeration or cooling water. The system is arranged so that the two cooling fans start automatically when the diesel starts and discharge filtered outside air into the diesel room for cooling. The system supplies sufficient air to maintain the room temperature less than 120°F. Single fan or dual fan operation is acceptable dependent upon fan supply air temperature. Both cooling fans start automatically when the diesel starts. The Emergency Diesel Generator Air Handling System is described in FSAR Section 9.7.2.1.h.2. The AH-XL System draws outside air for cooling and exhaust air into the auxiliary building; therefore radioactive effluents are not applicable to this system.

Emergency Feedwater Initiation & Control (EFIC) Room HVAC (AH-XS)

The EFIC Room HVAC System is part of the overall Control Complex Air Handling System. As a safety function, the AH-XS System provides cooling and maintains the design temperature in the four cubicles of the EFIC Room during all modes of plant operations. One of two 100% redundant air handling units is operated continuously from the main control board in all plant modes. The EFIC Room Coolers are normally cooled by the Chilled Water (CH) System. One train of AH-XS is protected against the consequences of a fire by receiving alternate cooling from the Appendix R Chilled Water System. The EFIC Room HVAC System is described in FSAR Section 9.7.2.1.h.6. The AH-XS System recirculates and cools air from the control complex; therefore radioactive effluents are not applicable to this system.

Emergency Feedwater Pump Building HVAC System (AH-XU)

The Emergency Feedwater Pump Building (EFPB) houses diesel-driven Emergency Feedwater Pump EFP-3. The AH-XU System provides ventilation for the three separate rooms of the EFPB, and heating and air conditioning for the EFPB Battery Room. The three rooms are the EFPB Tank Room, EFPB Battery Room, and the EFPB Diesel Pump Room. The Emergency Feedwater Pump Building Air Handling System is described in FSAR Section 9.7.2.1.h.8. The EFPB Pump Room is not influenced by the plant environment; therefore radioactive effluents are not applicable.

Description of Analyses and Evaluations

The changes in heat loads under the EPU conditions in areas served by the ESF Ventilation Systems were evaluated to ensure that the ventilation systems are capable of performing their intended functions, including the ability of the system to control airborne particulate material accumulation.

Results

Decay Heat Closed Cycle Pump Cooling (DC) System (AH-XF)

The temperature at which the DC System is placed in operation does not change. Additionally, the thermal load from the DC pump motors, based on horsepower and efficiency, will be unchanged. Therefore the safety function to maintain operability of the DC Pumps (and motors) is not impacted by the EPU power uprate.

Spent Fuel Coolant Pump Cooling System (AH-XG)

The cooling provided by the AH-XG System is directed at maintaining cooling of the motor that can affect operability. Therefore, even though the Spent Fuel Cooling (SF) System may be subjected to a slightly higher process temperature from the fuel pools, which in turn may cause the SF pump casing to be hotter, this increase in fuel pool temperature remains within the design basis temperature for the spent fuel pool. The surface temperatures of the SF pump casing remains within the design basis due to insulation and have no impact on the AH-XG System cooling capability. The motor horsepower will remain unchanged, resulting in the thermal load of the motors being unchanged. The cooling coils of the AH-XG fans have a design capacity sufficient for cooling under postulated ambient conditions of 140°F. Therefore, there is no impact to the safety function to maintain SF Motor operability.

<u>Emergency Diesel Generator Air Handling System (AH-XL)</u>

The heat load for the EDG room was determined using a generator load of 3500 kW, which is the upper limit on the 30 minute rating. This load continues to bound EPU conditions, therefore the EDG room heat load will not change as a result of the EPU conditions. There is no impact on AH-XL System's ability to perform its safety function.

<u>Emergency Feedwater Initiation & Control (EFIC) Room HVAC (AH-XS)</u>

The EFIC cabinets are electronic control cabinets; these heat loads will not be changing as a result of the EPU power uprate. There is no impact on the safety function of the AH-XS System.

<u>Emergency Feedwater Pump Building HVAC System (AH-XU)</u>

Due to its remote location with respect to the CR-3 plant, the EFPB Pump Room is not influenced by the plant environment. The heat load for the diesel pump room when the diesel is not operating will not change as a result of the EPU conditions. Operation of the emergency feedwater pumps (EFPs) is independent of unit power level, it is concluded that the current ventilation will continue to adequately maintain the room temperature below maximum during engine operation for the EPU conditions. Therefore, the current ventilation requirements will not change and the safety function to maintain non-operational temperature will continue to be met.

2.7.6.3 Conclusion

CR-3 has assessed the effects of the proposed EPU on the ESF Ventilation System. CR-3 concludes that the evaluation adequately accounted for the effects of the proposed EPU on the ability of the ESF Ventilation System to provide a suitable and controlled environment for ESF components. The CR-3 review further concluded that the ESF Ventilation System will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU, and will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, CR-3 concludes that the ESF Ventilation System will continue to meet the requirements of FSAR Sections 1.4.24, 1.4.39, 1.4.70, and 9.7.1. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Engineered Safety Feature Ventilation System.

2.7.6.4 References

None

2.7.7 Reactor Building Ventilation Systems

2.7.7.1 Regulatory Evaluation

The functions of the Reactor Building Ventilation Systems are to provide heat removal from the containment atmosphere and to control containment temperature thereby providing containment pressure control under normal and accident conditions. Control of containment temperature and pressure control under accident conditions are evaluated in Sections 2.6.1, Primary Containment Functional Design and 2.6.5, Containment Heat Removal. The CR-3 review of the Reactor Building Ventilation Systems in this section focused on the effects that the proposed EPU will have on the operational, functional performance of the these systems under normal conditions. The exception is the Reactor Building Purge System (AH-XC) which is discussed in Section 2.7.2, Engineered Safety Feature Atmosphere Cleanup.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The NRC's acceptance criteria for the Reactor Building Ventilation Systems are based on:

- FSAR Section 5.5, Ventilation and Purge Systems, which describes the means of heat removal from the Reactor Building (RB) during normal operation with containment recirculation fan coolers and methods of containment purging or pressure equalization; and
- FSAR Section 9.7, Plant Ventilation Systems, which provides a more detailed description of the system, the functions of the systems and operation of the systems.

2.7.7.2 Technical Evaluation

Introduction

The functions of the Reactor Building Ventilation Systems are to provide heat removal from the containment atmosphere and to provide containment pressure control under normal conditions as described in FSAR Section 9.7.

The following systems and subsystems described in FSAR Section 9.7.2 are included within the Reactor Building Ventilation Systems at CR-3:

- Reactor Building Recirculation System (AH-XA)
- Reactor Building Cavity Cooling (AH-XB)
- Reactor Building Steam Generator Cooling System (AH-XB)
- Control Rod Drive Cooling System (AH-XB)
- Reactor Building Air Supply System (AH-XB)

- Reactor Building Purge System (AH-XC) which is discussed in Section 2.7.2
- Penetration Cooling System (AH-XP)

FSAR Section 9.7, Plant Ventilation Systems, provides a more detailed description of the functions and operations of the ventilation systems that comprise the Reactor Building Systems listed above.

Description of Analyses and Evaluations

The changes in heat loads for ventilation subsystems in the containment were evaluated to ensure that the ventilation systems are capable of performing their intended functions under the normal EPU conditions. For the EPU versus Pre-EPU conditions, refer to Section 1.1, NSSS Parameters, Table 1.1-1.

Reactor Building Recirculation System (AH-XA)

The operational function of the Reactor Building Recirculation System (AH-XA) is to maintain the Reactor Building within the design maximum and minimum temperatures during normal operations.

The EPU will have a minor impact on the RB cooling fans non-safety function due to the increased heat load in the building. The ΔT across the hot-leg insulation increases by 6.4°F (~1.5%). Since the ΔT across the pressurizer insulation is unchanged and the ΔT across the cold-leg insulation is actually decreasing, the total heat loss from the RCS will increase by less than 1.5%. Thus the post-EPU building heat load will be no greater than 101.5% of the pre-EPU heat load, or 12.2×10^6 BTU/hr. The Industrial Cooling System (CI) is available to accommodate the RB Recirculation System normal cooling loads, with a capacity of 16.8×10^6 BTU/hr (1400 tons) [FSAR 5.5.1.2.a]. However, during the summer months, there are periods that the CI System capacity is reduced due to high ambient temperatures to the point that maintaining Reactor Building temperature less that 130°F (ITS limit) is challenged. This situation existed prior to the EPU and is only slightly exacerbated by the additional heat load due to the EPU. CR-3 has developed procedures to shift from CI to two trains of Nuclear Services Closed Cooling Water (SW) for cooling to the AH-XA cooling coils during these periods which provides the cooling capacity required to maintain Reactor Building bulk temperatures below the ITS limit. This capability is adequate to maintain bulk temperatures below 130°F with the increased heat load from the EPU.

Reactor Building Cavity Cooling (AH-XB)

The operational function of the Reactor Building Cavity Cooling System (AH-XB) is to remove heat from the reactor compartment cavity and nozzle penetration cavities.

The majority of the heat produced in the reactor cavity regions will be in the form of radiant heat from the surfaces of the insulated reactor vessel. For the EPU conditions, T_{COLD} decreases. The effect will be a small decrease in vessel temperature, resulting in a slight decrease in the ΔT across the vessel insulation. This should result in a slight decrease in reactor cavity heat load. Due to the cooler reactor coolant flow being channeled downward to the bottom of the vessel and then redirected upward to the bottom of the fuel assemblies, the amount of radiant heat coming from the insulated surfaces of the reactor vessel will not be adversely impacted. This downward coolant flow maintains the vessel temperature near to T_{COLD}.

The upper head regions will experience an increase in temperature because T_{HOT} increases. The heat transfer from the insulated surfaces of the upper head regions and the hot-leg piping will increase the ambient temperatures in the upper regions of the reactor cavity. The weighted percentage for the surface area of the vessel head regions and its hot-leg piping with respect to down comer regions and cold-leg piping, whose surfaces are exposed to T_{COLD}, will in effect have a relatively smaller weighted percentage. Therefore, the overall impact to the RB Cavity Cooling System is minimal, and the reactor compartment cavity and nozzle penetration cavity temperatures are expected to be maintained as required.

Reactor Building Steam Generator Cooling System (AH-XB)

The operational function of the Reactor Building Steam Generator Cooling (AH-AB) System is to remove heat from the steam generator compartments.

The majority of the heat produced in the Steam Generator compartment will be in the form of radiant heat from the surfaces of the insulated steam generators. For the EPU conditions, T_{HOT} increases and T_{COLD} decreases. The steam temperature decreases slightly and feedwater inlet temperature increases slightly for the EPU conditions. Some additional heat load would be transferred from the hot-legs, but since they are insulated and the temperature increase relatively small, the added heat load would not be significant. There are no specific temperature requirements associated with the Steam Generator Cooling System, and the system does not utilize cooling water. Cooling is accomplished by circulating local heated air to the overall Reactor Building air space. Heat removal is via the Reactor Building Recirculation (AH-XA) System. The AH-XB System supports the AH-XA System to ensure the average RB temperature is maintained below the ITS limit of 130°F.

Control Rod Drive Cooling System (AH-XB)

The operational function of CRD Cooling (AH-XB) is to remove heat around the CRD shroud to maintain cable and connector temperatures below rated values.

For the EPU conditions, CRD mechanism (CRDM) Service Structure operating temperatures will experience a negligible (less than 1.0°F) increase in temperature. However, the temperature limit for the CRD shroud temperature which is 150°F based on electrical connector and CRD position indicator enclosures located in the service structure has been challenged during the summer months. This condition existed prior to the EPU and the increase in temperature due to the EPU is negligible. While this is not considered an adverse condition since it has no immediate effect on equipment reliability, shortening the life of plant equipment is not economically desirable. Therefore, qualified component lifetime is trended with the cumulative impact monitored and preventive maintenance actions implemented as appropriate.

The EPU conditions do not affect the ability of the AH-XB System to perform the necessary heat removal to support the functions of the CRDMs as discussed in Section 2.8.4.1, Functional Design of Control Rod Drive System.

Reactor Building Air Supply System (AH-XB)

The operational function of the RB Air Supply (AH-XB) is to provide air flow to the operating floor and provide mixing of air throughout the Reactor Building during normal operations, utilizing the duct work arrangement.

For the EPU conditions, the changes to reactor coolant temperature do not impede the adequate mixing of air throughout the Reactor Building; therefore there is no impact to this system's function for the EPU.

Reactor Building Purge System (AH-XC)

The Reactor Building Purge System (AH-XC) is discussed in Section 2.7.2, Engineered Safety Feature Atmosphere Cleanup.

Penetration Cooling System (AH-XP)

The operational function of the Penetration Cooling (AH-XP) System is to supply cooled air to the hot pipe penetrations of the Reactor Building.

The four identical Main Steam System penetrations 105, 106, 107, and 201 will not be affected by the EPU conditions because the EPU steam temperature will decrease slightly (refer to Table 1.1-1). The two identical Feedwater System penetrations 108 and 423 also will not be affected by the EPU conditions because Feedwater temperature will be only slightly increased and remain below design temperature (refer to Section 2.5.5.4, Condensate and Feedwater). Decay Heat penetrations 342 and 343 provide the Decay Heat flowpath to the reactor vessel (refer to FSAR Figure 9-6, Sheet 1of 3). The temperature of these penetrations will not be affected by the EPU conditions since these penetrations are exposed only to return water temperatures, not RCS system temperatures. Decay Heat penetration 344 is the Reactor Coolant System (RCS) hot-leg supply to the Low Pressure Injection/Decay Heat System (refer to FSAR Figure 9-6, Sheet 2 of 3). Although Reactor Coolant hot-leg temperature is expected to be several degrees higher after the EPU, this penetration is not in service until the reactor has been shut down and the RCS has been depressurized and cooled below 280°F. Therefore, temperatures at this penetration will remain the same after the EPU.

The Penetration Cooling System receives its supply air from two sources, outdoor air and Turbine Building Ventilation supply. The EPU conditions will not affect the outdoor air supply temperature, which is determined by seasonal changes. The temperature from the air intake source from the Turbine Building will remain within design parameters as discussed in Section 2.7.5, Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems. The Turbine Building steady state air temperature is slightly higher as a result of the EPU, but Turbine Building ventilation is designed to maintain the building air temperature no more than 20°F higher than outside temperature at any time. Air temperature control for the AH-XP System includes modulation between outside air and Turbine Building Air and this air mixture is then cooled by the Chilled Water System (CH) in penetration coolers AHHE-13A/13B.

Refer to Section 2.6.5, Containment Heat Removal, for post accident operation of the Reactor Building Ventilation System and Containment Heat Removal System evaluation.

<u>Results</u>

For the EPU accident conditions, worst-case containment operational parameters are bounded by design conditions.

During power operation, the Reactor Building Ventilation System (AH-XA) will maintain containment bulk air temperature below the ITS limit (130°F) with the Industrial Cooling System (CI) capacity of $16.8x10^6$

BTU/hr (1400 tons) or Nuclear Services Closed Cooling Water System as required during estimated summer heat loads for the EPU.

The design and operation of the following systems are adequate to accommodate the EPU conditions at 3014 MWt:

- Reactor Building Cavity Cooling System (AH-XB)
- Reactor Building Steam Generator Cooling System (AH-XB)
- Control Rod Drive Cooling System (AH-XB)
- Reactor Building Air Supply System (AH-XB)
- The Penetration Cooling System (AH-XP)

2.7.7.3 Conclusion

The CR-3 review of the Reactor Building Ventilation Systems has adequately accounted for the effects of the proposed EPU on the ability of the Reactor Building Ventilation System to provide a suitable and controlled environment for the containment components during normal operation. Based on the above, the Reactor Building Ventilation Systems will continue to be acceptable following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Reactor Building Ventilation Systems.

2.7.7.4 References

None

2.8 Reactor Systems

2.8.1 Fuel System Design

2.8.1.1 Regulatory Evaluation

The Fuel System Design consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, and reactivity control rods. The CR-3 review of the Fuel System Design was to ensure that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The CR-3 review covered Fuel System Design damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents.

The NRC's acceptance criteria for the Fuel System Design are based on:

- 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance;
- GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs;
- GDC-27, insofar is it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and
- GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any loss-of-coolant accident (LOCA).

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including effects of AOOs [GDC-10];
- FSAR Section 1.4.28, Reactivity Hot Shutdown Capability; FSAR Section 1.4.29, Reactivity Shutdown Capability; and FSAR Section 1.4.30, Reactivity Holddown Capability, insofar as they require that the reactivity control systems be designed to have combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated

accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained [GDC-27]; and

- FSAR Section 1.4.37, Engineered Safety Features Basis for Design; FSAR Section 1.4.41, Engineered Safety Features Performance Capability; FSAR Section 1.4.42, Engineered Safety Features Components Capability; and FSAR Section 1.4.44, Emergency Core Cooling Systems Capability, insofar as they require that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA [GDC-35].

Additionally, FSAR Section 14.2.2.5 provides criteria for LOCA analyses insofar as it requires the use of an acceptable Emergency Core Cooling System evaluation model that realistically describes the behavior of the reactor during LOCAs, or an Emergency Core Cooling System evaluation model developed in conformance with 10 CFR 50.46.

2.8.1.2 Technical Evaluation

<u>Introduction</u>

Fuel System Design Features

The Mark-B-HTP fuel design was first introduced at CR-3, in Cycle 14 (Batch 16) beginning in 2003. A change from Zircaloy-4 HTP to M5® was introduced for the upper and intermediate grids, and instrument tubes in Cycle 15 (Batch 17), beginning in 2005. The Mark-B-HTP fuel design is a 15x15 fuel assembly design for operation in a Babcock & Wilcox 177 fuel assembly PWR reactor core. No changes to this fuel design are proposed for the CR-3 EPU.

The Mark-B-HTP fuel assembly has the following features:

- Fuel Rod Design: Standard 0.430" outside diameter with M5 cladding and short lower end caps. Zircaloy-4 end caps used for initial TIG-welded application
- Upper and Intermediate Grids: Zircaloy-4 HTP (Batch 16), M5 HTP (starting with Batch 17)
- Lower End Grid: Alloy 718 HMP (straight-channel HTP)
- Guide Tubes: M5 tubing with weep holes (as in older Mark-B designs) with lower end attachment screws using crimp-style locking mechanism (similar to Mark-BW)
- Grid Restraint System: Intermediate and upper end grids are welded to guide tubes with lower end grid secured by Zircaloy-4 rings welded above and below as in other HTP fuel assembly designs
- Instrument Tube: Zircaloy-4 (Batch 16), M5 (starting with Batch 17), MONOBLOC™ with modifications to the inner diameter to account for loss of spacer sleeve material
- Lower End Fitting: Debris resistant FUELGUARD
- Upper End Fitting: Standard removable (Mark-B10), with 6-leaf cruciform holddown spring

Mechanical analyses, as discussed in the following sections, have shown that the Mark-B-HTP components can withstand the stresses resulting from start-up, steady state operation, and shutdown conditions following the EPU. Mechanical analyses have also shown that Mark-B-HTP components maintain their functional integrity in the event of any post EPU major LOCA, and seismic loadings.

A full core of Mark-B-HTP fuel assemblies will reside in the CR-3 core for the EPU. For the purposes of the EPU analysis, bounding beginning of life (BOL) to end of life (EOL) Mark-B-HTP parameters have been used in the safety and design analyses. Table 2.8.1-1 and Figure 2.8.1-1 provide key features of the Mark-B-HTP fuel assembly.

Mechanical Compatibility and Performance

The mechanical design performance of the Mark-B-HTP fuel assembly for CR-3 pre and post EPU is shown to satisfy the Fuel System Design bases per FSAR Section 3. Compliance with the following acceptance criteria is confirmed by approved methodology of BAW-10133PA (References 8 and 9).

The EPU results in a larger ΔT_{CORE} which was addressed for the fuel mechanical structure. The impact of the higher coolant temperatures in the upper regions of the fuel assembly was addressed for fuel rod corrosion, guide tube corrosion, spacer grid corrosion, fuel assembly and spacer grid growth. Additionally, the change in the hydraulic lift load on the fuel assembly and spacer grids was evaluated in the holddown analysis and grid restraint system requirements.

The impact of potentially lower coolant temperatures in the lower regions of the fuel was evaluated considering any increased turbulent flow vibration and fretting. Change in lift on the fuel assembly and spacer grids was considered in confirming the adequacy of the existing holddown spring. Change in vertical LOCA loading as a result of the changes in fuel assembly and component pressure drop was evaluated. The seismic, fuel handling and shipping qualification is unaffected by the EPU. However, the current seismic loads were used in load combination with LOCA. The resulting effect on control rod insertability was evaluated.

Input Parameters, Assumptions, and Acceptance Criteria

In accordance with the BAW-10179PA and BAW-10133PA (References 5, 8 and 9) methodology, the various criteria for fuel damage and fuel rod failure, fuel coolability, and control rod insertability were screened for the EPU related factors. Each of the key design changes was then evaluated versus the applicable acceptance criteria. The objective of this fuel structural performance evaluation is to provide assurance that:

1. the fuel system is not damaged as a result of normal operation and AOOs,
2. fuel system damage is never so severe as to prevent control rod insertion when it is required,
3. the number of fuel rod failures is not underestimated for postulated accidents, and
4. fuel coolability is always maintained.

A "damaged" fuel system is defined as fuel rod functional capabilities that are reduced below the minimum required for the safe operation of the plant. Objective 1, above, is consistent with the FSAR Section 1.4.6 criterion, and the design limits that accomplish this are the SAFDLs. The term "fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10CFR50.67 for postulated accidents. Coolable geometry means that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after design basis accidents. Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR 50.46.

The results of the evaluation are presented here.

Description of Analyses and Evaluations

A series of mechanical and thermal analyses and evaluations that verify compliance with regulatory requirements was performed for Mark-B-HTP. The analyses that comprise the technical basis are per topical report BAW-10179PA (Reference 5).

The Mark-B-HTP fuel was evaluated for the EPU starting with CR-3 cycle 18. The fuel assemblies scheduled to be placed in-core for CR-3 EPU in conceptual cycles 18, 19, and 20 will be a full core of Mark-B-HTP fuel assemblies. No changes to Mark-B-HTP fuel design are proposed for CR-3 EPU. The design burnup for the Mark-B-HTP design is 62,000 MWd/mtU for the peak rod and for the CR-3 core is 60,000 MWd/mtU. A conservative evaluation was performed for the design life of equal to, or higher than, 62,000 MWd/mtU.

The current nuclear fuel design and design philosophy are used for the EPU evaluation. The approved methods and models (References 1 through 5, 7 through 12, and 14) that were used for the EPU are the same as those used for a typical reload. The approved fuel performance code COPERNIC (Reference 6) was used to obtain the M5 cladding corrosion evaluation.

Testing

Comprehensive test programs were conducted at AREVA NP's Global Test Facilities to characterize and verify the mechanical, thermal-hydraulic and flow-induced-vibration performance of the Mark-B-HTP fuel assembly design, according to the methods and criteria established in BAW-10179PA (Reference 5) and BAW-10133PA with Addenda 1 and 2 (References 8 and 9). These tests were performed prior to the EPU; however, due to the conservative nature of these tests, the results are also applicable to the EPU assemblies. The test program is also supplemented by in-reactor operation of more than 600 Mark-B-HTP and more than 1700 fuel assemblies utilizing the HTP grid and a FUELGUARD lower tie plate (end fitting) design features.

Fuel Assembly Mechanical and Performance Results

Fuel assembly mechanical performance was shown to satisfy all applicable structural criteria to maintain safe plant operation and maintain a coolable geometry under all plant design conditions of the CR-3 EPU. The Mark-B-HTP fuel assembly was evaluated to ensure safe and reliable operation under the loading associated with the normal operation, anticipated operational occurrences and postulated accident events per the methods and criteria established in Section 4 of BAW-10179PA (Reference 5). The seismic Safe Shutdown Earthquake (SSE) and LOCA are postulated accident events. The mechanical analysis demonstrates that the fuel assembly structure satisfies the requirements of the 10CFR50. NRC approved state of the-art methods and tools were used in the structural analyses. Methodologies for the fuel assembly faulted structural evaluations are described in NRC approved topical report BAW-10133PA with Addenda 1 and 2 (References 8 and 9). The fuel assembly structural evaluation was based on the

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Reference 13) as a guideline.

A series of analyses have been performed for the M5 fuel rod design to confirm it's in reactor mechanical performance. The stress, strain, fatigue, and buckling or collapse evaluation was performed for loadings associated with the following loads.

For the normal operation, AOOs and postulated accident events from BOL to EOL, the following were considered:

- Weight, pressure and hydraulic loads
- Thermal and irradiation growth
- Seismic (operational base earthquake and safe shutdown earthquake)
- Loss-of-Coolant Accident
- Flow induced vibration

Normal operation and anticipated operational occurrences are considered Normal Conditions in the stress analysis. Postulated accident conditions are considered as Faulted Conditions.

Margins reported are calculated by the following:

$$\% \text{ Margin} = \frac{\text{Allowable - Predicted}}{\text{Predicted}} \times 100$$

Given that the methods of analysis for each criterion and that the inputs used are conservative, any margin greater than 0% is sufficient.

Fuel Assembly Holddown Springs

The design bases for the Mark-B-HTP fuel holddown springs require that the springs be capable of maintaining fuel assembly contact with the lower core plate during normal operating conditions for maximum flow conditions, including a fourth pump startup.

The fuel assembly holddown spring analysis was performed on the Mark-B-HTP assembly using the statistical holddown spring methodology approved in BAW-10243PA (Reference 7). The net holddown forces on the fuel assembly throughout its design lifetime were evaluated taking into account fuel assembly growth and spring relaxation on a cycle-by-cycle basis. The analysis accounts for the opposing forces that act on each fuel assembly due to assembly weight, buoyancy, spring forces, and lift force. The analysis demonstrated that there was a positive net fuel assembly holddown force on the bottom core plate. This criterion is satisfied for the Mark-B-HTP eight-grid fuel assembly design under the CR-3 EPU conditions.

The Mark-B-HTP holddown spring assembly stress calculations show that the cruciform leaf holddown springs and the spring assembly components, including the spring nut, bolt, and retainer are structurally adequate under all static and fatigue loading conditions under plant design conditions of the CR-3 EPU.

Guide Tube

The guide tube loads were determined based on an axial model of the Mark-B-HTP fuel assembly, which was benchmarked to the fuel assembly and component mechanical tests. Conservative loads were evaluated considering maximum holddown loads, fuel rod slip loads, fuel assembly weight, the CR-3 EPU hydraulic loads, seismic and LOCA loads. Both BOL and EOL conditions were considered in the evaluation. {

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The post-EPU guide tube wear is the same as the pre-EPU condition. The design bases for the guide tube state that no buckling of the tubes shall occur during normal operation or any transient condition under which control rod insertion is required. There is adequate margin against guide tube buckling. In addition, all guide tube assembly components meet the allowable stress criteria.

Spacer Grids

The design bases for fuel assembly spacer grids require that no crushing deformations occur for normal operation and Operational Base Earthquake (OBE) conditions. The grids must also maintain sufficient geometry to ensure control rod insertability for SSE conditions. Grids must provide adequate support to maintain the fuel rods in a coolable configuration under all conditions, including SSE and LOCA conditions. The deformation limits must be consistent with the ECCS and safety analysis. {

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The mechanical design bases of the Mark-B-HTP spacer grids were confirmed through a series of static and dynamic tests. All testing indicates that the grids provide adequate design margins under plant design conditions of the CR-3 EPU.

The maximum grid impact forces for the SSE and SSE plus LOCA conditions were shown to be below the allowable elastic limits for all cases, including BOL, EOL and grid offset conditions. The Mark-B-HTP grids remain elastic and a core coolable geometry will be maintained for all the normal and faulted conditions post EPU.

Upper and Lower End Fittings

The upper and lower end fittings design bases follow those outlined in Section III of the ASME Boiler and Pressure Vessel Code (Reference 13). Finite-element analyses of the upper end fitting and the FUELGUARD lower end fitting, using the ANSYS computer code, were performed to show that the designs are adequate to withstand the normal operating loads under the EPU conditions. The loads used for these analyses were based on conditions including EOL hot operating and shutdown conditions, which provide the maximum holddown force. For the lower end fitting, the weight of the fuel assembly was also considered when analyzing the structural integrity of the grillage. The finite-element analysis shows the adequacy of the FUELGUARD debris-resistant lower end fitting design. The upper and lower end fitting stresses, for the Normal Operating and Faulted (including seismic SSE and LOCA) conditions of the EPU, have adequate margins against the allowables.

Connections

The Mark-B-HTP spacer grid-to-guide tube weld joint and the Mark-B HMP spacer grid capture ring-to-guide tube weld joint were tested to determine the stiffness and strength of the interfaces. The results of these tests, coupled with those of the HTP and HMP spacer grid/fuel rod slip load tests were used in the evaluation of the intermediate and end grid restraint systems. Using applicable ASME Code safety factors, the evaluation showed that sufficient margin exists for the weld joints under normal operation, faulted and handling conditions of the CR-3 EPU.

The resistance weld process utilized is a qualified procedure ensuring that consistent strength requirements are maintained. The guide tube upper collar and lock nut, and the guide tube lower end plug and cap screw connections, were verified through testing and/or analysis. Process qualifications are also performed for the upper collar lip weld and lock nut and lower cap screw crimp-type connections to ensure consistent strength requirements. The connections have adequate margins to allowable limits under the plant design conditions of the CR-3 EPU.

Fuel Rod Bow

Fuel rod bow is the deviation from straightness of the fuel rods in the fuel assembly. The presence of fuel rod bow is identified by the change in water channel gap from nominal conditions. The primary effects of rod bow are a decrease in the departure from nucleate boiling ratio (DNBR) and increase in local power peaking. The secondary effects of fuel rod bow include fuel clad fretting at full gap closure, though the probability of rod-to-rod fretting is very low.

The geometry of the fuel rods (outer diameter, inner diameter, fuel pellet diameter, and fuel rod pitch) in the Mark-B-HTP design is similar to that for fuel rods in fuel assembly designs residing in other 177-fuel assembly plants. Based on a review of the Mark-B-HTP fuel assembly design, the fuel rod bow has been shown to be bounded by the rod bow correlations from BAW-10147PA (Reference 12) under the plant design conditions of the CR-3 EPU.

Growth Allowance Evaluations

The axial clearance between core plates and the upper and lower end fittings should allow sufficient margin for fuel assembly growth during the assembly lifetime. The design basis for axial growth requires that adequate clearance be maintained between the fuel rod and the upper and lower end fittings to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly. The design basis for radial growth requires that adequate clearance be maintained between the fuel assemblies and reactor internals to accommodate the differences in the growth of fuel assembly and the growth of the reactor internals.

The fuel assembly and its components grow during operation. There are two components of the growth, the thermal expansion growth and irradiation growth. The average RCS temperature is increased by 3°F for the EPU conditions. This change is minor for assembly thermal expansion and insignificant relative to material property influence (yield strength, elastic modulus). The second growth is irradiation growth and is a function of burnup. The CR-3 post EPU design burnup value is unchanged from the pre-EPU value.

The irradiation growth curve for the Mark-B-HTP has been developed based on post irradiation measurements. The best estimate growth was determined from the Mark-B-HTP measured data and extrapolation to higher burnup. However, since the Mark-B-HTP measured data were available for a limited burnup range, the upper and lower 95/95 tolerance limit of the growth was conservatively determined from the measured data for Mark-B10K, Mark-B11A and Mark-B12 fuel assemblies. AREVA continually updates the applicable models as new data becomes available. Any design changes, due to updated assembly growth model, will be made following the normal design change process.

Growth allowance evaluations were performed for the Mark-B-HTP fuel assembly. The axial gap between the upper end fitting and reactor upper core plate was conservatively analyzed using the upper tolerance limit to show that these gaps allow sufficient margin to accommodate the fuel assembly growth.

For the shoulder gap, growth allowance evaluations were performed for the axial gaps between the upper end fitting grillage and fuel rods. The shoulder gap was conservatively analyzed using the upper tolerance limit and show that these gaps allow accommodating the fuel assembly and fuel rod growth up to fuel rod peak burnup of 62 GWd/mtU.

The spacer grid growth due to corrosion, thermal and irradiation effects has adequate margin against a solid-core situation under the plant design conditions of the CR-3 EPU.

Seismic and LOCA

The current licensing basis for seismic and LOCA is identified in FSAR Section 4.1 and BAW-10133PA (References 8 and 9). The Mark-B-HTP fuel structural integrity was evaluated in detail for faulted loads by means of testing, linear and nonlinear dynamic load analyses, and stress analyses.

Input Parameters, Assumptions, and Acceptance Criteria

Bases/Criteria - Earthquakes and postulated pipe breaks in the Reactor Coolant system would result in external forces on the fuel assembly. The acceptance criteria for the seismic loading design are that fragmentation of the fuel rod must not occur as a result of the seismic loads, and control rod insertability and coolable geometry must be maintained. The principal acceptance criteria for a LOCA event are that fragmentation of the fuel rod must not occur as a direct result of the blowdown load, and control rod insertability and coolable geometry must be maintained.

Description of Analyses and Evaluations

Fuel assembly responses resulting from seismic excitation and LOCA were analyzed in accordance with topical report BAW-10133PA, Addendum 1 (Reference 9). Seismic displacement time histories at the upper and lower core plates and at the upper end of the baffle plate for B&W 177 fuel assembly (FA) plants including CR-3 were used.

The static lateral stiffness and dynamic natural frequency and damping characteristics of the Mark-B-HTP fuel assemblies were determined experimentally and used to benchmark the fuel assembly lateral model. The grid crush strength was established based on analysis of the 95% confidence level on the true mean of the test data at operating temperature. The analysis parameters - natural frequencies, mode shapes, and span masses of the fuel assembly combined with the structural damping were used to generate a simplified lumped-mass-spring fuel assembly model. The number of fuel assemblies, and the gap clearances between fuel-assemblies and at the baffles were used to generate the reactor internals model.

The LOCA and SSE analyses were performed using the time-history numerical integration technique. The maximum grid impact forces obtained from both were combined using the square root of the sum of squares (SRSS) method. The maximum loads were compared with the allowable grid crush strength.

For LOCA Analysis, all B&W 177 FA plants, including CR-3, are qualified for Leak-Before-Break, (Reference 10). The resulting LOCA displacement time histories evaluated include those associated with a worst-case attached pipe break (i.e., core flood line, decay heat line and surge line breaks).

In the accident analyses, the lateral effect (LOCA and seismic) and the vertical effect (LOCA) were analyzed separately. The post-EPU seismic loads are the same as the pre-EPU condition.

CR-3 EPU will have a full core of Mark-B-HTP fuel assemblies. Fuel assembly models were combined to represent the row configurations in the core. The shortest row in the core has 5 assemblies and the largest has 15 assemblies. Row models with 5 and 15 assemblies were created. The impact forces under a horizontal seismic and LOCA loading were calculated for the given core configurations.

Vertical LOCA Analysis

A vertical analysis was performed to determine the limiting component loads during a LOCA event. The LOCA analysis uses the post-EPU hydraulic loads and core plate motions. The limiting loads were then combined with SSE loads for faulted condition evaluation for component structural integrity, including the guide tubes in order to maintain control rod insertability during a LOCA. The vertical LOCA method per Reference 9, in conjunction with the general-purpose finite-element program ANSYS, was used in the analysis. The methodology distributes the given hydraulic force time histories for various pipe breaks over a large number of mass nodes within the ANSYS model.

The component forces were obtained from the analysis. The guide tube critical buckling is the limiting criterion for the vertical LOCA condition. For conservatism, a load factor of 1.2 was used on the guide tube load to account for unequal loading due to external factors, fabrication differences and inherent design factors. The guide tube buckling load limit calculated in the analysis is conservative since it is based on the column secant formula, in terms of compressive stress, rather than using classical Euler’s buckling theory. From this analysis, it was confirmed that the forces on the guide tube are below conservatively calculated allowable loads under the plant design conditions of the CR-3 EPU.

Also, under the plant design conditions of the CR-3 EPU the guide tube stresses resulting from the fuel assembly deflection and axial load were calculated. These forces were well below conservatively calculated allowable loads for the guide tubes and fuel rods. The results of the analysis also showed that the fuel assembly does not impact the upper core plate during the LOCA.

Fuel Assembly Component Stresses Under Faulted Condition Loads

The Mark-B-HTP fuel assembly faulted condition component stress analyses were performed using axial and lateral loads generated from seismic and LOCA loading analyses. The loads for the worst case LOCA break were conservatively combined with those of the SSE using square root of sum of squares to determine maximum fuel assembly loads.

The component stress intensity limits for the components are based on the Level D service limit of the Section III of the ASME Code. The component analysis and determination of failure loads were conducted utilizing both standard engineering techniques and testing. The design margin results indicate that all components of the Mark-B-HTP fuel assembly meet the design requirements for the SSE and SSE plus LOCA loading events under the plant design conditions of the CR-3 EPU.

Seismic and LOCA Results

The maximum grid loads obtained from SSE and LOCA loading analyses were combined using the SRSS method. The results of the combined seismic and LOCA analyses indicate that the maximum impact forces for the Mark-B-HTP eight-grid assembly design using the two-direction grid characteristics were less than the respective allowable grid strengths. The allowable grid strengths were established at the 95% confidence level on the true mean from the distribution of experimentally determined grid crush data at temperature. The Mark-B-HTP fuel assembly component design shows positive margin for guide tube buckling and faulted condition stresses. The fragmentation of fuel rods will not occur. Core coolable geometry requirements are met. The reactor can be safely shutdown under the combined faulted-condition loads. The conservatism inherent in the criteria and methodology indicate that the fuel design has sufficient margin as shown by comparison of calculated results to allowable criteria for safe plant operation and coolable geometry for the CR-3 EPU.

Fuel Rod Performance

Fuel rod performance for CR-3 Mark-B-HTP fuel is shown to satisfy the fuel rod design bases as described in FSAR Section 3. Compliance with reload cycles is confirmed via the approved reload methodology of BAW-10179PA (Reference 5). Compliance with fuel rod design criteria is confirmed by approved methodology of BAW-10133PA and BAW-10227PA (References 8, 9, and 11).

The fuel assemblies evaluated for CR-3 EPU in conceptual cycles 18, 19, and 20 will be a full core of Mark-B-HTP fuel assemblies which contain M5 alloy fuel rod cladding and M5 rod end caps. Use of the alloy M5 material for fuel rod cladding permits higher burnup and EPU of the fuel. All calculations support the use of the Mark-B-HTP M5 fuel rod assemblies to a peak pin burnup of 62 GWd/mtU.

Description of Fuel Rod Analyses and Evaluations

A series of analyses were performed for the Mark-B-HTP M5 fuel rod design to confirm the reactor mechanical performance under EPU conditions. The areas that were analyzed include:

- Cladding Stress and Collapse Load
- Cladding Strain
- Cladding Fatigue
- Cladding Creep Collapse
- Cladding Corrosion / Oxidation
- Fuel Rod Fretting Wear
- Fuel Rod Axial Growth
- Centerline Fuel Melt
- Fuel Rod Internal Gas Pressure

The current nuclear fuel design and design philosophy were used for the EPU evaluation. The approved methods and models (References 1 through 5, 7 through 12 and 14) that were used for the EPU are the same as those used for a typical reload. The approved fuel performance code COPERNIC (Reference 6) was used to obtain the M5 cladding corrosion evaluation.

In early, 2009 the NRC informed US fuel suppliers of its concern regarding adequacy of fuel rod models used in legacy codes by all of the US fuel suppliers. Specifically, the question of proper accounting for the burnup degradation of fuel thermal conductivity (reduction of thermal conductivity resulting from the creation of fission products during irradiation) was raised. As a result, the NRC informally asked AREVA NP to evaluate the broad issue of how thermal conductivity degradation is addressed in the current approved fuel rod thermal performance models. The NRC issued an Information Notice (IN) 2009-23 later in 2009 requesting recipients to review the information within the IN for applicability to their fuel performance codes; no specific action was required by the IN.

AREVA NP conducted a formal assessment to address this issue. The results indicated that previous adjustments were not sufficient to offset the degradation indicated with recent data. Augmentation factors were derived for TACO3 (Reference 1) and GDTACO (Reference 2) to correct cladding strain and centerline fuel melt limits for the effects of degradation of fuel thermal conductivity. Fuel rod internal gas pressure analyses were examined and were determined to not be adversely impacted.

Additionally, an investigation was performed for deterministic loss-of-coolant accident (LOCA) analyses for those plants dependent on TACO3/GDTACO code methodology in their Emergency Core Cooling System (ECCS) analyses. It was concluded that the methods used to perform LOCA fuel temperature initializations, when applied together with the TACO3 predicted LOCA initialization fuel rod temperatures, adequately account for the degraded thermal conductivity with burnup for any PCT-limited analysis. The similarities between the TACO3 and GDTACO codes and methods allow the same conclusion to be extended to GDTACO. For more information on LOCA and ECCS evaluations for the EPU, see Section 2.8.5.6.3.

The AREVA NP new generation fuel performance code COPERNIC fully accounts for thermal conductivity degradation with burn up. COPERNIC is currently in the licensing basis for CR-3 by virtue of its inclusion in the approved safety criteria and methodology Topical Report BAW-1 0179P-A. For ECCS evaluations, the TACO3 and GDTACO codes and methods will continue to be used in the CR-3 licensing basis.

Cladding Stress and Collapse Load

Bases/Criteria - The specified acceptable fuel design limits fuel damage criteria for cladding stress should ensure that Fuel System Design dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. The fuel rod cladding design basis is that the Fuel System will be functional and will not be damaged due to excessive stresses. These criteria are based on guidelines established in Section III of the ASME Boiler Pressure Vessel Code (Reference 13) along with the approved methodology of BAW-10227PA (Reference 11).

Evaluation - The fuel rod cladding was analyzed for the stresses induced during operation. Conservative values were used for cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure, differential temperature, and unirradiated cladding yield strength. The low yield strength of the recrystalized annealed (RXA) M5 cladding does not allow for the use of the ASME-based stress intensity limits for compressive stresses. With significant fast neutron fluence the cladding strength will increase to much higher levels, and the generic stress calculations will be enveloping. The worst-case tensile stress condition occurs late in the life of the fuel rod and is enveloped by the generic stress calculations for the hardened cladding. The resulting stress categories were divided into compressive stresses and tensile stresses according to the criteria of (Reference 11). The fuel rod stress analysis calculates the worst-case cladding stress state based on the thinnest clad wall and largest cladding ovality. The likelihood of these conditions occurring at the same location on the cladding is remote. Therefore, the use of the two conditions together to calculate the cladding stress state is conservative.

The types of stresses analyzed were as follows:

- Pressure Stresses - These are membrane stresses due to the external and internal pressure on the fuel rod cladding.
- Flow Induced Vibration Stresses - These are longitudinal (axial) bending stresses due to vibration of the fuel rod. The vibration is caused by coolant flow around the fuel rod.
- Ovality Stresses - These are bending stresses due to external and internal pressure on the fuel rod cladding that is oval due to initial manufacturing tolerances. The stresses resulting from creep ovalization into an axial gap are discussed in 'Cladding Creep Collapse' section.
- Thermal Stresses - These are secondary stresses that arise from the temperature gradient across the fuel rod during reactor operation.
- Fuel Rod Growth Stresses - These secondary stresses are due to the fuel rod slipping through the spacer grids. These may be due to the fuel assembly expanding more than the fuel rod due to heat-up and/or they may be due to fuel rod growth from irradiation.
- Fuel Rod Spacer Grid Interaction Stresses - These are localized stresses due to contact between the fuel rod cladding and the spacer grid stops.

The fuel rod stresses for the normal operating and faulted conditions, including seismic SSE and LOCA conditions, have adequate margin against allowable stresses. Therefore, fuel rod fragmentation will not occur. The fuel cladding collapse under external pressure has adequate margin. The fuel rod has adequate margin against buckling for both normal operating and faulted condition loads. Thus, the core cooling geometry will be preserved.

Cladding Strain

Bases/Criteria - The design criterion for fuel rod cladding strain is that the maximum uniform hoop strain (elastic plus plastic) shall not exceed 1%.

Evaluation - The analysis was conducted using the NRC approved fuel rod thermal analysis computer programs TACO3 (Reference 1) and GDTACO (Reference 2). With the EPU, there is an increase in the core average linear heat generation rate. The nominal core average linear heat rate (LHR) for the EPU is 6.684 kW/ft. Enveloping power history curves for the CR-3 EPU urania and gadolinia rods were developed which limit the 1% cladding strain rate at 65 and 62 GWd/mtU burnup, respectively. The CR-3 conceptual fuel cycles 18, 19 and 20 power histories for each batch of fuel was compared against these envelopes. The comparison shows that the CR-3 EPU power histories were below these envelopes.

The Mark-B-HTP fuel rod was analyzed to determine the maximum transient LHR the fuel rod cladding could experience before the transient strain limit of 1% is exceeded. The transient strain limit uses cladding circumferential changes before and after a linear heat rate transient to determine the strain. Transient axial flux shapes were used in the derivation. The resulting transient LHR limits are used in the maneuvering analysis to determine axial power imbalance protective limits and allowable nuclear overpower and axial power imbalance.

Cladding Fatigue

Bases/Criteria - The design criterion for cladding fatigue is that the cumulative fatigue usage factor be less than 0.9 when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is the most conservative, is imposed as per the O'Donnell and Langer design curve for fatigue usage.

Evaluation -The fuel rod was analyzed for the total fatigue usage factor using the approved methodology of Reference 11 and the procedures outlined in the ASME Code. Testing has been conducted by AREVA NP to determine the fatigue performance of M5 cladding. These tests have shown similar fatigue endurance performance for RXA claddings as compared to Zircaloy-4 with the lower yield strength of the RXA claddings limiting the applied stresses. The values for alternating stress versus number of cycles obtained are well enveloped by the standard O'Donnell-Langer design fatigue curve (Reference 15) for irradiated Zircaloy-4. For the fatigue analysis, a fuel rod in-core operating life of 10 calendar years is used. This fuel rod life bounds the planned exposure of the fuel at the CR-3 EPU. The possible normal and anticipated operational occurrence condition events were analyzed to determine the total fatigue usage factor experienced by the fuel rod. Conservative inputs in terms of cladding thickness, oxide layer buildup, external pressure, internal fuel rod pressure and differential temperature across the cladding were assumed for the analysis. The results of the fatigue analysis for the CR-3 EPU Mark-B-HTP fuel rod show a maximum fatigue usage factor of { }.

Cladding Creep Collapse

Bases/Criteria - If axial gaps in the fuel pellet column were to occur due to fuel densification, the potential exists for the cladding to collapse into the gap. Because of the large local strains that would result from collapse, the cladding is then assumed to fail. The fuel rod design criterion is that cladding collapse is precluded during the fuel rod design lifetime.

Evaluation – The CR-3 EPU Mark-B-HTP fuel rods were analyzed for creep collapse lifetime using the NRC-approved method BAW-10084PA (Reference 3). The acceptance criterion is that the predicted creep collapse life of the fuel rod must exceed the maximum expected in core life. The fuel rod will fail due to creep collapse when either of the following happens:

- The rate of creep ovalization exceeds 0.1 mils/hr.
- The maximum fiber stress exceeds the yield strength of the cladding.

Creep collapse lifetime analysis was performed using conservative assumptions on fuel rod pre-pressure, fission gas release, pellet densification, cladding thickness and ovality and using a bounding power history. The results show that creep collapse lifetime of the Mark-B-HTP fuel rod design for the CR-3 EPU exceeds the design lifetime and thus this criterion is met for the CR-3 EPU.

Cladding Corrosion / Oxidation

Bases/Criteria – The fuel rod maximum acceptable predicted oxide thickness limit is 100 microns (Reference 4).

Evaluation - The corrosion reduces the cladding thickness by converting base metal to an oxide, thus decreasing margin to mechanical failure. Oxidation reduces the ductility and thermal conductivity of the fuel rod cladding, thus limiting the fuel rod mechanical performance. Corrosion leads to cladding hydrogen uptake that occurs as a by-product of oxidation. After the cladding oxide layer thickness exceeds a certain threshold, spalling can occur, which leaves localized cold spots due to losses of cladding and oxide. The combined effects of oxidation can significantly reduce the thermal and mechanical performance of the fuel rod cladding.

The Mark-B-HTP fuel rod uses M5 cladding which has been shown through in-reactor testing to exhibit acceptable performance in terms of oxide film thickness growth and hydrogen uptake. This cladding material has demonstrated superior ductility at extended burnup to meet current NRC licensing criteria, and it has been extensively tested with in-core and out-of-core tests to show its resistance to spallation.

The maximum oxide thickness for the highest burnup rod in each core was predicted using BAW-10231PA (Reference 6). The predicted maximum cladding oxide thickness level for the best estimate enveloping power history is 33 microns. Using conservative bounding power history, the predicted maximum oxide thickness is 58 microns. Since the cladding oxide thickness is less than 100 microns, the acceptance criteria is met. This level of corrosion will not adversely affect the structural integrity of the fuel rod during its CR-3 EPU design lifetime. The maximum hydrogen level is 171 ppm. At this oxide thickness and hydride content spalling does not occur and there is no significant loss of cladding ductility during reactor operation due to hydrogen content.

Fuel Rod Fretting Wear

Bases/Criteria - Fretting wear is a concern for fuel rods and guide tubes. Fretting, or wear, may occur on the fuel in contact with the spacer grids if there is a reduction in grid spacing loads in combination with flow induced vibratory forces. The design criterion for fretting wear is that the assembly design shall provide sufficient support to limit rod vibration and fretting wear.

Evaluation - The Mark-B-HTP fuel rod fretting wear performance has been verified based on the proven performance of more than 600 Mark-B-HTP fuel assemblies and 1700 HTP design types utilizing the FUELGUARD type lower end fitting.

The life and wear test comprised a 1000 hour endurance test in an environment representative of in-reactor conditions. The testing showed no deleterious axial flow effects. Life and wear testing showed no discernable fuel rod fretting wear for grid cell conditions considered bounding for EOL. This test was performed pre-EPU. Due to conservative nature of life and wear test, the test results are also applicable to the post-EPU conditions.

The Mark-B-HTP flow and fretting test was comprised of a full scale EOL Mark-B-HTP fuel assembly in the PETER Loop under simulated in-reactor baffle slot and LOCA hole environments of B&W 177 fuel assembly plants. The flow test was followed by a single rod fretting test in an autoclave test facility using bounding rod amplitude measured from full scale flow test. At the end of 1000 hours the rod wear was measured. For conservative EOL condition with gapped support rod, the measured wear was { } and this fretting wear is so small to conclude that the fretting failure wear will not occur during expected lifetime of the Mark-B-HTP.

The post-EPU guide tube wear is the same as the pre-EPU conditions.

Fuel Rod Axial Growth

Bases/Criteria - The design basis for axial growth is that adequate clearance be maintained between the rod ends and the top and bottom nozzles to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly.

Evaluation - This criterion ensures that there is sufficient axial space, known as shoulder gap, to accommodate the maximum expected fuel rod growth. Fuel rods are designed with adequate clearance between the fuel rod and the upper and lower end fittings to accommodate the differences in the growth of fuel rods and the growth of the fuel assembly skeleton to preclude interference of these members. The CR-3 EPU fuel rod growth evaluation, based on measured growth data, demonstrates that there is adequate margin to the fuel rod growth design limit for the Mark-B-HTP fuel.

Fuel Rod Thermal Performance

Bases/Criteria - To operate the core safely at the CR-3 EPU conditions, the fuel must not exceed melt temperatures and fuel rod internal gas pressures must meet licensed limits during operation in the core.

Evaluation - The codes and methodologies relevant to fuel thermal performance were reviewed and found to be applicable to CR-3 at the core conditions that will follow the EPU.

The fuel thermal performance was evaluated using the approved TACO3 (Reference 1) and GDTACO (Reference 2) computer programs. TACO3 was used for fuel rods containing urania fuel whereas GDTACO was used for fuel rods containing urania-gadolinia fuel. The models and methods found in both programs are essentially identical. The principal difference is that the properties and power profile for the fuel pellet have been modified in GDTACO to reflect the presence of gadolinia. Therefore, the discussion of TACO3 that follows applies to GDTACO as well.

TACO3 is a best-estimate code that was used to determine the fuel linear heat generation rates that cause centerline fuel melting at beginning-of-life (BOL) and at any time-in-life (TIL). The best-estimate fuel melt temperature used in TACO3 is reduced to ensure that there is a 95 percent probability at 95 percent confidence that the fuel will not melt. This melt temperature limit was further reduced during the TIL analysis to account for fuel burnup effects. The limit effectively accounts for code and manufacturing uncertainties, transient fission gas release, and cladding oxide effects.

TACO3 was also used to evaluate fuel rod internal gas pressures. When used in this mode, code uncertainties, power history uncertainties, and manufacturing variations were considered and are used to calculate an internal pressure that is a conservative (95 percent probability at 95 percent confidence) bounding value. This bounding pressure value can be no greater than the system pressure plus an AREVA NP proprietary pressure increment (Reference 14). In addition to this constraint, there can be no fuel-clad liftoff for significant linear heat generation rates. That is, the fuel to clad gap cannot open up, which would result in gap heat transfer degradation and the exacerbation of fuel rod pressures and temperatures.

Centerline Fuel Melt

Analyses were performed to determine the linear heat generation rates that define the centerline fuel melt limit at BOL and at TIL for the CR-3 EPU conditions. Analyses were performed for Mark-B-HTP assemblies containing only urania as well as for Mark-B-HTP assemblies containing urania and gadolinia (gadolinia assemblies).

Results of the analyses for the Mark-B-HTP urania fuel yield a limiting linear heat rate to melt (LHRTM) at EOL with a linear heat generation rate of 22.2 kW/ft at the melt temperature limit. The burnup at this limiting condition is approximately 60,000 MWd/mtU, which bound the maximum EOL burnup for urania fuel in the fuel cycle study.

Analyses of the Mark-B-HTP gadolinia fuel consider gadolinia concentrations of three, four, six, and eight percent by weight. Fuel melt limit heat rates decrease with increasing gadolinia content. The limiting LHRTM for the eight-weight-percent gadolinia assemblies occur at EOL, with a burnup of approximately 50,000 MWd/mtU, at a linear heat generation rate of 20.4 kW/ft at the melt temperature limit. The 50,000 MWd/mtU burnup and 20.4 kW/ft LHR bounds the maximum gadolinia fuel EOL values in the fuel cycle study.

Table 2.8.1-2 summarizes the CR-3 EPU LHRTM results and also provides comparison to the pre-EPU results. The resulting CFM LHR limits are used in the maneuvering analysis to establish the axial power protective imbalance limits.

Fuel Rod Internal Gas Pressure

Analyses were performed to evaluate the EOL internal gas pressure of Mark-B-HTP urania fuel rods operating at the EPU conditions. The maximum EOL burnup is 62,000 MWd/mtU. Bounding and/or conservative inputs were used for power histories and axial flux shapes, reflecting conservative axial offset limits and a conservative maximum rod insertion limit. Analysis results show that the EOL fuel rod internal gas pressure remains below the licensed pressure limit.

Internal gas pressures at EOL (62,000 MWd/mtU) were also evaluated for Mark-B-HTP gadolinia fuel rods. For the range of gadolinia concentrations considered, EOL internal gas pressure remains below the licensed pressure limit.

Table 2.8.1-3 summarizes the post-EPU results and also provides the comparison with the pre-EPU licensed pressure limit.

Results

The fuel rod design analysis was performed on a cycle-specific basis. The analyses presented here were based on the bounding high-temperature nuclear design cases representing three conceptual fuel cycles at the EPU conditions developed for the Nuclear Design. Fuel rod design evaluations for the Mark-B-HTP fuel were performed using NRC-approved models and NRC-approved design criteria methods (Reference 5) to demonstrate that all fuel rod and fuel assembly mechanical design criteria are satisfied.

Fuel performance evaluations for the Mark-B-HTP fuel demonstrate that the design criteria can be satisfied for the fuel rod under the planned EPU operating conditions of a power uprate to 3014 MWt.

Fuel thermal performance analyses performed with TACO3 and GDTACO show that fuel will not exceed melt temperatures and fuel rod internal gas pressures will meet limits during operation in the core.

Each of these key fuel rod design criteria has been evaluated for application to the AREVA NP Mark-B-HTP fuel assembly design in CR-3. It is concluded that each design criterion was satisfied for the Mark-B-HTP design for the EPU conditions.

2.8.1.3 Conclusion

CR-3 has reviewed the analyses related to the effects of the proposed EPU on the Fuel System Design of the fuel assemblies, control systems, and reactor core. CR-3 concludes that the analyses has adequately accounted for the effects of the proposed EPU on the Fuel System and demonstrated that (1) the Fuel System will not be damaged as a result of normal operation and AOOs, (2) the Fuel System damage will never be so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures will not be underestimated for postulated accidents, and (4) coolability will always be maintained. Based on this, CR-3 concludes that the Fuel System and associated analyses will continue to meet the requirements of 10 CFR 50.46 and FSAR Sections 1.4.6, 1.4.28, 1.4.29, 1.4.30, 1.4.37, 1.4.41, 1.4.42, and 1.4.44. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Fuel System Design.

2.8.1.4 References

1. BAW-10162P-A (Proprietary), “TACO3 - Fuel Pin Thermal Analysis Computer Code.’

2. BAW-10184P-A (Proprietary), “GDTACO - Urania Gadolinia Fuel Pin Thermal Analysis Code.”

3. BAW-10084P-A, Rev. 3 (Proprietary), “Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse.”

4. BAW-10186P-A, Rev. 2 (Proprietary), “Extended Burnup Evaluation.”

5. BAW-10179P-A, Rev. 7 (Proprietary), “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.”

6. BAW-10231P-A, Rev. 1 (Proprietary), “COPERNIC Fuel Rod Design Computer Code.”

7. BAW-10243P-A (Proprietary), “Statistical Fuel Assembly Hold Down Methodology.”

8. BAW-10133P-A, Rev. 1 (Proprietary), “Mark-C Fuel Assembly LOCA-Seismic Analyses.”

9. BAW-10133P-A, Rev. 1 (Proprietary), Addendum 1 and Addendum 2, “Mark-C Fuel Assembly LOCA Seismic Analyses.”

10. BAW-1847, Rev. 1, “Leak-Before-Break Evaluation of Margin Against Full Break for RCS Primary Piping of B&W Designed NSS.”

11. BAW-10227P-A, Rev. 1 (Proprietary), “Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor Fuel.”

12. BAW-10147P-A, Rev. 1 (Proprietary), “Fuel Rod Bowing in Babcock & Wilcox Fuel Designs.”

13. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 2000 Edition.

14. BAW-10183P-A (Proprietary), “Fuel Rod Gas Pressure Criterion.”

15. “Fatigue Design Basis for Zircaloy Components,” by W.J. O'Donnell and B.F. Langer, Nuclear Science and Engineering, Volume 20, 1964.

Table 2.8.1-1: Mark-B-HTP Fuel Design Summary

Parameter	Mark-B-HTP
Pellets	
Fuel Pellet Material	Enriched UO_2 Gadolinia
Fuel Pellet Diameter, in	0.3735
Fuel Pellet Initial Density, %	96
Fuel Rods	
Fuel Rod Length, inch	155
Fuel Rod Cladding Material	M5
Fuel Rod Inside Diameter, in	0.38
Fuel Rod Outside Diameter, in	0.43
Active Fuel Column, in	143
Maximum Fuel Rod Burnup, MWd/mtU	62,000
Fuel Assembly	
Fuel Assembly Overall Length, in	166
Lattice Geometry	15 x 15
Fuel Rod Pitch, in	0.568
Number of Fuel Rods per Assembly	208
Number of Guide Tubes	16
Guide Tube Material	M5
Number of Instrument Tube	1
Instrument Tube Material	M5
Number of Spacer Grids	8
Upper End Grid Material and Design	M5 / HTP
Intermediate Grids Material and Design	M5 / HTP
Lower End Grid Material and Design	Alloy 718 / HMP

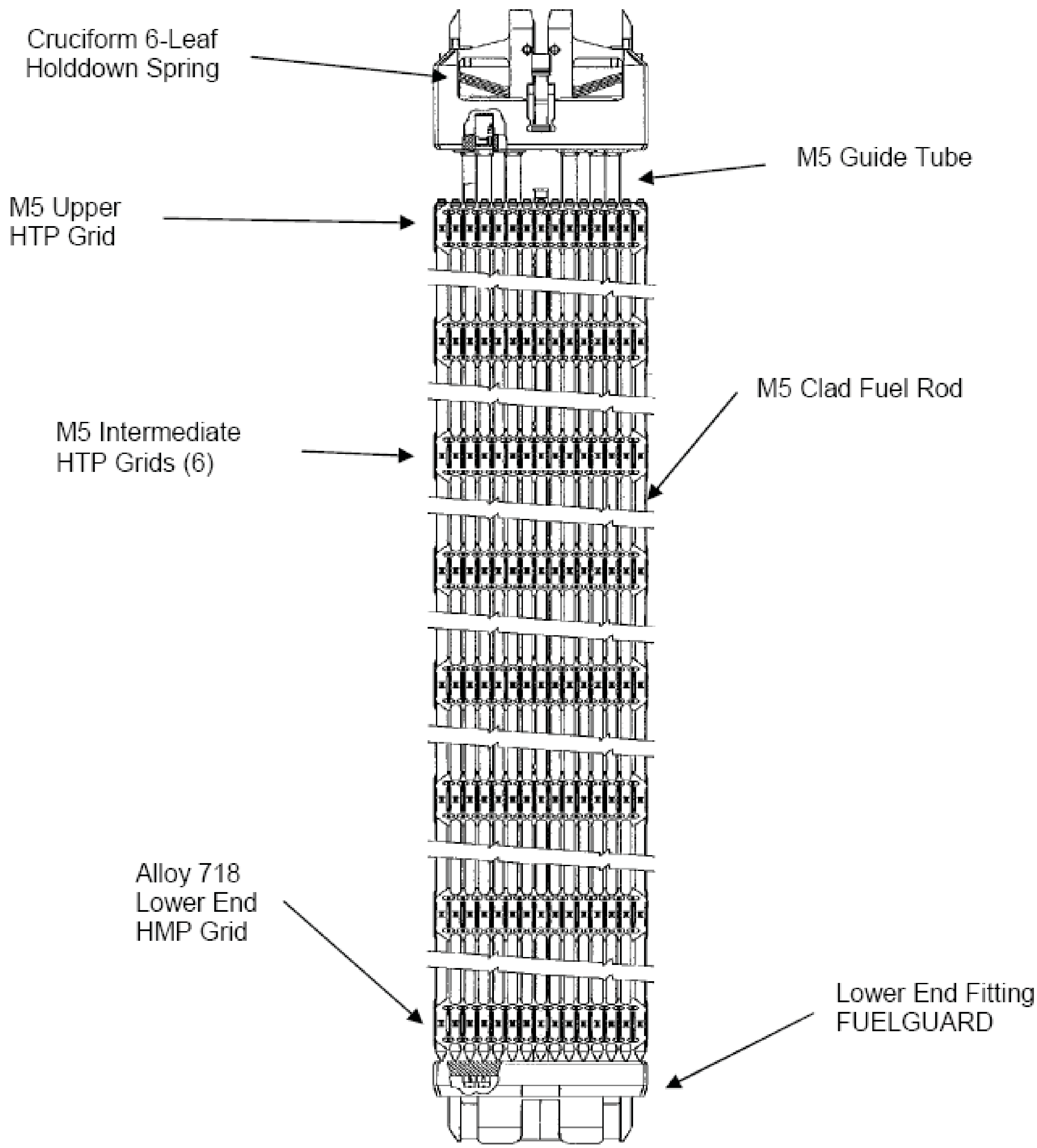
Table 2.8.1-2: Linear Heat Rate to Melt for Urania and Gadolinia Fuel Rods

	LHR Limits (kW/ft)				
	Urania Fuel Rod	Gadolinia Fuel Rod			
		3 wt %	4 wt %	6 wt %	8 wt %
Pre-EPU	{ }	{ }	{ }	{ }	{ }
Post-EPU	{ }	{ }	{ }	{ }	{ }

Table 2.8.1-3: Licensing Above System Pressure Limits for Urania and Gadolinia Fuel Rods

	Fuel Rod Average Burnup GWd/mtU				
	Urania Fuel Rod	Gadolinia Fuel Rod			
		3 wt %	4 wt %	6 wt %	8 wt %
Pre-EPU	{ }	{ }	{ }	{ }	{ }
Post-EPU	{ }	{ }	{ }	{ }	{ }

Figure 2.8.1-1: Mark-B-HTP Fuel Assembly



2.8.2 Nuclear Design

2.8.2.1 Regulatory Evaluation

A review was performed for the Nuclear Design of the CR-3 fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary (RCPB) or impair the capability to cool the core. The review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation.

The NRC's acceptance criteria for the Nuclear Design are based on:

- GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs);
- GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity;
- GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed;
- GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges;
- GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important-to-safety under accident conditions;
- GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems;
- GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes;
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and

- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the reactor core be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences [GDC-10];
- FSAR Section 1.4.7, Suppression of Power Oscillations, insofar as it requires that the reactor core be designed to ensure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can be reliably and readily detected and suppressed [GDC-12];
- FSAR Section 1.4.8, Overall Power Coefficient, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity [GDC-11];
- FSAR Section 1.4.12, Instrumentation and Control Systems; and FSAR Section 1.4.13, Fission Process Monitors and Controls, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges [GDC-13];
- FSAR Section 1.4.14, Core Protection Systems; and FSAR Section 1.4.15, Engineered Safety Features Protection Systems; insofar as it requires that the protection system be designed to automatically initiate the reactivity control systems to ensure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to automatically initiate operation of systems and components important-to-safety under accident conditions [GDC-20];
- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes [GDC-26];
- FSAR Section 1.4.28, Reactivity Hot Shutdown Capability; FSAR Section 1.4.29, Reactivity Shutdown Capability; and FSAR Section 1.4.30, Reactivity Holdown Capability, insofar as it requires that the reactivity control systems be designed to have a combined capability, in

conjunction with poison addition by the Emergency Core Cooling System, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to ensure the capability to cool the core is maintained [GDC-27];

- FSAR Section 1.4.31, Reactivity Control Systems Malfunction, insofar as it requires that the protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems [GDC-25]; and

- FSAR Section 1.4.32, Maximum Reactivity Worth of Control Rods, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core [GDC-28].

2.8.2.2 Technical Evaluation

<u>Introduction</u>

The Nuclear Design basis for the reload core nuclear design is defined in FSAR Sections 3.1.2.1, 3.1.2.2, and 3.2.2. The CR-3 EPU Nuclear Design analysis includes the development of: fuel cycle design models at uprated power conditions, key neutronic safety parameters important to reload safety analysis requirements, and core power distribution analysis. The purpose of the CR-3 EPU Nuclear Design analysis is to determine prior to the cycle-specific reload design if previously used values for key safety parameters remain applicable for the plant power uprate. The EPU nuclear design is also evaluated with respect to axial power oscillation suppression, available margins to power peaking limits, and the impact on COLR limits related to power distribution. The results of these design analyses are evaluated with respect to the design criteria listed in Section 2.8.2.1.

The NRC approved methods used in the Nuclear Design analysis portions of the EPU evaluation are described in Reference 1. These approved methods are used for CR-3 and other B&W 177 fuel assembly core reload designs. No changes to the nuclear design analysis methods (Reference 1) are necessary for the EPU with the exception of the control rod ejection accident methodology. Reference 2 provides the EPU control rod ejection accident methodology, which has been approved by the NRC for use at CR-3. No changes to the Nuclear Design codes (References 3 and 4) are necessary for the EPU.

The reload evaluation methodology described in Reference 1 identifies the key safety parameters and core power distribution requirements necessary for the FSAR safety analysis and the reload design analysis. These key safety parameters and core power distributions continue to be evaluated for each CR-3 reload cycle. The results of the key safety parameter evaluations and core power distribution evaluations are documented in the Reload Report and COLR for each cycle. The Reload Report is used to update the FSAR for each reload cycle. Reload licensing analyses also support development of 10CFR50.59 evaluations for each reload cycle.

Methodology Change for Control Rod Ejection Accident

The consequences of the rod ejection accident (REA) at power are affected by the EPU. The methodology in Reference 2 is employed to evaluate the fuel consequences of such an event which changes both the codes used and the criteria to be met. Section 2.8.5, Non-LOCA Analyses, contains a more complete safety evaluation description of the REA. A checklist of key physics parameters to validate the analysis is developed to verify its applicability to each reload analysis.

Description of Analyses and Evaluations

Nuclear Design characteristics are evaluated for three CR-3 conceptual fuel cycles at the EPU conditions using the Mark-B-HTP fuel assembly design described in Section 2.8.1. These fuel cycles form the basis for the evaluation of key safety parameters, core power distributions, and Technical Specifications/COLR impact.

Standard nuclear design analytical models and methods (References 1, 3, and 4) accurately describe the neutronic behavior of the CR-3 EPU Nuclear Design and the Mark-B-HTP fuel design. The specific design bases comply with NRC approved design criteria listed in Section 2.8.2.1. The analytical models and methods and the Mark-B-HTP fuel assembly design are currently used in operating CR-3 cycles. The design burnup for the Mark-B-HTP design is 62,000 MWd/mtU for the peak rod (Reference 5).

The key safety parameters evaluated for the EPU fuel cycle designs are bounded by values and limits used in the FSAR safety analysis (Section 2.8.5, Non-LOCA Analyses). Core power distribution analysis quantifies margin that exists for operating limits. Results from this Nuclear Design analysis are evaluated for impact on Improved Technical Specification and COLR requirements, and FSAR revision for each EPU reload cycle.

EPU Fuel Cycle Designs

CR-3 EPU fuel cycle design analyses are performed using the CASMO-NEMO code system (References 3 and 4). The thermal-hydraulic inputs to the NEMO code are adjusted for the power level increase. A fuel management strategy that is capable of achieving two-year fuel cycles with high capacity factors is modeled for three conceptual fuel cycles, the last of which represents a near-equilibrium fuel cycle. The burnable absorber used in these fuel cycles is gadolinia (Gd_2O_3), the same integral burnable absorber that is used in the current CR-3 fuel cycles. The designs are based upon the EPU power level of 3014 MWt, an uprate from the current cycle at 2609 MWt RTP. These EPU fuel cycles form the basis for the evaluation of key safety parameters, core power distribution analysis, and impact to Technical Specification/COLR requirements. These fuel cycles are not intended to represent limiting loading patterns, but are instead developed with the intent to show that enough margin exists between typical safety parameter values and the corresponding limits to allow flexibility in designing the future EPU reload cores. The actual cycle-specific core design is evaluated each cycle to confirm the design limits are maintained.

CR-3 transition and near equilibrium fuel cycle characteristics for CR-3 EPU are provided in Table 2.8.2-1.

EPU Nuclear Parameter Analysis

Nuclear Design aspects of control systems and the reactor core are evaluated at the EPU conditions. Key physics safety parameters for CR-3 are evaluated for the three conceptual fuel cycles at the EPU conditions. The specific values for safety parameters, e.g., shutdown margin, reactivity coefficients, soluble boron worths, control rod worths, critical boron concentrations, shutdown boron concentrations, and kinetic parameters are evaluated and shown to be within cycle-to-cycle variations for current reloads or, for parameters that have significantly changed as a result of the EPU, shown to be acceptable. The CR-3 cycle 16 (pre-EPU) design is used for comparison of safety parameters to evaluate the continued adequacy of margins between typical safety parameter values and the corresponding limits. FSAR accident evaluations based upon key safety parameters generated for the conceptual EPU cycles are evaluated in Section 2.8.5.

Nuclear Design Improved Technical Specification (ITS) changes for the EPU are provided below.

- Shutdown margin is increased from 1.0 %Δk/k to 1.3 %Δk/k for Modes 1 and 2 operation. The increased shutdown margin for Modes 1 and 2 is required for the Main Steam Line Break Accident Evaluation at the EPU conditions (Section 2.8.5). Modes 3, 4, and 5 are unaffected by the EPU and the increase in shutdown margin for the Main Steam Line Break Accident Evaluation is not required for these modes. The shutdown margin limits will be provided in the Core Operating Limits Report. Refer to Attachment 1 for additional discussion.

- An increase in the Borated Water Storage Tank minimum boron concentration from 2270 ppmB to 2600 ppmB is required (ITS 3.5.4). The same increase in the Core Flood Tank minimum boron concentration is required (ITS 3.5.1). These increases in minimum boron concentration are necessary to support borated water volume requirements for cold shutdown at the EPU conditions.

- A change to support credit for inserted control rods in refueling boron concentration calculations (ITS Bases 3.9.1) is proposed. Current refueling boron concentrations are calculated with all control rods removed from the core. Credit for inserted control rods will decrease refueling boron concentrations and prevent exceeding the allowable maximum boron concentrations for the borated water storage tanks.

- A change to update discussion for providing and maintaining shutdown margin requirements for modes 3, 4, and 5 using boric acid (ITS Bases 3.1.1) is proposed. Shutdown margin requirements can be satisfied with increases in boric acid volumes in the Boric Acid Storage Tank(s). Increases in the boric acid volume necessary to meet the shutdown margin requirements are consistent with changes to values used in the ITS Bases 3.1.1 ACTIONS A.1 example.

These changes are reflected in ITS and Bases markups in a separate attachment for the EPU.

Table 2.8.2-2 provides the key safety parameter ranges compared to the current values.

EPU Core Power Distribution Analysis

The loading patterns developed for the three high-capacity two-year fuel cycles based on projected energy requirements for operation at a rated thermal power of 3014 MWt at CR-3 were evaluated with respect to available margins to power peaking limits, the impact on COLR limits related to power distribution, and the ability to suppress axial power oscillations. Conceptual cycles 18 and 20 were evaluated in detail. Cycle 19 was not evaluated because it is nearly identical to cycle 20 with respect to power distributions.

Based on these fuel cycle designs and approved reload analysis methodology (Reference 1), limiting core power distributions and peaking margins were calculated at equilibrium xenon and simulated transient xenon conditions at the limiting times in life in each cycle. Margins to power peaking limits based on centerline fuel melt, transient cladding strain, and steady-state departure from nucleate boiling (DNB) criteria were calculated to evaluate the Reactor Protection System (RPS) axial offset limits that would be input to the determination of the RPS power/imbalance/flow trip function. Margins to power peaking limits based on LOCA and initial-condition DNB criteria were calculated to evaluate the Limiting Conditions of Operation (LCO) axial offset limits that would be input to the determination of the axial power imbalance operating envelope, regulating rod insertion limits, and operational flexibility of the uprated cores. The impact of fuel rod bow is included in the evaluation of the RPS and LCO axial offset limits, and the impact of the limiting overcooling transient as well as dropped/misaligned control rod is included in the evaluation of the LCO axial offset limits.

The core power distributions were analyzed to determine the dependence of the core power distribution on fuel burnup, thermal power level, control rod positions, and xenon distribution. The core power distribution evaluations performed for the uprate to 3014 MWt addressed these parameters at rated thermal power and design overpower conditions. From these margins to peaking limits, a set of RPS axial offset limits, LCO axial offset limits, and peaking-based and shutdown margin-based rod insertion limits for operation at 3014 MWt were determined and compared to corresponding values for Cycle 16 which is licensed for operation at 2609 MWt. See Tables 2.8.2-3 and 2.8.2-4 for comparisons of offset limits, LCO rod insertion limits, and shutdown margin-based rod insertion limits. The final rod insertion limits would be the more restrictive of the shutdown margin-based or the control rod ejection accident-based rod insertion limits. The results of the evaluation indicated that, while the magnitudes of RPS offset limits, LCO offset limits and shutdown margin-based rod insertion limits in some instances are more restrictive than previous CR-3 fuel cycles, these limits are still acceptable for normal operation and anticipated operational transients. The installation of a bypass line at the Makeup System Tank (MUT-1), as described in Section 2.1.11, Chemical and Volume Control System, will improve the response time in core reactivity control and thereby assist in maintaining or restoring API operating limits. The RPS offset limits, LCO offset limits, and rod insertion limits preserve fuel rod power peaking limits and provide adequate operating margins. Therefore, the cycles evaluated are acceptable for operation at the EPU conditions.

The cycle-specific power distribution analysis prepared for each reload core will implicitly include the effects of the EPU on core power distribution and peaking margins. The COLR limits will be adjusted on a cycle-specific basis, as appropriate, to preserve the core peaking limits and provide adequate operating margin for each reload cycle.

The xenon stability evaluation determined the ability of the reactor core to suppress axial power oscillations. The EPU cores have a negative xenon stability index which means axial power oscillations are convergent (i.e., they are naturally dampened without use of the controlling rod bank).

Codes and Methods Applicability for CR-3 EPU

The relevant neutronics codes and methods topical reports are reviewed for applicability to the CR-3 EPU. These NRC approved codes and methods are applied to the current CR-3 operating cycle at 2609 MWt. These codes and methods are also applied to reload licensing analyses supporting a B&W plant operating at 2817 MWt. In addition, these neutronics codes are also applied to reload licensing analyses supporting the Sequoyah Units 1 and 2 plants currently operating at a rated thermal power of 3455 MWt. No limitations or restrictions are placed on these codes or methodologies used to support this higher power level.

The reviewed topical reports include:

- Safety Criteria and Methodology for Acceptable Cycle Reload Analyses (BAW-10179P-A) (Reference 1)
- NEMO-Nodal Expansion Method Optimized (BAW-10180-A) (Reference 4)
- NEMO-K, A Kinetics Solution (BAW-10221P-A) (Reference 6)
- Evaluation of Replacement Rods in BWFC Fuel Assemblies (BAW-2149-A) (Reference 7)
- Core Calculational Techniques and Procedures (BAW-10118-A) (Reference 8)
- Normal Operating Controls (BAW-10122-A) (Reference 9)
- Rod Bowing in B&W Fuel Designs (BAW-10147P-A) (Reference 10)
- Fuel Densification Report (BAW-10054P-A) (Reference 11)
- Comparison of Core Physics Calculations with Measurements (BAW-10120P-A) (Reference 12)

No changes to the Nuclear Design analysis methods are necessary for the EPU with the exception of the control rod ejection accident methodology. Reference 2 provides the EPU control rod ejection accident methodology, which has been approved by the NRC for use at CR-3. No changes to the Nuclear Design codes are necessary for the EPU. In this review, it was determined that the neutronics topical reports listed above remain applicable for the CR-3 EPU.

Evaluation of Control Rod Ejection Accident (Reactivity Insertion Accident)

One of three options can be performed in order to meet any changes in reload design requirements for the control rod ejection accident analysis of record:

1. The current analysis of record can be shown to be applicable to the fuel cycle design.
2. Portions of the analysis can be repeated for the fuel cycle design.
3. A complete reanalysis can be performed.

Based on the methodology in Reference 2, a checklist of reload acceptance criteria provided in Table 2.8.2-5, is to be verified for each new fuel cycle design. This checklist validates the cycle specific verification to the safety analysis in Section 2.8.5. For beginning of cycle and end of cycle, the hot zero power, 20 percent power, and hot full power parameters are verified each cycle as identified in Option 1 above. Options 2 and 3 can be exercised if the fuel cycle designs exceed the requirements of the checklist. These options would require a reanalysis and would follow the methodology as described in Reference 2.

The criteria in Table 2.8.2-5 are calculated for the ejected rod worth conditions and the respective uncertainties applied to those values as defined in Reference 2. For the analysis, the control rod must be ejected from a position equal to the limit in Figure 2.8.2-1 (this figure is contained in both this section and Section 2.8.5) or farther withdrawn. If a farther withdrawn initial position is used, it becomes the limit for the cycle. The plant implemented rod index is error adjusted for power and rod index uncertainties for plant alarms. The number of pins failed uses the methodology as described in Reference 2. These checks were performed for conceptual cycles 18, 19, and 20 and are shown in Tables 2.8.2-6, 2.8.2-7, and 2.8.2-8, respectively. All of the verification criteria are met.

<u>Results</u>

Fuel cycle design characteristics for three conceptual EPU fuel cycles are summarized in Table 2.8.2-1. Key nuclear parameters for all three CR-3 conceptual EPU fuel cycles are summarized in Table 2.8.2-2. Offset limits and shutdown margin-based rod insertion limits are provided in Tables 2.8.2-3 and 2.8.2-4, respectively. Margin to key safety parameter limits is not significantly reduced by the EPU to 3014 MWt.

The change in fuel design characteristics, key safety parameters, and core power distributions resulting from the CR-3 EPU are generally within cycle-to-cycle variations. In some cases (increased boron concentrations, reduced available shutdown margin, more restrictive rod insertion limits, etc.) significant differences are noted. Key safety parameters and core power distribution results that have significantly changed with respect to the currently operating CR-3 fuel cycle and FSAR limits as a result of the uprated power conditions are shown to be acceptable for the EPU. Any additional changes in the Nuclear Design that result from EPU operating conditions will be addressed as part of the future reload safety analysis process.

The discharge burnups and assembly energy requirements have increased, relative to the current design, due to the increase in core power (Table 2.8.2-1). These will vary from cycle-to-cycle based on actual energy requirements. The current methods of feed enrichment variation and insertion of fresh burnable absorbers will be employed to control peaking factors.

Reload cycle verification checks were performed for the EPU conceptual cycles using the EPU control rod ejection accident methodology (Reference 2). Results of these checks for conceptual cycles 18, 19, and 20 are acceptable as shown in Tables 2.8.2-6, 2.8.2-7, 2.8.2-8. All of the verification checklist criteria are met.

The calculated fluence on the vessel was evaluated for the impact of the proposed EPU on the reactor vessel integrity evaluations. The results of this evaluation are provided in Section 2.1.1, "Reactor Vessel Material Surveillance Program."

No changes to the Nuclear Design analysis methods are necessary for the EPU with the exception of utilizing the approved EPU control rod ejection accident methodology (Reference 2). No changes to the Nuclear Design codes are necessary for the EPU.

2.8.2.3 Conclusion

The effects of the proposed CR-3 EPU have been analyzed for the Nuclear Design of the fuel assemblies, control systems, and reactor core. This analysis adequately accounts for the effects of the proposed EPU on the Nuclear Design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB nor impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the Fuel System Design, thermal and hydraulic design, and the transient and accident analyses, CR-3 concludes that the Nuclear Design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of FSAR Sections 1.4.6, 1.4.7, 1.4.8, 1.4.12, 1.4.13, 1.4.14, 1.4.15, 1.4.27, 1.4.28, 1.4.29, 1.4.30, 1.4.31, and 1.4.32. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Nuclear Design.

2.8.2.4 References

1. BAW-10179P-A, Rev. 7 (Proprietary), “Safety Criteria and Methodology for Acceptable Cycle Reload Analysis.”

2. ANP-2788P, “Crystal River 3 Rod Ejection Accident Methodology Report.”

3. CASMO-3 – A Fuel Assembly Burnup Program, STUDSVIK/NFA-89/3, November 1989.

4. BAW-10180-A, Rev. 1, “NEMO – Nodal Expansion Method Optimized.”

5. BAW-10186P-A, Rev. 1 (Proprietary), “Extended Burnup Evaluation.”

6. BAW-10221P-A (Proprietary), “NEMO-K, A Kinetics Solution in NEMO.”

7. BAW-2149-A, “Evaluation of Replacement Rods in BWFC Fuel Assemblies.”

8. BAW-10118-A, “Core Calculational Techniques and Procedures.”

9. BAW-10122-A, Rev. 1, “Normal Operating Controls.”

10. BAW-10147P-A, Rev. 1 (Proprietary), “Fuel Rod Bowing in Babcock and Wilcox Fuel Designs.”

11. BAW-10054P-A, Rev. 2 (Proprietary), “Fuel Densification Report.”

12. BAW-10120P-A, “Comparison of Core Physics Calculations with Measurements.”

Table 2.8.2-1: CR-3 EPU Fuel Cycle Characteristics

Cycle	Feed Assem.	Feed Batch Enrich. (wt% U235)	HFP FΔh	BOC HFP EQ XE Critical Boron (ppmB)	Coolant System Press. (psia)	Core Avg. Coolant Temp. HFP (°F)	Reactor Core Power (MWt)	End of Licensed Length (EFPD)
16	73	4.30	1.485	1492	2200	580	2609	667
18[(a)]	88	4.95	1.441	1682	2200	583	3014	715
19[(a)]	88	4.95	1.439	1658	2200	583	3014	715
20[(a)]	88	4.95	1.436	1655	2200	583	3014	715

Notes:

(a) Values provided are for conceptual EPU cycles. The actual cycle-specific core design is evaluated each cycle to confirm the design limits are maintained.

Table 2.8.2-2: Range of Key Safety Parameters

Safety Parameter	Current Design Values (CR-3 Cycle 16)	EPU Analysis Values[(a)] (EPU Conceptual Cycles)
Most Positive MTC (pcm/°F)	≤ + 3.3 (HZP) ≤ -1.9 (Power ≥ 80%)	≤ + 3.7 (HZP) ≤ -1.6 (Power ≥ 80%)
Most Negative MTC (pcm/°F)	-33.15	-35.75 to -35.71
EOC HFP COLR MTC (pcm/°F)	-35.8	-37.5
Doppler Temperature Coefficient (pcm/°F)	-1.66 to -1.77	-1.53 to -1.73
Total Power Deficit, Most Positive, HZP to HFP (%Δk/k)	-0.42	-0.43
Total Power Deficit, Most Negative, HZP to HFP (%Δk/k)	-2.80	-3.14
Beta-Effective	0.0064 to 0.0053	0.0066 to 0.0053
Minimum Shutdown Margin (%Δk/k)	2.54	1.57 to 1.67
BOC HZP SDM Total Rod Worth (%Δk/k)	7.37	6.73 to 6.91
HZP Maximum Stuck Rod Worth (%Δk/k)	1.20	1.22 to 1.25
Maximum Dropped Rod Worth (%Δk/k)	0.12	0.10 to 0.11
HFP Inverse Boron Worth (ppmB/%Δk/k)	163	182 to 183
BOC HFP ARO Critical Boron Concentration (ppmB)	1904	2117 to 2145
BOC Cold Shutdown Boron Concentration (ppmB)	1713	2013 to 2019
EOLL Cold Shutdown Boron Concentration (ppmB)	513	657 to 665

Notes:

(a) Values provided are for the conceptual EPU cycles 18-20. These values may change for actual EPU cycles in the future.

Table 2.8.2-3: EPU versus Cycle 16 Offset and LCO Rod Insertion Limits

Limit	Cycle 16 (2609 MWt)	Cycle 18[a] (3014 MWt)	Cycle 20[a] (3014 MWt)
4-RCP RPS Offset (%)	-46.6 / +34.6	-35.9 / +33.5	-37.2 / +34.9
3-RCP RPS Offset (%)	-46.6 / +37.6	-35.9 / +33.8	-37.2 / +35.2
LCO Offset (%)	-25.7[b] / +18.6[b]	-14.3 / +20.0	-15.6 / +20.0
LCO Rod Insertion (%WD)	264	264	264

Notes:

(a) Values provided are for the conceptual EPU cycles 18 and 20. These values may change for actual EPU cycles in the future.
(b) Cycle 16 COLR imbalance limits were set at -17.0% and +14.0%.

Table 2.8.2-4: End of Cycle SDM-Based Rod Insertion Limits (RIL)

Limit	Cycle 16 (2609 MWt)	Cycle 18[a] (3014 MWt)	Cycle 20[a] (3014 MWt)
SDM RIL (%WD)	195	261	270

Notes:

(a) Values provided are for the conceptual EPU cycles 18 and 20. These values may change for actual EPU cycles in the future.

Table 2.8.2-5: Control Rod Ejection Accident Cycle Verification Checklist

Parameter	Acceptable Values	Cycle Specific Criteria					
		BOC			EOC		
		HZP	20%	HFP	HZP	20%	HFP
Maximum Ejected Rod Worth, pcm	≤	715	556	60	741	535	73
β_{eff}	≥	0.0058	0.0058	0.0058	0.0048	0.0048	0.0048
MTC, pcm/°F	≤	2.5	0.0	-2.0	-14.5	-25.0	-26.0
DTC, pcm/°F	≤	-1.30	-1.24	-1.00	-1.40	-1.36	-1.20
Initial F_Q	≤	NA [(a)]	3.48	2.53	NA [(a)]	5.37	2.25
Static F_Q After Ejection	≤	14.84	8.88	3.07	27.23	12.70	3.73
Initial $F_{\Delta H}$	≤	NA[(a)]	2.27	1.71	NA[(a)]	2.27	1.71
Static $F_{\Delta H}$ After Ejection	≤	8.15	5.51	2.20	7.59	4.85	2.31
Equivalent Nominal Rods Failed, %	≤	0	14.6[(b)]	14.6[(b)]	0	0	14.6[(b)]

Notes:

(a) Not applicable for HZP conditions since initial stored energy above the coolant temperature is zero.

(b) The release fractions for failures are based on Regulatory Guide 1.183, Appendix H values.

Table 2.8.2-6: Control Rod Ejection Accident Conceptual Cycle 18 Verification Checklist

Parameter	Acceptable	Cycle Specific Values					
		BOC			EOC		
		HZP	20%	HFP	HZP	20%	HFP
Maximum Ejected Rod Worth, pcm	$\leq$ Limit - yes	465	332	56	377	331	69
β_{eff}	$\geq$ Limit - yes	0.0063	0.0063	0.0063	0.0051	0.0051	0.0051
MTC, pcm/°F	$\leq$ Limit - yes	-1.45	-3.35	-5.54	-18.85	-31.78	-32.50
DTC, pcm/°F	$\leq$ Limit - yes	-1.40	-1.35	-1.13	-1.56	-1.50	-1.31
Initial F_Q	$\leq$ Limit - yes	NA [(a)]	2.92	2.20	NA [(a)]	4.44	1.67
Static F_Q After Ejection	$\leq$ Limit - yes	12.90	7.05	2.84	17.23	10.49	3.04
Initial $F_{\Delta H}$	$\leq$ Limit - yes	NA [(a)]	1.85	1.62	NA [(a)]	1.73	1.47
Static $F_{\Delta H}$ After Ejection	$\leq$ Limit - yes	6.65	4.25	1.99	5.22	3.86	1.86
Equivalent Nominal Rods Failed, %	$\leq$ Limit - yes	* [(b)]	0.0	0.1	* [(b)]	* [(b)]	0.3

Notes:

(a) Not applicable for HZP conditions since initial stored energy above the coolant temperature is zero.

(b) "*" means no specific check on failures is needed because no failures were observed for the peak pin and if the peaking criteria are met, then no failures occurred.

Table 2.8.2-7: Control Rod Ejection Accident Conceptual Cycle 19 Verification Checklist

Parameter	Acceptable	Cycle 19 Specific Values					
		BOC			EOC		
		HZP	20%	HFP	HZP	20%	HFP
Maximum Ejected Rod Worth, pcm	$\leq$ Limit - yes	512	346	56	366	334	68
β_{eff}	$\geq$ Limit - yes	0.0063	0.0063	0.0063	0.0051	0.0051	0.0051
MTC, pcm/°F	$\leq$ Limit - yes	-1.97	-4.09	-6.19	-18.82	-31.88	-32.54
DTC, pcm/°F	$\leq$ Limit - yes	-1.40	-1.35	-1.13	-1.56	-1.50	-1.31
Initial F_Q	$\leq$ Limit - yes	NA (a)	2.96	2.20	NA (a)	4.43	1.66
Static F_Q After Ejection	$\leq$ Limit - yes	13.57	7.21	2.83	16.76	10.37	3.04
Initial $F_{\Delta H}$	$\leq$ Limit - yes	NA (a)	1.86	1.62	NA (a)	1.72	1.47
Static $F_{\Delta H}$ After Ejection	$\leq$ Limit - yes	7.08	4.32	1.98	5.10	3.82	1.85
Equivalent Nominal Rods Failed, %	$\leq$ Limit - yes	* (b)	0.0	0.1	* (b)	* (b)	0.2

Notes:

(a) Not applicable for HZP conditions since initial stored energy above the coolant temperature is zero.

(b) "*" means no specific check on failures is needed because no failures were observed for the peak pin and if the peaking criteria are met, then no failures occurred.

Table 2.8.2-8: Control Rod Ejection Accident Conceptual Cycle 20 Verification Checklist

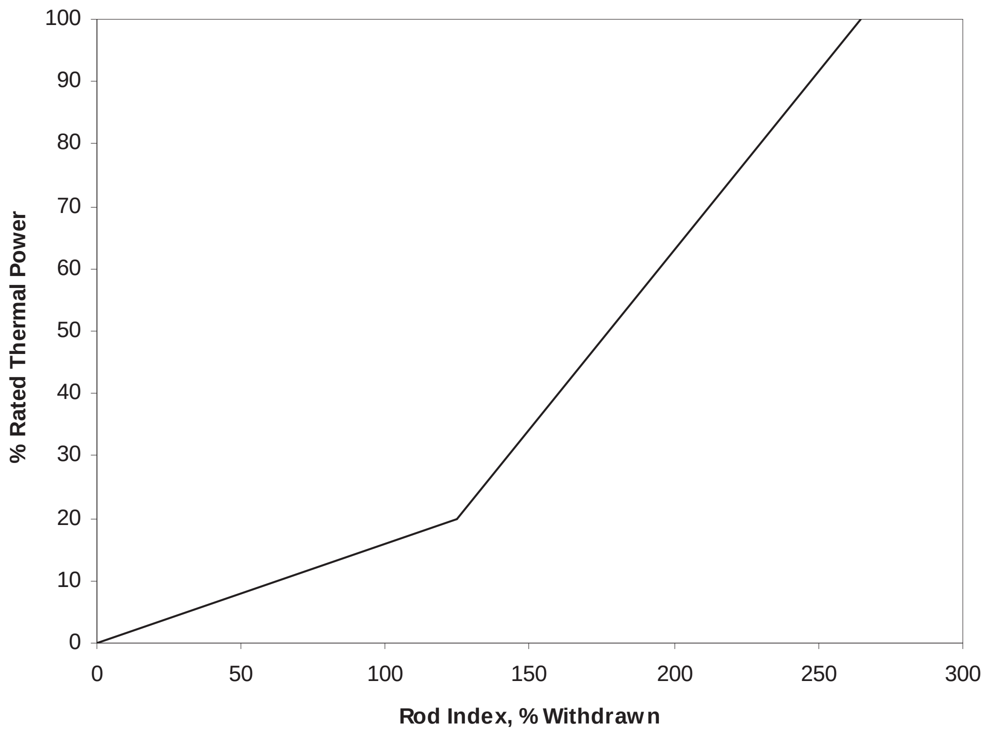
Parameter	Acceptable	Cycle 20 Specific Values					
		BOC			EOC		
		HZP	20%	HFP	HZP	20%	HFP
Maximum Ejected Rod Worth, pcm	$\leq$ Limit - yes	498	339	59	362	330	69
β_{eff}	$\geq$ Limit - yes	0.0063	0.0063	0.0063	0.0051	0.0051	0.0051
MTC, pcm/°F	$\leq$ Limit - yes	-2.02	-4.11	-6.20	-18.83	-31.89	-32.53
DTC, pcm/°F	$\leq$ Limit - yes	-1.40	-1.35	-1.13	-1.55	-1.50	-1.31
Initial F_Q	$\leq$ Limit - yes	NA [(a)]	2.97	2.18	NA [(a)]	4.42	1.66
Static F_Q After Ejection	$\leq$ Limit - yes	13.61	7.17	2.82	16.66	10.30	3.02
Initial $F_{\Delta H}$	$\leq$ Limit - yes	NA [(a)]	1.86	1.62	NA [(a)]	1.72	1.47
Static $F_{\Delta H}$ After Ejection	$\leq$ Limit - yes	6.96	4.27	1.98	5.08	3.80	1.84
Equivalent Nominal Rods Failed, %	$\leq$ Limit - yes	* [(b)]	0.0	0.1	* [(b)]	* [(b)]	0.2

Notes:

(a) Not applicable for HZP conditions since initial stored energy above the coolant temperature is zero.

(b) "*" means no specific check on failures is needed because no failures were observed for the peak pin and if the peaking criteria are met, then no failures occurred.

Figure 2.8.2-1: Control Rod Position Limits for Rod Ejection Accident Analysis



2.8.3 Thermal and Hydraulic Design

2.8.3.1 Regulatory Evaluation

CR-3 reviewed the Thermal and Hydraulic Design of the core and the Reactor Coolant System (RCS) to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and design basis accident (DBA) conditions and core thermal-hydraulic stability under normal operation and anticipated transients without scram (ATWS) events.

The NRC's acceptance criteria for the Thermal and Hydraulic Design are based on:

- GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during and condition of normal operation, including the effects of anticipated operational occurrences (AOOs); and
- GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can reliably and readily be detected and suppressed.

<u>CR-3 Current Licensing Basis</u>

As noted in the FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided today in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during and condition of normal operation, including the effects of anticipated operational occurrences (AOOs) [GDC-10]; and
- FSAR Section 1.4.7, Suppression of Power Oscillations, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding specified acceptable fuel design limits, are not possible or can reliably and readily be detected and suppressed [GDC-12].

Continued applicability of the EPU safety analysis for the CR-3 Thermal-Hydraulic Design will be evaluated during the cycle specific reload analysis licensing process for the EPU reload cycles. The reload analysis licensing methodology includes the evaluation of the thermal-hydraulic parameters, limits, and analyses which comprise the thermal-hydraulic input to the FSAR safety analysis and Technical Specification/Core Operating Limits Report (COLR) requirements for each reload cycle.

2.8.3.2 Technical Evaluation

Introduction

This section describes the thermal-hydraulic analysis supporting the CR-3 EPU, which increases the reactor thermal power to 3014 MWt. The current Thermal-Hydraulic Design basis for CR-3 includes the prevention of departure from nucleate boiling (DNB) on the limiting fuel rod with a 95% probability at a 95% confidence level (99.9% probability at a 95% confidence level for the remainder of the core) and criteria to ensure fuel cladding integrity, and is documented in the CR-3 FSAR Section 3.1.2.3. The EPU analysis is based on this licensing basis analysis incorporating the increased core power.

Input Parameters, Assumptions, and Acceptance Criteria

Thermal-hydraulic characteristics are evaluated for three conceptual fuel cycles at the EPU conditions using the Mark-B-HTP fuel assembly and NRC approved BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," consistent with the current licensing basis. The actual cycle-specific core design is evaluated each cycle utilizing this methodology to confirm the design limits are maintained. This analysis forms the basis for the evaluation of thermal-hydraulic parameters, limits, and Technical Specifications/COLR impact. The Fuel System Design for the CR-3 EPU core conditions is provided in Section 2.8.1.

Standard Thermal-Hydraulic Design analytical models and methods (References 1, 2, and 5) accurately describe the thermal-hydraulic behavior of the CR-3 EPU Thermal-Hydraulic Design and the Mark-B-HTP fuel design. The analytical models and methods and the Mark-B-HTP fuel assembly design are used in the current operating CR-3 cycle.

The basic parameters that are used in the Thermal-Hydraulic Design analysis are listed in Table 2.8.3-1. Values used in the EPU analysis are provided along with current (Cycle 16) values for comparison.

The EPU Thermal-Hydraulic Design analyses require changes to Improved Technical Specifications (ITS) for the uprated power conditions. Specifically, the Reactor Coolant System DNB Safety Limits that describe the minimum core outlet pressure for a given reactor outlet temperature are changed. Additionally, the changes to the Technical Specifications as a result of the T-H analyses are:

- ITS Figure 2.1.1-1, Reactor Coolant System DNB Safety Limits – this figure is updated to match the pressure-temperature limits calculated by the DNB analyses.
- ITS Surveillance Requirement 3.4.1.2 – The RCS hot leg temperature limit is changed from ≤ 605.8°F to ≤ 611.2°F. The EPU DNB analysis determined 611.2°F to be the maximum vessel outlet temperature for a 582°F core average temperature, accounting for uncertainties.
- ITS Surveillance Requirement 3.4.1.3 – The RCS total flow rate is changed from ≥ 133.5 E6 lb/hr with four RCPs operating and ≥ 99.7 E6 lb/hr with three RCPs operating to

$\geq$ 139.4 E6 lb/hr with four RCPs operating and $\geq$ 104.2 E6 lb/hr with three RCPs operating. The minimum flow rate was calculated in the EPU DNB analysis using the minimum flow, maximum RCS average temperature, minimum nominal core exit pressure, and minimum nominal power level, based on their uncertainties.

Description of Analyses and Evaluations

Core Thermal-Hydraulic Margin Assessment

The codes and methodologies relevant to the core thermal-hydraulic margin assessment were reviewed and found to be applicable to CR-3 at the core conditions that will follow the EPU. The applicable codes and methodologies are described below.

LYNXT

The LYNXT computer program performs thermal-hydraulic calculations as described in BAW-10156-A (Reference 2). This code calculates coolant density, mass velocity, enthalpy, void fractions, quality, static pressure, steady state and transient Departure from Nucleate Boiling Ratio (DNBR) distributions, and cross-flow along flow channels within a reactor core. LYNXT is used in DNB calculations for both steady state and transient conditions, and to provide pressure drop inputs to statistical holddown calculations.

Safety Criteria and Methodology for Acceptable Cycle Reload Analyses

The Safety Criteria and Methodology for Acceptable Cycle Reload Analyses topical report BAW-10179P-A (Reference 5) describes the entire spectrum of methodologies that are acceptable to the reload fuel supplied by AREVA for B&W 177-FA plants.

Statistical Core Design

Statistical Core Design (SCD), as described in BAW-10187P-A (Reference 1), is a thermal-hydraulic analysis technique that provides an increase in core thermal (DNB) margin by treating core state and bundle uncertainties statistically. The uncertainty distribution for each of the applicable variables is subjected to a Monte Carlo propagation analysis to determine an overall statistical DNBR penalty which is used to establish a statistical design limit (SDL). The SDL developed provides 95 percent protection at a 95 percent confidence level against hot pin DNB. The corresponding corewide protection on a pin-by-pin basis using real peaking distributions is greater than 99.9 percent. With this method, the nominal values for core and bundle parameters is used in the analyses, as opposed to using the most conservative value deterministically based on the nominal value and the uncertainty.

The ability to operate the core safely at the uprated power level is determined through steady-state and transient analyses. Minimum departure from nucleate boiling ratios (MDNBRs) are computed and compared to the Statistical Core Design (SCD) thermal design limit (TDL) to assure that there is adequate thermal-hydraulic margin. The details are provided below.

The Thermal-Hydraulic Design is performed using approved SCD techniques. The SCD methodology for B&W designed, 177 fuel assembly plants is described in detail in (Reference 1). Basically, the SCD approach treats core state and rod bundle uncertainties statistically. The traditional method is to assume that the worst level of each uncertainty occurs simultaneously, which is highly unlikely. Applying statistical techniques allows for a realistic assessment of core DNB protection.

The uncertainty distribution for each of the applicable variables is used in a Monte Carlo propagation analysis to determine an overall statistical DNBR penalty that is used to establish a Statistical Design Limit (SDL). The variables treated in this way are then input to the approved thermal-hydraulic analysis computer code LYNXT (Reference 2) at their nominal values. The nominal values for key parameters whose uncertainties are treated statistically are shown in Table 2.8.3-1 and their uncertainties are given in Table 2.8.3-2. Variables not used in deriving the SDL continue to be input at their most conservative value. The input variables that are treated statistically, and the number of uncertainties associated with each, are listed in Table 2.8.3-2. The limiting parameter direction for each variable is shown in Table 2.8.3-4. Uncertainty values are given in Table 2.8.3-2.

The SDL developed for the CR-3 EPU, using SCD techniques, is { } for the BWC Critical Heat Flux (CHF) region (below the first intermediate spacer grid) and { } in the BHTP region (the remainder of the fuel height). This DNBR provides 95% probability at a 95% confidence level that the hot fuel pin in the core will not experience DNB during normal operation or during moderate frequency events (Condition I and II events defined in ANSI/ANS-57.5-1981 and outlined in BAW-10179P-A). The corresponding core wide protection on a pin-by-pin basis using real peaking distributions is greater than 99.9%. In addition to the conservatism inherent in the SDL, additional DNBR margin is introduced by using a Thermal Design Limit (TDL) of 1.477 for the BWC CHF region and 1.45 for the BHTP region. The margin between the TDL and the SDL is available to offset cycle to cycle abnormalities (such as transition core effects or deviations in uncertainty values from those incorporated in the SDL) or to provide flexibility in the fuel cycle design. As long as the calculated minimum DNBR for events of interest is above the TDL, the fuel is more than adequately protected against damage due to DNB.

Steady State Analysis

A collection of analyses was performed to assess the steady-state performance of the core at the EPU power level of 3014 MWt. The cases covered the expected nominal and overpower conditions for a varying number of reactor coolant pumps in operation. The minimum DNBR for each required by BAW-10179P-A (Reference 5) was above the TDL as shown in Table 2.8.3-3.

DNBR steady-state analyses were also performed to establish the Pressure-Temperature (PT) safety limits. These are sets of pressure/temperature combinations that result in a DNBR equal to the TDL and are used to define the variable low pressure trip function. The information was generated for both 4-pump and 3-pump operation as CR-3 does not allow 2 pump operation.

Reactor Protection System (RPS) Maximum Allowable Radial Peaking (MARP) limits were generated for combinations of axial peaking values and axial peak locations. This task was accomplished by selecting an axial peak and location, and then varying the radial peak until the minimum DNBR was equal to the TDL. The MARP limits are established at the limiting pressure/temperature setpoints as described above. These limits are used as input to the core power distribution peaking margin analysis.

A comprehensive discussion of MARP limits can be found in BAW-10179P-A (Reference 5). Reference 5 discusses the safety criteria and methodology used in performing cycle reload analyses. As such, it is also the basis for the analysis described below.

The CR-3 EPU Thermal-Hydraulic Design analyses require a change to Technical Specifications for the uprated power conditions. Specifically, the Reactor Coolant System DNB Safety Limits that describe the minimum core outlet pressure for a given reactor outlet temperature are changed.

Transient Analysis

A number of loss-of-coolant flow (LOCF) events are assessed, all reflecting the EPU nominal power level of 3014 MWt. The transients evaluated are:

- one pump coastdown from four pump operation
- four pump coastdown
- locked rotor from four pump operation

All of the events are initiated at 100% of nominal power. The power and flow coastdown inputs used in these analyses are taken from the transient analyses described in Section 2.8.5.3, Decrease in Reactor Coolant System Flow. Section 2.8.5.3.1, Loss of Forced Reactor Coolant Flow, describes the analyses for the one pump coastdown and the four pump coastdown transients, and Section 2.8.5.3.2, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break, describes the analysis for the locked rotor transient. The DNB portion of each transient analysis is described here, as it is a thermal-hydraulic analysis.

The analysis results in Table 2.8.3-3 show that the minimum DNBR for the two coastdown events are above the target DNBR (TDL), thus indicating that no DNB event is expected during either of these transients and no fuel failure is predicted. The analysis results in Table 2.8.3-3 show that the DNBR in the locked rotor event is below the TDL. Because the minimum DNBR for the locked rotor event is below the TDL, a set of locked rotor MARP limits were developed. This is discussed in more detail below.

Operating Limit (OL) MARP limits for the CR-3 EPU are generated for the most limiting Condition II event (as defined in Section 2.8.5.0, Non-LOCA Analyses Introduction) (Reference 5), which is the four pump coastdown. The minimum DNBR during this event is 1.54 (Table 2.8.3-3) using design peaking values (Table 2.8.3-1). The OL MARP limits are obtained for various combinations of axial peaking values and axial peak locations. For a selected axial peak and location, the radial peaking is varied until the minimum DNBR during the transient is equal to a given minimum DNBR (the limiting transient minimum DNBR of 1.54). The process is then repeated for other axial peaks and locations. The OL MARP limits are used as input to the core power distribution peaking margin analysis.

Locked rotor MARP limits are generated for the locked rotor event because the MDNBR for that event – 1.41 (Table 2.8.3-3) – is below the TDL of 1.45. The locked rotor MARP limits describe the power peaking that will cause fuel failure during a locked rotor event. They are generated in the same manner as the OL MARP limits, except that the peaking was varied in each case until the minimum DNBR during the transient is equal to the TDL of 1.45. In the core power distribution peaking margin analysis (Section 2.8.2, Nuclear Design), it is determined that other analyses generate operating limits that bound the locked rotor MARP limits. This indicates that the operating limits protect the core from suffering failed fuel during a locked rotor event for the conceptual core designs used in the EPU analyses. If the operating limits do not protect the core from suffering failed fuel during a locked rotor event in future cycles, a core pin census can be performed and checked against the dose limit for this event.

Guide Tube Cooling

Guide tube cooling at the EPU conditions is assessed. The general practice is that long term bulk boiling within the guide tubes shall be precluded. Prolonged guide tube boiling increases the possibility of corrosion on the guide tube and control elements, and may interfere with the function of the control components. An evaluation is performed for the control rod assembly (CRA) with the maximum control rod insertion permitted in the COLR.

Guide tube cooling is evaluated using the approved LYNXT code (Reference 2). LYNXT is used to calculate the coolant quality at the top of the guide tube flow channel; if the quality at the top of the guide tube is positive, then it is determined that bulk boiling will occur. Fuel rod powers are the same as those used in the steady state analysis, and the power generated in the guide tube is calculated based on the geometries and heating rates of the various components inside the guide tube.

The guide tube cooling analysis calculates a quality of { } at the top of the guide tube flow channel, and therefore concludes that no long term bulk boiling is expected in the CR-3 guide tubes following the uprate.

Main Steam Line Break

A DNB analysis is performed using LYNXT (Reference 2) which models the core during a main steam line break accident. A main steam line break causes an overcooling of the core which in turn causes an increase in power due to the negative moderator temperature coefficient (MTC); this could cause the fuel to go through DNB. This analysis uses radial power distributions and axial power shapes which model the core conditions during such an accident. If the minimum DNBR is sufficiently low, then DNB is possible during a main steam line break accident and failed fuel is predicted. Since the power levels analyzed for a steam-line break are outside the valid range of the BHTP correlation, the W-3 correlation is used in LYNXT for this analysis.

For the EPU analysis, three conceptual cycles were analyzed using a full-core Mark-B-HTP core design and assumed core conditions. The actual cycle-specific core design is evaluated each cycle by utilizing this methodology to confirm the design limits are maintained with corresponding cycle-specific operating limits reported in the COLR, by validating an existing analysis' applicability to the designed cycle, or by justifying that no thermal-hydraulic analysis is necessary to satisfy the requirements of topical report BAW-10179P-A (Reference 5).

The analysis uses multiple core power distributions for each conceptual cycle to find the bounding power distribution and ensure that DNB is not expected in the event of a main steam line break accident. For the three conceptual cycles, the most limiting MDNBR is 1.87, which is above the W-3 SDL of { }. Therefore, no DNB is expected while CR-3 is operating at the EPU power level.

Core Hydraulics

The codes and methodologies relevant to the core hydraulics assessment were reviewed and found to be applicable to CR-3 at the core conditions that will follow the EPU. The applicable codes and methodologies are described below.

LYNXT

The LYNXT computer program performs thermal-hydraulic calculations as described in BAW-10156-A (Reference 2). This code calculates coolant density, mass velocity, enthalpy, void fractions, quality, static pressure, steady state and transient DNBR distributions, and cross-flow along flow channels within a reactor core. LYNXT is used in DNB calculations for both steady state and transient conditions, and to provide pressure drop inputs to statistical holddown calculations.

Safety Criteria and Methodology for Acceptable Cycle Reload Analyses

The Safety Criteria and Methodology for Acceptable Cycle Reload Analyses topical report BAW-10179P-A (Reference 5) describes the entire spectrum of methodologies that are acceptable to the reload fuel supplied by AREVA for B&W 177-FA plants.

Statistical Fuel Assembly Hold Down Methodology

The Statistical Fuel Assembly Hold Down Methodology topical report BAW-10243P-A (Reference 6) describes the statistical methodology that is used to demonstrate that the fuel assembly design provides sufficient downward force to counteract the vertical hydraulic lift force created by the core flow so that the fuel assembly remains in a seated position during normal operation and anticipated transients.

One of the mechanical constraints on operating the core is that the fuel assemblies must remain seated on the lower grid plate of the reactor vessel internals. Coolant flow through an assembly generates a lift force that is counteracted by the weight of the assembly and an additional force supplied by the holddown springs.

Fuel assembly pressure losses are evaluated using the approved LYNXT code (Reference 2). Each fuel assembly is represented by a single channel in the LYNXT model. Lift forces are obtained from the fuel assembly pressure losses calculated by LYNXT. The total lift forces are calculated for four conditions: hot full power, hot overpower, 400°F isothermal, and 500°F isothermal.

Hydraulic Lift

Hydraulic lift calculations are performed for the CR-3 EPU power level of 3014 MWt. For a given flow rate, the fuel assembly lift is calculated as a function of coolant temperature (lift decreases with an increase in coolant temperature due to the density effect). With this relationship and knowing the holddown force, the statistical hold down methodology is used to determine the temperature at which liftoff will occur. As long as the coolant temperature is higher than this value, the fuel assembly will remain seated. The results of this analysis are used to determine the coolant temperature at which the fourth reactor coolant pump can be started. The current analysis of the EPU core configuration indicates that sufficient holddown force exists to prevent liftoff at isothermal conditions or at power.

For the EPU analysis, a conceptual cycle was analyzed using a full-core Mark-B-HTP core design and assumed core conditions. The actual cycle-specific core design is evaluated each cycle utilizing this methodology to confirm the design limits are maintained, with corresponding cycle-specific operating limits reported in the COLR.

For the conceptual EPU analysis, lift forces were calculated in four cases – hot full power with 111% nominal RCS flow, an isothermal case at 400°F with 120% nominal RCS flow, an isothermal case at 500°F with 120% nominal RCS flow, and a hot overpower case at 112% rated thermal power and 120% nominal RCS flow. The resulting lift forces indicate that the assemblies will remain seated during operation provided that the coolant flow remains below a value which is a function of coolant temperature. The relationship between this flow limit and coolant temperature is

{ }

where Q_{RCS} is the maximum protected total RCS flow rate in percent and T is the coolant temperature in °F.

Component Pressure Losses

The basic LYNXT model used for the lift analysis is also used to calculate fuel assembly component (e.g., spacer grid) pressure losses, using 100% power conditions and maximum reactor coolant system flow. These pressure losses are used to obtain the forces acting on the individual components. The forces are used in a mechanical analysis for seismic and loss of coolant accident (LOCA) loading of the assembly.

Cross-Flow Velocities

The lift and pressure loss analysis using LYNXT also provides fuel assembly to fuel assembly cross-flow information. The regions of interest are the fuel rod spans between spacer grids. The mechanical requirement is that the span average cross-flow velocity remains below the mechanical limit for the fuel assembly.

For the EPU analysis, a conceptual cycle was analyzed using a full-core Mark-B-HTP core design and assumed core conditions. The actual cycle-specific core design is evaluated each cycle utilizing this methodology to confirm the design limits are maintained, with corresponding cycle-specific operating limits reported in the COLR.

For the conceptual EPU analysis, crossflow velocities were calculated in four cases – hot full power with 111% nominal RCS flow, an isothermal case at 400°F with 120% nominal RCS flow, an isothermal case at 500°F with 120% nominal RCS flow, and a hot overpower case at 112% rated thermal power and 120% nominal RCS flow. The largest crossflow velocity calculated in any of these LYNXT cases is { } ft/s, which is lower than the 2.0 ft/s limit imposed by the mechanical analysis for the conceptual EPU cycles.

<u>Results</u>

Thermal-hydraulic characteristics for the CR-3 EPU are summarized in Table 2.8.3-1.

The minimum DNBR for each steady-state condition and transient event required by BAW-10179P-A (Reference 5) is calculated (Table 2.8.3-3). With the exception of the locked rotor transient, this analysis indicates that DNB is not expected because the minimum DNBR is sufficiently high; for the locked rotor event, a set of maximum allowable peaking limits are developed that protect the fuel in the core from experiencing DNB.

Guide tube cooling is analyzed for the EPU. It is determined that no long term bulk boiling in the guide tubes is expected during operation at the EPU power level.

The main steam line break analysis shows that the fuel is not expected to experience DNB during a main steam line break event while the plant is operating at the EPU power level. This analysis will have to be repeated or validated on a cycle to cycle basis following the uprate to the EPU power level.

The core hydraulics analysis shows that sufficient holddown forces exist to prevent assembly liftoff during isothermal conditions and at power and that span-average cross-flow velocities remain sufficiently low. Component pressure losses are also calculated for use in mechanical analyses. These analyses will have to be repeated or validated on a cycle to cycle basis following the uprate to the EPU power level.

The CR-3 EPU Thermal-Hydraulic Design is implemented with a change to Technical Specifications for the uprated power conditions. Specifically, the Reactor Coolant System DNB Safety Limits that describe the minimum core outlet pressure for a given reactor outlet temperature are changed.

CR-3 acceptance criteria will be maintained during the cycle design process after the power has been increased to the EPU power level.

2.8.3.3 Conclusion

CR-3 analyses related to the effects of the proposed EPU on the Thermal and Hydraulic Design of the core and the RCS conclude that acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs are maintained. Based on the above, the steady state, transient, guide tube boiling, steam line break, hydraulic lift, component pressure losses, and cross-flow velocities analyses will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of CR-3 FSAR design criteria 1.4.6 and 1.4.7. Therefore, CR-3 finds the proposed EPU acceptable with respect to Thermal and Hydraulic Design.

2.8.3.4 References

1. Statistical Core Design for B&W-Designed 177-FA Plants, BAW-10187P-A, March 1994.
2. LYNXT – Core Transient Thermal-Hydraulic Program, BAW-10156-A, Rev.1, August 1993.
3. BWC Correlation of Critical Heat Flux, BAW-10143P-A, April 1985.
4. BHTP DNB Correlation Applied with LYNXT, BAW-10241P-A, Rev.1, July 2005.
5. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, BAW-10179P-A, Rev. 7, January 2008.
6. Statistical Fuel Assembly Hold Down Methodology, BAW-10243P-A, Rev. 0, September 2005.

Table 2.8.3-1: Thermal-Hydraulic Design Parameters for CR-3

Thermal and Hydraulic Design Parameters	Current Values (Cycle 16)	EPU Values
Design core heat output, MWt	2609	3014
Design core heat output, 10^6 Btu/hr	8908	10,291
Nominal system pressure (core exit), psia	2200	2200
Minimum vessel coolant flow (used for T-H analyses), gpm	367,840	383,680
Core Average Temperature, °F	579	582
Vessel coolant inlet temp, °F	554.77	555.41
Vessel coolant outlet temp, °F	603.23	608.59
Core coolant outlet temp, °F	606.43	612.04
Total core heat transfer surface area, ft^2	49,505	49,505
Core flow area effective for heat transfer, ft^2	49.66	49.66
Average thermal output, kW/ft	5.786	6.684
Maximum thermal output, kW/ft	17.185	19.456
Average core heat flux, Btu/hr-ft^2	175,000	202,265
Maximum core heat flux, Btu/hr-ft^2	521,000	588,712
Max/avg power ratio (radial peak)	1.8	1.764
Max/avg power ratio (axial)	1.65	1.65
Location of axial peak, fraction of active length (x/L)	0.5	0.5
Power generated in fuel and cladding, %	97.3	97.3
Overall power ratio	2.97	2.91
Hot channel factor, pin power	1.011	1.011
Hot channel factor, local heat flux	1.014	1.014
Flow area reduction factor, interior bundle	0.98	0.98
Flow area reduction factor, peripheral bundle	0.97	0.97

Table 2.8.3-2: CR-3 SCD Input Variables

State Variable	Uncertainties
Core Power	{ }
Reactor Coolant System (RCS) Pressure	{ }
Core Flow	{ }
Inlet Core Subcooled Temperature	{ }
Hot Pin Radial Peaking Factor	{ }
DNBR	{ }
Hot Pin Axial Peaking Factor	{ }
Hot Pin Axial Peak Location	{ }

Table 2.8.3-3: Summary of DNBR Evaluations for CR-3 EPU Conditions

Criterion or Operating Condition	Cycle 16 DNB Ratio	EPU Analyses DNB Ratio
Thermal Design Limit (TDL)	1.50	1.45
Nominal conditions (100% power), 4 pump operation	2.22	1.93
Maximum overpower, 4 pump operation	1.99	1.68
Nominal conditions (75% power), 3 pump operation	2.61	2.26
Maximum overpower, 3 pump operation	2.13	1.87
1 pump coastdown	1.94	1.62
4 pump coastdown	1.73	1.54
Locked rotor from 4 pump operation	1.78	1.41

Table 2.8.3-4: Limiting Parameter Directions

Parameter	Limiting Direction for DNB
$F^N_{\Delta H}$, nuclear enthalpy rise hot-channel factor	maximum
Heat generated in fuel, %	maximum
Reactor core heat output, MWt	maximum
Average heat flux, BTU/hr-ft	maximum
Nominal vessel/core inlet temperature, °F	maximum
Core pressure, psia	minimum
Pressurizer pressure, psia	minimum
Core Flow, gpm	minimum

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

2.8.4.1.1 Regulatory Evaluation

The CR-3 review covered the Functional Design of the Control Rod Drive System (CRDS) to confirm that the system can effect a safe shutdown, respond within acceptable limits during anticipated operational occurrences (AOOs), and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements.

The NRC's acceptance criteria for the Functional Design of the Control Rod Drive System are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents;
- GDC-23, insofar as it requires that the protection system be designed to fail into a safe state;
- GDC-25, insofar as it requires that the protection system be designed to assure that specific allowable fuel design limits (SAFDLs) are not exceeded for any single malfunction of the reactivity control systems;
- GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes;
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;
- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and
- GDC-29, insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found to be acceptable by the NRC for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.26, Protection Systems Fail-Safe Design, insofar as it requires that the protection system be designed to fail into a safe state [GDC-23];
- FSAR Section 1.4.31, Reactivity Control Systems Malfunction, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems [GDC-25];
- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes [GDC-26];
- FSAR Section 1.4.28, Reactivity Hot Shutdown Capability; FSAR Section 1.4.29, Reactivity Shutdown Capability; and FSAR Section 1.4.30, Reactivity Holddown Capability, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained [GDC-27]; and
- FSAR Section 1.4.32, Maximum Reactivity Worth of Control Rods, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core [GDC-28].

Additionally, FSAR Section 3.2.4.3 provides design criteria for the CRD systems insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents [GDC-4].

Additionally, FSAR Section 7.1 provides criteria for protection systems insofar as it requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in event of AOOs [GDC-29].

2.8.4.1.2 Technical Evaluation

Introduction

The Control Rod Drive Mechanisms (CRDM) position the control rods within the reactor core for controlling reactivity using water tight reluctance motors that drive a non-rotating translating lead screw coupled to the control rods. The CRDMs also provide indication of control rod position in the core. In addition, for conditions where a rapid reactor shutdown is required (Scram), the drive mechanism for each CRDM releases allowing rapid insertion of the Control Rod Assembly (CRA) by gravity. Specific design criteria are described in FSAR Section 3.2.4.3.

Description of Analyses and Evaluations

The effects on the CRDM associated with increasing reactor core power from 2609 MWt to 3014 MWt are:

- Increased thermal effects due to increased Reactor Coolant System (RCS) and reactor vessel head temperatures
- Increased heat load to the CRDM cooling system due to the higher reactor vessel head temperatures (see Section 2.7.7, Reactor Building Ventilation Systems)

No changes in RCS design or operating pressures are being made as part of the power uprate. The design pressure of 2500 psig still envelopes the primary coolant pressure that the CRDMs will be subjected to inside the reactor vessel after the EPU implementation. There is no change in fuel design before and after the EPU implementation (see Section 2.8.1, Fuel System Design). Therefore, the impact of the EPU on the CRD system primarily resides in the thermal effects at a reactor power of 3014 MWt versus the current reactor power of 2609 MWt.

The T_{HOT} associated with the EPU is 608.7°F compared to 602.1°F for the current power level of 2609 MWt. The primary system design temperature is 650°F which includes the CRDM housings. The CRDM design contains a thermal barrier which restricts the circulation of hot primary fluid, and acts as an insulator between the reactor vessel head and the drive. Early testing performed at the Alliance Research Center and at Diamond Power Company confirmed that with simulated reactor vessel (RV) head fluid temperatures in the range of about 600°F to 615°F, the temperature in the region below the stator/rotor assembly was in the range of 290°F to 310°F.

In addition, cooling water flow is provided to a water jacket surrounding each stator assembly. This cooling water flow, which has a minimum flow rate of 2 gpm per drive at a temperature of 60°F to 120°F, is the main heat removal mechanism in the stator region and, as such, serves to maintain the temperatures in this region well below the CRDM stator qualification temperature of 175°C (347°F) This was confirmed during CRDM gasket testing which included a thermocouple mounted on the stator water jacket. The stator water jacket temperature showed very little change during heatup and steady state operation, with the maximum temperature reaching about 120°F.

The 6.6°F maximum increase in hot leg temperature associated with the EPU will result in a small increase in the temperatures in the region of the CRDM stator assembly due to the attenuation effects in axial thermal gradient in the CRDM, and the local cooling effects provided by the cooling water jacket. Therefore, these effects and the existing margin to the CRDM stator qualification temperature will ensure that the CRDM stator operating temperatures after the EPU implementation will be acceptable.

The negligible change in RCS flow under the EPU conditions presented in Section 1.1, Table 1.1-1, Case 3 have a negligible impact on control rod insertion times and the rod travel times specified in FSAR Table 3-27 following a Scram. No change to FSAR Table 3-27 are considered necessary.

With the parameters for CRDM operation meeting design requirements, as noted above, the system will be capable of functioning in compliance with the licensing basis for CRDM operation after implementing the proposed EPU.

Results

Based on the evaluation above, the CR-3 plant, with respect to the CRDM System, will remain in compliance with the current licensing basis after implementation of the EPU. Specific analysis results are as follows:

- The additional heat load to the CRDM water cooling system will have a minimal effect on the CRDM stator temperature. After the EPU implementation, the CRDM stator temperature will not exceed the stator qualification temperature of 347°F.
- The EPU conditions will have a negligible impact on the control rod insertion times and the rod travel times specified in FSAR Table 3-27following a Scram.

2.8.4.1.3 Conclusion

CR-3 has reviewed the analyses related to the effects of the proposed EPU on the functional design of the CRDS. CR-3 concludes that the evaluation has adequately accounted for the effects of the proposed EPU on the system and demonstrated the system's ability to effect a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. CR-3 further concludes that the evaluation has demonstrated that sufficient cooling exists to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU. Based on this, CR-3 concludes that the Fuel System Design and associated analyses will continue to meet the requirements of FSAR Sections 1.4.27, 1.4.28, 1.4.29, 1.4.30, 1.4.31, 1.4.32, 3.2.4.3, and 7.1 following implementation of the proposed EPU. Therefore, CR-3, finds the proposed EPU acceptable with respect to the Functional Design of the Control Rod Drive System.

2.8.4.1.4 References

None

2.8.4.2 Overpressure Protection During Power Operation

2.8.4.2.1 Regulatory Evaluation

Overpressure Protection During Power Operation for the reactor coolant pressure boundary (RCPB) is provided by relief and safety valves and the reactor protection system (RPS). The CR-3 review covered pressurizer relief and safety valves and the piping from these valves to the quench tank (pressurizer relief tank).

The NRC acceptance criteria for Overpressure Protection During Power Operation are based on:

- GDC-15 - insofar as it requires that the Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including Anticipated Operational Occurrences (AOOs), and

- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the general GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria overpressure protection for the RCPB during power operation:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. [GDC-15]

- FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention; and FSAR 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention, , insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and that the probability of rapidly propagating fracture is minimized. [GDC-31]

2.8.4.2.2 Technical Evaluation

Introduction

This section briefly summarizes analyses documented elsewhere within the Technical Report. Events of different classifications were reviewed to determine the most limiting within each category with respect to overpressure protection. The post-EPU margin discussion for each of those events is contained in their respective sections.

The limiting Condition II (as described in Section 2.8.5.0, Non-LOCA Analyses Introduction) event with respect to primary system overpressurization is the Uncontrolled Control Rod Assembly Withdrawal from Low Power, or Startup Accident (Section 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition). With respect to secondary system overpressurization, the limiting Condition II event is the Loss of Load (turbine trip) accident (Section 2.8.5.2.1, Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure). For Condition II events, primary and secondary pressures must remain below 110% of their respective design pressures at all times during the transient.

The limiting Condition IV (as described in Section 2.8.5.0, Non-LOCA Analyses Introduction) Control Rod Ejection accident provides the greatest challenge to the RCS pressure limits (Section 2.8.5.4.6, Spectrum of Rod Ejection Accidents). For unexpected system excess pressure transients, the results shall be such that the design limitations of ASME Service Level C are not exceeded. The acceptance criterion that is applied is that peak RCS pressure shall not exceed 3200 psia. Peak pressures below 3200 psia demonstrate that the Service Level C requirements are met (Reference 1).

The technical evaluation of the piping from the safety valves to the Pressurizer Relief Tank (PRT) is included in Section 2.5.2, Reactor Coolant Drain Tank.

Description of Analyses and Evaluations

See Section 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, for details of the startup accident analysis performed in support of the EPU. The results showed that the peak primary system pressure is 2746 psia compared with a limit of 2764.7 psia (110% of design).

Details of the turbine trip analysis performed considering the EPU conditions are given in Section 2.8.5.2.1 which demonstrates that the secondary pressures limits are met. Specifically, the maximum pressure in the steam lines is 1153 psia compared with a limit of 1169.7 psia (110% of design) and the maximum pressure in the steam generators is 1167 psia compared with a limit of 1279.7 psia (110% of design).

The FWLB analysis and results are described in Section 2.8.5.2.4, Feedwater System Pipe Breaks Inside and Outside Containment. The results of the FWLB analysis for the EPU have shown that the peak RCS pressure for a double-ended guillotine break is 2896 psia versus the limit of 3014.7 psia (120% of design).

As discussed in Section 2.8.5.4.6, Spectrum of Rod Ejection Accidents, the peak RCS pressure for the spectrum of rod ejection accidents analyzed was determined to be 3131 psia versus the limit of 3200 psia. The peak pressures experienced for Anticipated Transients Without Scram (ATWS) events (Section 2.8.5.7, Anticipated Transients Without Scram) are bounded by these results.

Results

No changes were necessary to the primary or secondary relief or safety valves in order to meet the applicable pressure limits. All field setpoints and flow capacities remain unchanged.

The analyses demonstrate that the applicable pressure limits continue to be met for CR-3 at the EPU conditions. No changes to any control or protection setpoints or any valve capacities were necessary to meet these limits.

2.8.4.2.3 Conclusion

CR-3 has reviewed the analyses related to the effects of the proposed EPU on the Overpressure Protection During Power Operation. CR-3 concludes that the analyses have (1) adequately accounted for the effects of the proposed EPU on pressurization events and overpressure protection features, and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded

Based on this, CR-3 concludes that the overpressure protection features will continue to provide adequate protection to meet the CR-3 current licensing basis requirements with respect to FSAR Sections 1.4.9, 1.4.34, and 1.4.35 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Overpressure Protection During Power Operation.

2.8.4.2.4 References

1. NUREG-1780, “Regulatory Effectiveness of the Anticipated Transient Without Scram Rule”.

2.8.4.3 Overpressure Protection During Low Temperature Operation

2.8.4.3.1 Regulatory Evaluation

Overpressure Protection During Low Temperature Operation for the reactor coolant pressure boundary (RCPB) of the plant is provided by pressure-relieving systems that function during low temperature operation. The CR-3 review covered relief valves with piping to the quench tank, the Makeup and Letdown system, and the Residual Heat Removal (RHR) System which may be operating when the primary system is water solid.

The NRC's acceptance criteria for the Overpressure Protection During Low Temperature operation are based on:

- GDC-15, insofar as it requires that the Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs), and

- GDC-31, in so far as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs [GDC-15]; and

- FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention and FSAR Section 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention, insofar as they require that the RCPB be designed with sufficient margin to assure that it behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized, [GDC-31].

2.8.4.3.2 Technical Evaluation

Introduction

This section addresses the impact that the EPU has on the overpressure protection of the reactor coolant system (RCS). The low temperature over-pressure protection (LTOP) System controls RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the pressure and temperature (P/T) limits of ASME Code Section XI, Appendix G. Specifically, the LTOP System is designed to mitigate pressure transients which cause a rapid increase in RCS pressure when the RCS is in a water solid condition in Modes 4, 5, and 6. The types of transients evaluated for CR-3, in Modes 4, 5, and 6, relate to either mass input or heat input: actuating the High Pressure Injection (HPI) System, discharging the core flood tanks (CFTs), energizing the pressurizer heaters, failing the makeup control valves open, losing decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer. The Power Operated Relief Valve (PORV) setpoint and Technical Specification limit are not directly a function of the rated power level, and do not change as a direct result of the power uprate.

A PORV is required to provide reactor vessel LTOP at the low pressure setpoint and limit unnecessary challenges to the code safety valves. This LTOP low pressure setpoint is selected when reactor coolant (RC) temperature is below the LTOP enable temperature to provide LTOP relief.

The CR-3 LTOP System provides over-pressure protection in Modes 4, 5, and 6 as specified by CR-3 Improved Technical Specification (ITS) 3.4.11.

Description of Analyses and Evaluations

The EPU impact on pressure/temperature (P/T) limits is addressed in Sections 2.1.1 ,Reactor Vessel Material Surveillance Program, 2.1.2 ,Pressure and Temperature Limits and Upper-Shelf Energy, and 2.1.3 (Pressurized Thermal Shock).

The current LTOP requirements of ITS 3.4.11 are based on P/T limits analyzed to 32 Effective Full Power Year (EFPY) (relative to 2544 MWt, or 27.5 EFPY relative to the EPU power). The existing LTOP PORV setpoint and Technical Specification limit were developed for operating conditions (specifically heatup and cooldown rates) at low temperatures that are independent of power level. Therefore, the PORV setpoint and Technical Specification limit are not required to change as a direct result of the EPU; up to the current analyzed condition of 32 EFPY (relative to 2544 MWt) pressure vessel integrated fluence. The plant, at the end of CY16, has accumulated approximately 23 EFPY integrated fluence equivalent (relative to 2544 MWt), and will have more than a full cycle available at the EPU power level until the limit is reached.

In support of future operation at the EPU power, P/T limits have been re-analyzed for a reactor vessel fluency of 50.3 EFPY at the EPU conditions and the resultant LTOP PORV setpoint will be implemented when the revised P/T limit curves are implemented, prior to exceeding 27.5 EFPY (relative to the EPU power). Maintaining the existing LTOP PORV setpoint until this time continues to provide appropriate low-temperature protection for the EPU conditions. Implementation of revised LTOP PORV setpoint will require separate submittal, and NRC approval of an Amendment to ITS 3.4.11. Submittal of this Amendment request will be made at least 12 months prior to reaching 27.5 EFPY (relative to the EPU power).

Results

The existing LTOP PORV setpoint and Technical Specification limit were developed for operating conditions (specifically heatup and cooldown rates) at low temperatures that are independent of power level. Therefore, the PORV setpoint and Technical Specification limit do not change as a direct result of the EPU, for plant operation up to the current analyzed condition of 32 EFPY pressure vessel integrated fluence (relative to 2544 MWt, or 27.5 EFPY relative to the EPU power)..

2.8.4.3.3 Conclusion

CR-3 has reviewed the analyses related to the effects of the proposed EPU on the Overpressure Protection During Low Temperature operation. CR-3 concludes that:

(1) The analyses adequately accounted for the effects of the EPU on pressurization events and overpressure protection features; and

(2) The plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded.

Based on this, CR-3 concludes that the low temperature overpressure protection features will continue to provide adequate protection to meet the CR-3 current licensing basis requirements with respect to FSAR Sections 1.4.9, 1.4.34 and 1.4.35 following implementation of the EPU. Therefore, CR-3 finds the proposed EPU is acceptable with respect to Overpressure Protection During Low Temperature Operation.

2.8.4.3.4 References

None

2.8.4.4 Residual Heat Removal System

2.8.4.4.1 Regulatory Evaluation

The Decay Heat Removal (DH) System is used to cool down the Reactor Coolant System (RCS) following shutdown. The DH system is typically a low pressure system that takes over the shutdown cooling function when the RCS temperature is reduced. The CR-3 review covered the effect of the proposed EPU on the functional capability of the DH System to cool the RCS following shutdown and provide decay heat removal.

The NRC's acceptance criteria for the Decay Heat Removal System are based on:

- GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects;
- GDC-5, insofar as it requires SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions; and
- GDC-34, which specifies requirements for a Residual Heat Removal (RHR) System.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.40, Missile Protection, insofar as it requires SSCs important to safety to be protected against dynamic effects [GDC-4];
- FSAR Section 1.4.4, Sharing of Systems, insofar as it requires SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions [GDC 5]; and
- FSAR Sections 1.4.37, Engineered Safety Features Basis for Design, and 1.4.42, Engineered Safety Features Components Capability, which specify requirements for an RHR System [GDC 34].

2.8.4.4.2 Technical Evaluation

Introduction

The DH Removal System is described in FSAR Section 9.4. The primary function of the DH System is to remove decay heat from the core and reduce the temperature of the RCS during the second phase of a plant cooldown. During the first phase of the cooldown, the temperature of the RCS is reduced to 280°F by transferring heat to the steam generators. In the second phase, DH System is placed in service at or below a RCS temperature of 280°F to complete the RCS cooldown to ≤ 200°F.

Description of Analyses and Evaluations

The EPU increases the residual heat generated in the core during normal cooldown and refueling operations. This provides a higher heat load on the DH Heat Exchangers (HX) during cooldown and also during refueling outages.

The EPU affects the plant cooldown time(s) since the core thermal power increases, and therefore, the decay heat increases. A plant cooldown calculation was performed for the EPU conditions to demonstrate that the DH System continues to comply with its design basis functional requirements and performance criteria for plant cooldown. The one-train system alignment was considered to address the design capability in the FSAR Section 9.4. In addition, a cooldown calculation was performed to support the worst-case scenario for the 10 CFR Part 50 Appendix R fire hazards and safe shutdown analysis. Also, a calculation was performed that demonstrates the existing technical specification cooldown time limits will be achieved at the EPU conditions.

Normal Plant Cooldown

During normal operations, the steam generators in conjunction with the steam dump valves reduce the reactor coolant temperature to 280°F. Decay heat cooling is then initiated with the pump taking suction on the RCS hot leg line and discharging through the DH HXs into the reactor vessel. The EPU increases the decay heat generated in the core during normal cooldown. This provides a higher heat load on the DH HXs during cooldown and also during refueling outages. The increased heat loads will be transferred to the Decay Heat Closed Cycle Cooling (DC) System and ultimately to the Nuclear Services Seawater System and the Decay Heat Seawater System which together comprise the RW System.

A calculation was performed to determine the DH HXs performance after the EPU implementation. The normal plant cooldown time for initiating refueling (Mode 6) is 140°F. The results are summarized in Table 2.8.4.4-1. Since there is no design criterion for normal plant cooldown times, these increases in calculated values are acceptable.

Appendix R Cooldown

The Appendix R Safe Shutdown requires that Cold Shutdown (Mode 5 $\leq$ 200°F) be achieved in 72-hours after reactor shutdown. This requirement is addressed in Section 2.5.1.4, Fire Protection. Based on analysis, the DH System remains capable of meeting Appendix R requirements with no DH System changes required.

Technical Specifications Cooldown

The Improved Technical Specifications contain actions that could require that the plant be in Cold Shutdown (Mode 5) within 36 hours. A calculation was performed at the EPU conditions to demonstrate continued capability to achieve compliance with the Improved Technical Specifications. The calculation conservatively assumes that the DH System is placed into service 6 hours after shutdown which is the time needed for the steam generators, EFW and the steam dump valves to have reduced the RCS temperature to 280°F. This conservative assumption maximizes the decay heat load. For a controlled cooldown from power operation, additional time to reach shutdown and commencing the initial cooldown would increase the decay time. The cooldown time limits to achieve Mode 5 ($T_{AVG} \leq$ 200°F) were determined at the EPU conditions for one train and two train operations. The results are summarized in Table 2.8.4.4-2. Based on the results, the DH System is capable of meeting the Improved Technical Specifications cooldown requirements.

Results

Continued compliance with the DH System cooldown performance requirements was demonstrated at the EPU conditions with no DH System changes necessary.

The EPU cooldown calculation results are as follows:

- The normal plant cooldown time from 280°F to 140°F with one train of DH increased by 83 hours, from 189 hours for the current power rating to 272 hours for the EPU conditions. With two trains of DH, the time increased by 19 hours from 34 hours, for the current power rating, to 53 hours for the EPU condition. There is no design criterion for normal plant cooldown time to refueling conditions.

- At the EPU conditions, the time required to achieve cold shutdown, increased by 2.1 hours, from 66.7 for the current power rating to 68.8 hours for the EPU conditions. For Appendix R Safe Shutdown, cold shutdown will continue to be achieved within the 72-hour time limit.

- With both trains of DH and DC equipment, the time required to cooldown from 280°F to 200°F increased from 7.3 hours to 7.5 hours at the EPU conditions if DH operation is initiated 6.0 hours after reactor shutdown. With one train of DH and DC, the time required to cooldown from 280°F to 200°F increased from 16.2 hours to 23.8 at the EPU conditions if DH operation is initiated 6 hours after reactor shutdown. The DH System will continue to meet cooldown requirements to support Improved Technical Specification compliance.

2.8.4.4.3 Conclusion

CR-3 has reviewed the analyses related to the effects of the proposed EPU on the DH System. CR-3 concludes that the analyses of the proposed EPU has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the DH System will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this evaluation, CR-3 concludes that the DH System will continue to meet the requirements of FSAR Sections 1.4.4, 1.4.37, 1.4.40, and 1.4.42 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Decay Heat Removal System.

2.8.4.4.4 References

None

Table 2.8.4.4-1: Decay Heat System Normal Cooldown Performance

	Pre-EPU Time to achieve 140°F (hours)	Post EPU Time to achieve 140°F (hours)
1 DH System operating	189	272
2 DH Systems operating	34	53

Table 2.8.4.4-2: Decay Heat System Improved Technical Specification Cooldown Performance

	Time Required from 280°F to Mode 5 with 1 DH System (hours)	Time Required from 280°F to Mode 5 with 2 DH Systems (hours)
Pre–EPU	16.2	7.3
Post-EPU	23.8	7.5

2.8.5 Accident and Transient Analyses

2.8.5.0 Non-LOCA Analyses Introduction

This Section summarizes the Non-Loss-of-Coolant Accident (LOCA) Transient Analyses and evaluations performed to support the EPU program at CR-3.

2.8.5.0.1 Fuel Design Mechanical Features

The fuel design currently in use at CR-3 and planned for use in the EPU is the 15x15 Mark-B-HTP. Detailed information for the Mark-B-HTP fuel design is provided in Section 2.8.1, (Fuel System Design).

With respect to the Non-LOCA Transient Analyses, the effects of fuel design mechanical features were accounted for in fuel-related input assumptions, such as fuel and cladding dimensions, cladding material, fuel temperatures, and core bypass flow.

2.8.5.0.2 Peaking Factors

For Mark-B-HTP fuel, the core power and coolant flow distributions used in the Departure from Nucleate Boiling (DNB) analyses of three non-LOCA transients – one pump coastdown, four pump coastdown, and locked rotor – are characterized by peaking factors as detailed in Section 2.8.3 (Thermal and Hydraulic Design). The pin power and local heat flux hot channel factors increase the average and local (hot channel) heat generation rates, respectively, and a flow area reduction factor is applied to bundle flow areas across the core. These are applied to account for the effects of fuel manufacturing variations.

2.8.5.0.3 EPU Program Features

Key EPU Program features that were considered in the non-LOCA transient analyses were as follows:

- An initial core power level of 3014 MWt (increase of approximately 15.5%).
- A Reactor Coolant System (RCS) thermal design flow of 374,880 gpm, which includes considerations for the EPU and 5% OTSG tube plugging.
- A nominal full power T_{AVG} of 582°F and a nominal operating pressure of 2170 psia (measured at hot leg pressure tap).
- A nominal, full-power MFW temperature of 460°F.
- A nominal turbine header pressure of 930 psia.

The inputs chosen for the analysis are generally consistent with the current licensing basis of the plant. However, for some inputs defined by the Improved Technical Specifications (ITS), changes were required in order to support the extended power uprate. The proposed ITS changes are described in Attachment 2, Operating License and Technical Specification Changes (Markup). The major changes affecting the non-LOCA analyses are described below.

- The minimum required shutdown margin at hot zero power, with the most reactive Rod Cluster Control Assembly (RCCA) fully withdrawn, is increased from 1.0%Δk/k to

1.3%Δk/k. The increase in shutdown margin was required for the mitigation of the steam line break event to prevent post-trip return to criticality, as described in Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment.

- The minimum Borated Water Storage Tank (BWST) concentration is increased from 2270 ppmB to 2600 ppmB to support safety analysis Post-LOCA Sump boron concentration requirements and boron concentration requirements for cold shutdown, as described in Section 2.8.4.1, Functional Design of Control Rod Drive System. This change was credited in the main steam line break analysis, as described in Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment.

2.8.5.0.4 Other Major Assumptions

Table 2.8.5.0-1 lists the non-LOCA initial condition assumptions used. Other major assumptions considered in the non-LOCA transient analyses are discussed below:

- Staggered lift setpoints were modeled for the Main Steam Safety Valves (MSSVs) using plant-specific setpoints. The nominal setpoints are shown in Table 2.8.5.0-2. The analyses modeling MSSVs considered lift tolerances of up to 3% of the lift pressure, with an accumulation of up to 3%.
- The lift setpoint for the pressurizer safety valve (PSV) was modeled assuming a nominal setpoint of 2500 psig. A lift tolerance of 3% of the lift pressure was considered.
- The decay heat of 1.0 times the ANS 1971 decay heat standard plus B&W heavy isotopes is used for all non-LOCA accidents except MSLB. For MSLB, 0.9 times the ANS 1971 decay heat standard (with no actinides) is used.
- A nominal core design bypass flow percentage of 6.7% was considered for the Mark-BHTP fuel.

2.8.5.0.5 RPS and ESAS Functions Assumed in Analyses

Table 2.8.5.0-3 contains a list of the different Reactor Protection System (RPS) and Engineered Safeguards Actuation System (ESAS) functions credited in the Non-LOCA Transient Analyses. The safety analysis setpoints, as well as the time delays associated with each of these functions, are also presented in Table 2.8.5.0-3. The setpoints are consistent with the current Technical Specifications with appropriate instrument uncertainties applied for the specified event. As demonstrated by the various analyses addressed in Section 2.8.5, none of the Technical Specification RPS or ESAS setpoint values was required to change for the EPU.

2.8.5.0.6 Reactivity Coefficients

The transient response of the reactor core is dependent on reactivity feedback effects, in particular the Moderator Temperature Coefficient (MTC) and the Doppler coefficient. Depending upon event-specific characteristics, conservatism dictates the use of either maximum or minimum reactivity coefficient values. Justification for the use of the reactivity coefficient values was treated on an event-specific basis. Table 2.8.5.0-4 presents the core kinetics parameters and reactivity feedback coefficients assumed in the Non-LOCA Analyses.

2.8.5.0.7 Control Rod Assemblies (CRAs) Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the acceleration of the CRAs and the variation in rod worth as a function of rod position. With respect to the Non-LOCA Transient Analyses, the critical parameter was the time from beginning of CRA insertion to dashpot entry, or approximately 2/3 of the CRA travel, although negative reactivity addition continued to be modeled until rods were completely inserted. For the Non-LOCA Analyses, the assumed insertion time from fully withdrawn to 2/3 insertion was 1.4 seconds. Two tables relating to CRA drop time and reactivity worth are presented in this report. The normalized rod worth versus the time after trip for power levels greater than 15% Full Power (FP) is presented in Table 2.8.5.0-5. The normalized rod worth versus the time after trip for power levels less than 15% FP is presented in Table 2.8.5.0-6.

2.8.5.0.8 Computer Codes Utilized

Summary descriptions of the principal computer codes used in the Non-LOCA Transient Analyses are provided below. Table 2.8.5.0-1 lists initial conditions used in each of the Non-LOCA Analyses.

RELAP5/MOD2-B&W

The RELAP5/MOD2-B&W code (Reference 1) has been approved by the NRC for use in on-LOCA safety analyses (Reference 2). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The RCS model provides for heat transfer from the core, transport of the coolant to the steam generators (SGs), and heat transfer to the SGs. The secondary model includes a detailed depiction of the Main Steam System, including steam relief to the atmosphere through the MSSVs, turbine bypass valves (TBVs), and simulation of the Turbine Stop Valve (TSVs). The secondary model also includes the delivery of feedwater, both main and emergency, to the SGs.

COPERNIC

COPERNIC is a fuel performance code (Reference 3) that is used to obtain the gap conductance for both NEMO-K and LYNXT. The fuel property correlation equations from COPERNIC are used in NEMO-K and to develop inputs for LYNXT. The only application of COPERNIC within the Non-LOCA Analyses is the Rod Ejection Accident (Section 2.8.5.4.6, Spectrum of Rod Ejection Accidents).

NEMO-K

NEMO-K is a 3-dimensional kinetics code (Reference 4) that is used to set initial boundary conditions for the ejected rod transient and to simulate the ejected rod transient. If there is not a high flux trip, the core power response from NEMO-K is input to RELAP5/MOD2.

LYNXT

The LYNXT computer program performs thermal-hydraulic calculations (Reference 5). This code calculates coolant density, mass velocity, enthalpy, void fractions, quality, static pressure, steady state and transient Departure from Nucleate Boiling Ratio (DNBR) distributions, and cross-flow along flow channels within a reactor core. LYNXT is used in Departure from Nucleate Boiling (DNB) calculations for both steady state and transient conditions, and to provide pressure drop inputs to statistical holddown calculations.

All of the codes listed above are approved by the NRC and are acceptable for use in the licensing analyses performed in support of the EPU. The codes listed above are each referenced in the B&W reload analysis methodology (Reference 7), which applies to the current licensing basis for the CR-3 plant.

2.8.5.0.9 Events Evaluated or Analyzed

With the exception of the uncompensated operating reactivity changes evaluation discussed below, each of the FSAR transients listed in Table 2.8.5.0-7 was analyzed in support of the EPU conditions. These transient evaluation and analyses demonstrate that all applicable safety analysis acceptance criteria are satisfied for the EPU, as demonstrated in the subsections of Section 2.8.5. Table 2.8.5.0-7 summarizes the results obtained for each of the Non-LOCA Transient Analyses. The Non-LOCA Analyses did not identify the need to add any new system components to achieve accident mitigation. The results of the Non-LOCA Analyses show that components credited for the accident analysis will be exposed to acceptable internal and external environments. Uncompensated operating reactivity changes transient was evaluated at EPU conditions. There are two acceptance criteria for this accident. First, the rate of reactivity will be much less than the rate at which the operator can compensate for the addition. Second, the rate of temperature change will be much less than the rate at which the automatic control system can compensate for the change. The plant and control system response to reactivity changes resulting from fuel depletion, burnable poison depletion, and changes in fission product poison concentration are not significantly affected by the initial core power level. As a result, the change in the magnitude of reactivity changes caused by fuel depletion, burnable poison depletions, and/or changes in fission product poison concentration will be negligible. The analysis was initiated at 2575 MWt and is insensitive to initial core power. As such, an increase in the analyzed power to 100.4% of 3014 MWt does not result in any appreciable change in the accident as previously analyzed.

The normal operation and possible transient modes of nuclear plants are categorized into four conditions commonly referred to as normal, moderate frequency, infrequent events, and limiting faults. The specific definitions for these conditions are taken from Reference 8, and are described in the following table.

Condition	Description	Applicable Events
I	Condition I events are those that are expected frequently or regularly in the normal course of power operation. The design requirement for these events is that they shall be accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.	N/A
II	Condition II events are those that are expected to occur during the life of a plant that may result in reactor shutdown. The design requirement for these events is that they shall be accommodated with, at most, a shutdown of the reactor with the plant capable of returning to power operation after corrective action.	2.8.5.1.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve 2.8.5.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam-Pressure Regulatory Failure 2.8.5.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries 2.8.5.2.3 Loss of Normal Feedwater 2.8.5.3.1 Loss of Forced Reactor Coolant Flow 2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition 2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power 2.8.5.4.3 Control Rod Mis-Operation 2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature 2.8.5.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant 2.8.5.5 Inadvertent Operation ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory 2.8.5.6.1 Inadvertent Opening of a Pressurizer Pressure Relief Valve

Condition	Description	Applicable Events
III	Condition III events are incidents that may occur infrequently, if at all, during the life of the plant. The design requirement for these events is that they shall not cause more than a small fraction of the fuel elements in the reactor to be damaged, although sufficient fuel element damage might occur to preclude resumption of operation for a considerable outage time.	N/A
IV	These events are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Condition IV events represent the limiting design case. The design requirement for these events is that they shall not cause a release of radioactive material that result in an undue risk to public health and safety exceeding the 10 CFR 50.67 limits. A single Condition IV event shall not cause a consequential loss of system functions needed to cope with the event.	2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment 2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment 2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break 2.8.5.4.6 Spectrum of Rod Ejection Accidents 2.8.5.6.2 Steam Generator Tube Rupture

The NRC categorizes plant operation into three conditions (Reference 7). They are normal operation, anticipated operational occurrences, and accidents. Compliance with the NRC regulations is assured by requiring the limiting Condition III transient to meet the acceptance criteria for Condition II events.

2.8.5.0.10 Analysis Methodology

The transient-specific analysis methodologies that were applied to CR-3 have been reviewed and approved by the NRC via transient-specific topical reports and/or through the review and approval of plant-specific safety analysis reports. The approved codes and methods used in the extended power uprate analyses are described in References 1 through 6. Reference 6 is a transient-specific topical that has been approved by the NRC; the extended power uprate analyses represent the initial implementation of the methodology for CR-3.

2.8.5.0.11 Operator Actions

The events for which operator actions are credited in the analysis include the Station Blackout (Section2.3.5) and Steam Generator Tube Rupture (Section 2.8.5.6.2).

2.8.5.0.12 References

1. AREVA NP Document 43-10164PA-06, RELAP5/MOD2-B&W An Advanced Computer Program For Light Water Reactor LOCA and Non-LOCA Transient Analysis.

2. AREVA NP Document 43-10193PA-00, RELAP5/MOD2-B&W For Safety Analysis of B&W Designed Pressurized Water Reactors.

3. BAW-10231PA, Revision 1, "COPERNIC Fuel Rod Design Computer Code," Framatome ANP, January 2004.

4. BAW-10221PA, "NEMO-K a Kinetics Solution in NEMO," September 1998.

5. BAW-10156A, Revision 1, "LYNXT Core Transient Thermal-Hydraulic Program," B&W Fuel Company, August 1993.

6. ANP-2788P, "Crystal River 3 Rod Ejection Accident Methodology Report," February 2009.

7. BAW-10179PA, Revision 007, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses".

8. ANSI/ANS-57.5-1981.

Table 2.8.5.0-1 Non-LOCA Plant Initial Conditions

Parameter	Value	Notes
NSSS power (MWt)	3014	
Total net RCP heat (MWt)	16.4	1
RCS flow – minimum DNB flow (gpm)	374,880	-
HFP T_{AVG}, (°F)	582	-
HZP T_{AVG} (°F)	532	-
RCS pressure – hot leg (psia)	2170	-
Steam pressure – turbine header (psia)	930	-
Nominal pressurizer level (inches)	220	2
Main feedwater temperature (°F)	460	3
OTSG water level (% OR)	~ 70	4

Notes:
1. Total RCP heat input minus RCS thermal losses.
2. An uncertainty of 20 inches was applied when conservative.
3. The HFP MFW temperature is applicable to power levels between 50 %FP and 102 %FP.
4. The steam generator water level is a target since the mass inventory is determined by the computer code to achieve the required heat balance. The MSLB uses a targeted level of ≥ 96% OR to maximize the initial steam generator inventory.

Table 2.8.5.0-2 – MSSV Setpoints

Number of MSSVs per Steam Generator	Setpoint (psig)
2	**1050**
2	**1070**
2	**1090**
1	**1100**
1	**1100**

Note: All of the valves are of identical size with the exception of one of the 1100 psig setpoint valves, which is slightly smaller in capacity.

Table 2.8.5.0-3 Summary of RPS and ESAS Functions Actuated

FSAR Section	Event Description	RPS and ESAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
14.1.2.1	Uncompensated Operating Reactivity Changes	N/A	N/A	N/A
14.1.2.2	Startup Accident (Section 2.8.5.4.1)	RPS High RCS Pressure Rx Trip	2400 psia	0.61 sec
14.1.2.3	Rod Withdrawal at Rated Power Operation Accident (Section 2.8.5.4.2)	RPS High RCS Pressure Rx Trip	2400 psia	0.61 sec
		RPS High Flux Rx Trip	112 %RTP	0.42 sec
14.1.2.4	Moderator Dilution Accident (Section 2.8.5.4.5)	RPS High RCS Pressure Rx Trip	2400 psia	0.61 sec
14.1.2.5	Cold Water Accident (Section 2.8.5.4.4)	N/A	N/A	N/A
14.1.2.6	Loss-of-Coolant Flow Accidents (Sections 2.8.5.3.1 & 2.8.5.3.2)	4 Pump Coast Down (PCD): RPS Power/Pump Monitors Rx Trip	Note 1	1.60 sec
		1 PCD: RPS Flux/Flow Rx Trip	1.13 %FP/%flow	2.18 sec
		Locked Rotor: RPS Flux/Flow Rx Trip	1.13 %FP/%flow	2.18 sec
14.1.2.7	Stuck-Out, Stuck-In, or Dropped Control Rod Accident (Section 2.8.5.4.3)	RPS Low RCS Pressure Rx Trip	1893.95 psia	0.61 sec
14.1.2.8	Load Rejection Accident / Turbine Trip (Section 2.8.5.2.1)	RPS High RCS Pressure Rx Trip	2400 psia	0.61 sec
14.2.2.1	Steam Line Failure Accident (Section 2.8.5.1.2)	RPS Low RCS Pressure Rx Trip	Note 2	Note 2
		ESAS Low RCS Pressure Trip		
14.2.2.2	Steam Generator Tube Rupture Accident (Section 2.8.5.6.2)	RPS Low RCS Pressure Rx Trip	1893.95 psia	0.61 sec
		ESAS Low RCS Pressure Trip	1714 psia (Note 5)	0 sec

Table 2.8.5.0-3 Summary of RPS and ESAS Functions Actuated

FSAR Section	Event Description	RPS and ESAS Signal(s) Actuated	Analysis Setpoint	Delay (sec)
14.2.2.3	Fuel Handling Accidents (N/A)	N/A	N/A	N/A
14.2.2.4	Rod Ejection Accident (Section 2.8.5.4.6)	RPS High Flux	112 %RTP	0.42 sec
		RPS High RCS Pressure	2400 psia	0.61 sec
		RPS High RCS Temperature	620°F	5.67 sec
		RPS Variable Low Pressure (Note 3)	$11.59*T_{HOT}-5049.46$	5.67 sec
14.2.2.6	Makeup System Letdown Line Failure Accident (N/A)	ESAS Low RCS Pressure Trip	1640 psia	0 sec
14.2.2.7	Maximum Hypothetical Accident (N/A)	N/A	N/A	N/A
14.2.2.8	Waste Gas Decay Tank Rupture Accident (N/A)	N/A	N/A	N/A
14.2.2.9	Loss of Feedwater (Section 2.8.5.2.3)	RPS High RCS Pressure Rx Trip	2400 psia	0.61 sec
14.2.2.9	Main Feedwater Line Break (Section 2.8.5.2.4)	RPS High RCS Pressure Rx Trip	2445.45 psia (Note 4)	0.61 sec

Table 2.8.5.0-3 Summary of RPS and ESAS Functions Actuated Notes:

1. The 4 PCD event modeled a reactor trip on loss of power to the RCPs, which was initiated at time zero with a delay of 1.60 sec.
2. The RPS reactor trip signal and ESAS low RCS pressure trip were both assumed to occur at event initiation for conservatism.
3. The Variable Low Pressure Trip (VLPT) function is not currently credited in the safety analyses (pre-EPU), however it is an existing trip function with settings validated on a cycle-by-cycle basis. The VLPT is credited for the Rod Ejection Accident (Section 2.8.5.4.6).
4. The uncertainty added to the nominal high RCS pressure setpoint includes the effects of RB elevated pressure.
5. The nominal setpoint is biased high to maximize SGTR leakage.
6. The Letdown Line Failure Accident is not considered a credible Chapter 14 event and is analyzed for radiological purposes only in Section 2.9.2, “Radiological Consequence Analyses”.

Table 2.8.5.0-4 Core Kinetics Parameters and Reactivity Feedback Coefficients		
Parameter	**Beginning of Cycle**	**End of Cycle**
HFP MTC, $\Delta k/k/^{o}F$ HZP MTC, $\Delta k/k/^{\circ}F$	0.0 +0.75E-4	-5.0E-4[1] N/A
Doppler Temperature Coefficient, $\Delta k/k/^{\circ}F$	-1.30E-5	-2.0E-5
Prompt neutron generation time (µsec)	24.8	24.8
Effective delayed neutron fraction	0.007	0.0045
Initial boron concentration (ppm)	2270	N/A
Inverse boron worth (ppm/%Δk/k)	160	150

(1) Note: Refer to Section 2.8.5.1.2 for MSLB-Specific assumptions.

Table 2.8.5.0-5: Scram Curve > 15%FP	
Time After Reactor Trip	Reactivity Insertion
Sec	%
0.0	0.0
0.2	0.58
0.3	0.99
0.4	1.83
0.6	5.29
0.8	12.33
1.0	21.41
1.2	33.09
1.4	50.75
1.6	72.96
1.8	91.30
2.0	99.26
2.2	99.99
2.3	100.00

Note: This table is applicable for all safety analyses initiated above 15% power, except the control rod ejection accident (CREA) fuel rod performance analyses. For the CREA analyses, the reactivity insertion as a function of time is calculated by the code based on the rod velocity curves. The basis for the rod velocity curves is the same for CREA as for the other safety analyses.

Table 2.8.5.0-6: Scram Curve < 15%FP	
Time After Reactor Trip	Reactivity Insertion
sec	%
0.0	0.0
0.2	0.0
0.3	0.46
0.4	1.36
0.6	3.57
0.8	5.59
1.0	9.76
1.2	16.96
1.4	28.84
1.6	49.65
1.8	75.88
2.0	93.64
2.2	98.86
2.3	99.50
2.4	100.00

Note: This table is applicable for all safety analyses initiated below 15% power, except the control rod ejection accident (CREA) fuel rod performance analyses. For the CREA analyses, the reactivity insertion as a function of time is calculated by the code based on the rod velocity curves. The basis for the rod velocity curves is the same for CREA as for the other safety analyses.

Table 2.8.5.0-7 Non-LOCA EPU Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
14.1.2.1	Uncompensated Operating Reactivity Changes	N/A	N/A	N/A
14.1.2.2	Startup Accident	Peak thermal power	< 112 %RTP (Note 7)	53.6 %RTP
		Peak RCS pressure	< 2764.7 psia	2746.3 psia
14.1.2.3	Rod Withdrawal at Rated Power Operation Accident	Peak thermal power	< 112 %RTP (Note 7)	110.1 %RTP
		Peak RCS pressure	< 2764.7 psia	2674 psia
14.1.2.4	Moderator Dilution Accident	Peak thermal power	< 112 %RTP (Note 7)	109.5 %RTP
		Peak RCS pressure	< 2764.7psia	2698 psia
14.1.2.5	Cold Water Accident	DNBR	(Note 1)	(Note 1)
		Peak RCS pressure	< 2764.psia	2337 psia
14.1.2.6	Loss-of-Coolant Flow Accidents	4 PCD: DNBR	>1.45	1.54
		1 PCD: DNBR	>1.45	1.62
		Locked Rotor: DNBR	>1.45	1.40 (Note 9)
14.1.2.7	Stuck-Out, Stuck-In, or Dropped Control Rod Accident	DNBR	(Note 7)	(Note 7)
		Peak RCS pressure	< 2764.7 psia	< 2764.7 psia (Note 3)
14.1.2.8	Load Rejection Accident (Turbine Trip)	Peak RCS pressure	< 2764.7 psia	2569.8 psia
		Peak SG pressure	< 1279.7 psia	1166.6 psia
		Peak steam line pressure	< 1169.7 psia	1153 psia
		Fuel damage	(Note 1)	(Note 1)

Table 2.8.5.0-7 Non-LOCA EPU Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
14.2.2.1	Steam Line Failure Accident (core response)	Peak RCS pressure	< 2764.7 psia	< 2764.7 psia (Note 6)
		Peak thermal power	< 112 %RTP (Note 7)	< 112 %RTP (Note 6)
14.2.2.2	Steam Generator Tube Rupture Accident	Peak tensile tube-to-shell ΔT	< 100°F	~ 70°F
		Peak ruptured tube mass flow Ruptured tube flow duration	(Note 8)	(Note 8)
14.2.2.3	Fuel Handling Accidents	Radiological consequences	(Note 8)	(Note 8)
14.2.2.4	Rod Ejection Accident	Peak RCS pressure	< 3200 psia	< 3130.6 psia
		Maximum cal/g	< 150 (Note 4)	54.1 (Note 4)
		Maximum Δcal/g prompt	< 125 (Note 4)	34 (Note 4)
		Maximum fuel temperature	≤ fuel melt limit	3804°F (BOC) 1675°F (EOC) (Note 10)
		Maximum cladding temperature	≤ clad limit	1353°F (BOC) 1007°F (EOC) (Note 10)
		MDNBR/Limit for rod failure	≤1.000 (Note 5)	0.8 (Note 5)
		Equivalent nominal failed rods	<4.3%	1.4%
14.2.2.6	Makeup System Letdown Line Failure Accident	Radiological consequences.	(Note 8)	(Note 8)
14.2.2.7	Maximum Hypothetical Accident	Radiological consequences.	N/A	N/A

Table 2.8.5.0-7 Non-LOCA EPU Analysis Limits and Analysis Results

FSAR Section	Event Description	Result Parameter	Analysis Result	
			Analysis Limit	Limiting Case
14.2.2.8	Waste Gas Decay Tank Rupture Accident	Radiological consequences.	2.5 rem TEDE	(Note 11)
14.2.2.9	Loss of Feedwater	Peak RCS pressure	< 2764.7 psia	2750.7 psia
14.2.2.9	Main Feedwater Line Break	Peak RCS pressure	< 3014.7 psia (Note 2)	2896.2 psia

Notes:

1. The criterion was met by confirming that the normalized power-to-normalized flow ratio remained equal to or less than the initial value throughout the transient.
2. The RCS pressure limit is based on faulted conditions (120%of design pressure).
3. A spectrum of dropped rod cases were run at different times-in-life and all cases predicted peak RCS pressures less than 2764.7 psia.
4. These parameters and limits apply to the REA cases with initial power levels less than 5% RTP. Limiting case is EOC at HZP.
5. This parameter and limit applies to the REA cases with initial power levels greater than 5% RTP. Limiting case is BOC at 20% RTP.
6. The steam line break accident results in a decrease in primary and secondary system pressures. Following dryout and depressurization of the affected steam generator, the primary system and unaffected steam generators would repressurize to normal post-trip conditions. Consequently, system pressure limits are not challenged during a MSLB accident. The reactor is conservatively tripped at time zero and the post-trip maximum thermal power does not exceed 26% RTP.
7. The limit on peak thermal power is related to DNBR. For B&W 177-FA plants, maximum allowable peaking limits are developed at the RCS DNB safety limit statepoints. The RCS DNB safety limits represent a locus of points for which the minimum DNBR is equal to the analysis limit. These safety limits are calculated at the design overpower condition (112% of reactor thermal power). Thus, demonstration that thermal power remains below 112% provides assurance that the minimum DNBR remains above limits. The limiting DNB transients are loss-of-coolant-flow transients, which are specifically analyzed for DNBR. The acceptance criterion with respect to DNBR during the Dropped Rod accident is evaluated on a cycle-specific basis using statepoints derived from the CR-3 EPU Dropped Rod accident evaluations.
8. Radiological Consequences are provided in Table 2.9.2-1.
9. See Section 2.8.3 for discussion of the acceptability of this result.
10. The acceptance criteria for the fuel melt and clad temperature limits is proprietary to AREVA NP. The results presented meet the acceptance criterion defined by AREVA NP, as documented in topical report ANP-2788P (Reference 6). The results provided are for the cases deemed limiting at EOC and BOC with respect to fuel failure potential.
11. As discussed in Section 2.5.6.1 and Section 2.10, there is no impact from EPU on waste gas decay tank content limit, which is controlled by the Offsite Dose Calculation Manual, to maintain tank contents within the limit specified in Improved Technical Specification 5.6.2.13, Explosive Gas and Storage Tank Radioactivity Monitoring Program.

2.8.5.1 Increase in Heat Removal by the Secondary System

2.8.5.1.1 Decrease In Feedwater Temperature, Increase In Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

2.8.5.1.1.1 Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase can result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses.

The NRC’s acceptance criteria are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs);
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the Reactor Coolant Pressure Boundary (RCPB) are not exceeded during any condition of normal operations;
- GDC-20, insofar as it requires that the Reactor Protection System (RPS) be designed to initiate automatically the operation of appropriate systems, including reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs. [GDC-10]
- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the Reactor Coolant Pressure Boundary (RCPB) are not exceeded during any condition of normal operations. [GDC-15]
- FSAR Sections 1.4.14, Core Protection Systems, and 1.4.15, Engineered Safety Features Protection Systems, insofar as they require that the Reactor Protection System be designed to initiate automatically the operation of appropriate systems, including reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs. [GDC-20]
- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. [GDC-26].

2.8.5.1.1.2 Technical Evaluation

Introduction

For the excessive heat removal events addressed in this section, the CR-3 FSAR does not contain: (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) a sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the Reactor Protection System, (6) operator actions, and (7) the results of the transient analyses. However, the following qualitative evaluation is provided for the EPU conditions.

These events are bounded by other events analyzed in the FSAR. The evaluation that follows summarizes the events determined to be bounding versus the excess heat removal events.

Three scenarios by which excessive heat removal can challenge the CR-3 acceptance criteria are:

- a decrease in feedwater temperature,
- an increase in feedwater flow, and
- an increase in steam flow (includes the inadvertent opening of the steam generator relief or safety valve including atmospheric dump valves, and turbine bypass valves).

In conjunction with the EPU, CR-3 is implementing changes to feedwater and steam systems. Main Feedwater (MFW) System components such as the main feedwater and condensate booster pumps are being modified. Also, the atmospheric dump valves (ADVs) and turbine bypass valves (TBVs) are increasing in capacity. These changes have an impact on the excessive heat removal events, however, the evaluation that follows, which is based on comparison to bounding events, remains valid for the EPU. These changes are further described in Appendix E, Major Plant Modifications.

Description of Analyses and Evaluations

Decrease in Feedwater Temperature

A decrease in feedwater temperature results in overcooling of the RCS that then leads to increasing core power. The Reactor Protection System (RPS) trip on overpower (high flux) prevents any power increase that could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit. Due to the limited nature of the reactivity excursion, the plant may evolve to a new steady-state condition without initiating an RPS trip.

An extreme example of excess heat removal by the Feedwater (FW) System is the transient associated with the accidental opening of the bypass valve that diverts flow around the high-pressure feedwater heaters. In the event of an accidental opening of the bypass valve, there is a sudden reduction in inlet feedwater temperature to the steam generators. The increased subcooling would create a greater load demand on the primary system that can potentially lead to a reactor trip. Normally, the Integrated Control System (ICS) would modify the total feedwater flow demand signal to maintain a balance in the energy exchanged between the primary and secondary side of the steam generators.

The Feedwater System malfunction is a Condition II event (moderate frequency), as defined in Section 2.8.5.0, Non-LOCA Analysis Introduction. The acceptance criteria for the moderate frequency event are related to DNBR and overpressure of the RCS. The feedwater temperature reduction transient was analyzed at EPU conditions to evaluate RCS overpressure response and was performed in accordance with the methodology approved by the Nuclear Regulatory Commission (NRC) (Reference 1). The analysis confirmed that the peak RCS pressure and DNBR meet the acceptance criteria for this transient.

The reactivity addition due to the decrease in reactor coolant temperature is bounded by other moderate frequency events such as the uncontrolled rod withdrawal events described in Sections 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, and 2.8.5.4.2, Uncontrolled Control Rod Assembly Withdrawal at Power. Also, on a cycle-by-cycle basis Limiting Conditions of Operation (LCO) axial offset limits are developed that ensure preservation of DNB criteria throughout the cycle. The development of the offset limits includes the impact of temperature-induced neutron flux errors due to reactor vessel downcomer cooling. The limiting overcooling event, while not included in FSAR Chapter 14, is performed in a conservative manner and ensures that DNBR protection is provided each fuel cycle. As described in Section 2.8.2, Nuclear Design, the development of LCO axial offset limits will continue to be implemented for the EPU. The overcooling aspects of the feedwater temperature reduction transient are bounded by the limiting overcooling transient with respect to DNBR response.

Increase in Feedwater Flow

An increase in feedwater flow results in overcooling of the RCS that then leads to increasing core power. The RPS trip on overpower (high flux) prevents any power increase that could lead to a DNBR less than the safety analysis limit. Due to the limited nature of the reactivity excursion, the plant may evolve to a new steady-state condition without initiating an RPS trip.

The worst case of excess heat removal due to increased feedwater flow is the transient associated with the opening of feedwater control valves or bypass valves to greater than the normal operating position. In such an event, there is a sudden increase in feedwater flow to the steam generators. The increased cooling would create an increase in core power within the primary system that can potentially lead to a reactor trip. Normally, the ICS would modify the total feedwater flow demand signal to maintain a balance in the energy exchanged between the primary and secondary side of the steam generators.

The excessive heat removal event is a Condition II event (moderate frequency), as defined in Section 2.8.5.0, Non-LOCA Analysis Introduction. The acceptance criteria for the moderate frequency event are related to DNBR and overpressure of the RCS. The increase in feedwater flow transient was analyzed at EPU conditions to evaluate RCS overpressure response and was performed in accordance with the methodology approved by the (NRC) (Reference 1). The analysis confirmed that the peak RCS pressure and DNBR meet the acceptance criteria for this transient.

The reactivity addition due to the decrease in reactor coolant temperature is bounded by other moderate frequency events such as the uncontrolled rod withdrawal events described in Sections 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, and 2.8.5.4.2, Uncontrolled Control Rod Assembly Withdrawal at Power. Also, on a cycle-by-cycle basis LCO axial offset limits are developed that ensure preservation of DNB criteria throughout the cycle. The development of the offset limits includes the impact of temperature-induced neutron flux errors due to reactor vessel downcomer cooling. The limiting overcooling event, while not included in FSAR Chapter 14, is performed in a conservative manner and ensures that DNBR protection is provided each fuel cycle. As described in Section 2.8.2, Nuclear Design, the development of LCO axial offset limits will continue to be implemented for the EPU. The overcooling aspects of the increase in feedwater flow transient are bounded by the limiting overcooling transient with respect to DNBR response.

Increase in Steam Flow

The excessive load increase accident is defined as a sudden increase in secondary-side steam flow causing a mismatch between the reactor core power production and the steam generator heat demand. This accident could result from the inadvertent opening of a steam relief (including atmospheric dump valves) or turbine bypass valve by the operator or an equipment malfunction such as a pressure regulator failure. The Steam Conversion System adjusts to load increases within the limits of its automatic control operation. When load increases cannot be accommodated, the RPS will trip the reactor on low reactor coolant pressure or high neutron flux. Regardless of the rate of load increase, the RPS will trip the reactor in time to prevent the DNBR from going below the limit value. Increases in steam load to more than design flow are analyzed as the steam line rupture event in Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment.

An excessive load increase which cannot be accommodated by the automatic control operation of the Steam Conversion System is analogous to a steam line break. The inadvertent opening of a steam relief or turbine bypass valve by the operator or an equipment malfunction such as a pressure regulator failure will cause a sudden decrease in the secondary system pressure. The reduction in steam pressure is accompanied by an increase in the steam flow through the steam generator which decreases the RCS temperature and pressure. The Steam Conversion System will adjust to load increases within its automatic control operation. However, if the load increase is not within its automatic control operation, the RCS temperature and pressure will continue to decrease until the RPS trips the reactor on low reactor coolant pressure or high neutron flux.

The excessive heat removal event is a Condition II event (moderate frequency), as defined in Section 2.8.5.0, Non-LOCA Analysis Introduction. The reactivity addition due to the decrease in reactor coolant temperature is bounded by other moderate frequency events such as the uncontrolled rod withdrawal events described in Sections 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, and 2.8.5.4.2, Uncontrolled Control Rod Assembly Withdrawal at Power. Also, on a cycle-by-cycle basis LCO axial offset limits are developed that ensure preservation of DNB criteria throughout the cycle. The development of the offset limits includes the impact of temperature-induced neutron flux errors due to reactor vessel downcomer cooling. The limiting overcooling event, while not included in FSAR Chapter 14, is performed in a conservative manner and ensures that DNBR protection is provided each fuel cycle. As described in Section 2.8.2, Nuclear Design, the development of LCO axial offset limits will continue to be implemented for the EPU. The overcooling aspects of the increase in steam flow transient are bounded by the limiting overcooling event with respect to DNBR response.

<u>Results</u>

The excessive heat removal events described in this section are non-limiting events at CR-3. The CR-3 evaluation has shown that DNBR response of these excessive heat removal events to be bounded by other reactivity insertion events such as the rod withdrawal events and the limiting overcooling event. The decrease in feedwater temperature transient and the increase in feedwater flow transient were analyzed at EPU conditions, to evaluate RCS overpressure response. The analyses confirmed that the peak RCS pressure and DNBR meet the acceptance criteria for these transients.

2.8.5.1.1.3 Conclusion

CR-3 has reviewed the analyses of the decrease in feedwater temperature transient and the increase in feedwater flow transient described above at the EPU conditions and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using approved methodology. CR-3 also concludes that DNBR response of excess heat removal events described in this section are bounded by other analyzed events. CR-3 further concludes that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of these events. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, 1.4.14, 1.4.15, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the events stated.

2.8.5.1.1.4 References

1. AREVA NP Inc. Document 43-10193PA-00, “RELAP5/MOD2-B&W for Safety Analysis of B&W Designed Pressurized Water Reactors,” January 2000.

2.8.5.1.2 Steam System Piping Failures Inside and Outside Containment

2.8.5.1.2.1 Regulatory Evaluation

The steam release resulting from a rupture of a main steam pipe will result in an increase in steam flow, a reduction of coolant temperature and pressure, and an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. Reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered (1) postulated initial core and reactor conditions; (2) methods of thermal and hydraulic analyses; (3) the sequence of events; (4) assumed responses of the reactor coolant and auxiliary systems; (5) functional and operational characteristics of the reactor protection system; (6) operator actions; (7) core power excursion due to power demand created by excessive steam flow; (8) variables influencing neutronics; and (9) the results of the transient analyses.

The NRC's acceptance criteria for Steam System Piping Failures Inside and Outside Containment are based on:

- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;

- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core;

- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; and

- GDC-35, insofar as it requires that the Reactor Coolant System (RCS) and associated auxiliaries be designed to provide abundant emergency core cooling.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.28, Reactivity Hot Shutdown Capability, FSAR Section 1.4.29, Reactivity Shutdown Capability, and FSAR Section 1.4.30, Reactivity Holddown Capability, insofar as these criteria require that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained [GDC-27];

- FSAR Section 1.4.32, Maximum Worth of Control Rods - insofar as these criteria require that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core [GDC-28];

- FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention, and FSAR Section 1.4.35 Reactor Coolant Pressure Boundary Brittle Fracture Prevention, insofar as these criteria require that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized; [GDC 31] and;

- FSAR Section 1.4.37, Engineered Safety Features Basis for Design, FSAR Section 1.4.41, Engineered Safety Features Performance Capability, FSAR Section 1.4.42, Engineered Safety Features Components Capability, and FSAR Section 1.4.44, Emergency Core Cooling Systems Capability, insofar as these criteria require that the Reactor Coolant System (RCS) and associated auxiliaries be designed to provide abundant emergency core cooling. [GDC-35];

2.8.5.1.2.2 Technical Evaluation

<u>Introduction</u>

The major rupture of a main steam pipe is the most-limiting post-trip cooldown transient. The steam release from a major rupture of a main steam pipe will result in an initial increase in steam flow that decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient (MTC), the cooldown results in a positive reactivity insertion and subsequent reduction in core shutdown margin. A return to critical following a steam pipe rupture is a concern primarily because of the high-power peaking factors that could exist assuming the most-reactive control rod assembly (CRA) is stuck in its fully withdrawn position. Isolation of the secondary side leads to a reduction in the once through steam generator (OTSG) inventory that eventually terminates the overcooling and ends the positive reactivity insertion. Shutdown margin increases again as the RCS temperatures begin to rise and as boric acid is injected by the ECCS. The most limiting initial condition for the transient is at full power, since the initial steam generator inventory is maximized at full power resulting in the largest overcooling potential. Further, the break location inside containment, close to the steam generator is most limiting because the steam generator-side break path is shorter, which minimizes line losses and maximizes break flow.

The primary design features which provide protection for steam pipe ruptures are the Engineered Safeguards Actuation System (ESAS), described in FSAR section 7.1.3, and the Reactor Protection

System (RPS), described in FSAR section 7.1.2. The Emergency Feedwater Initiation and Control (EFIC) system (FSAR section 7.2.4) initiates emergency feedwater from low pressure in either OTSG, however, the Feed Only Good Generator (FOGG) logic prevents emergency feedwater from being provided to the faulted OTSG. The EFIC system also initiates steam line isolation.

Several plant modifications are being implemented in conjunction with the EPU, some of which have an impact on the main steam line break (MSLB) (refer to Appendix E, Major Plant Modifications). Section 2.5.5.4, Condensate and Feedwater, describes the Condensate and Feedwater Systems. Modifications to these systems include modification of Main Feedwater (MFW) and condensate pumps to provide the increased MFW flow required to remove the additional power associated with the EPU. The MFW pumps and condensate pumps are modeled explicitly within the thermal-hydraulic code used to analyze the MSLB. Therefore, the increased initial flow, as well as the transient behavior of the system is accounted for. The reconfiguration of the MFW components may result in increased MFW runout flow. Another MFW System change described in Section 2.5.5.4, Condensate and Feedwater, impacts the isolation time for MFW. New MFW pump suction isolation valves will be installed that are capable of achieving faster closure times. Thus, the safety analysis assumption relative to MFW isolation time is changed versus the historical assumptions.

Another consideration for the EPU is the inventory associated with the OTSG. The OTSGs were replaced in a prior cycle, however, due to the increase in power level, the inventory in the OTSG increases. The detailed thermal-hydraulic model of the OTSG captures the increase in initial mass inventory. The increase in inventory due to the EPU contributes to more limiting results; however, the decrease in isolation time offsets the negative impact.

The minimum required shutdown margin has been increased for the EPU, primarily due to the MSLB post-trip analysis results. The increased minimum shutdown margin was required in order to ensure that the reactor remains subcritical during the post-trip phase of the MSLB.

Plant modifications are planned to ensure that a single high pressure injection (HPI) train provides adequate flow to meet 10 CFR 50.46 criteria. The resulting increase in HPI flow has an impact on MSLB, since the relatively cold water injected by HPI exacerbates the overcooling of the RCS post-MSLB, especially prior to flushing of the system and delivery of borated water. The increase in HPI flow was considered as an input to the core response analyses described herein.

Finally, end-of-cycle moderator temperature coefficients are more negative for the EPU. Likewise, analysis inputs for MTC are assumed to be more negative, resulting in greater post-trip power response. The impact of this difference is offset with increased minimum shutdown margin, and reduced time for MFW isolation.

The evaluation of a steam line rupture is described in FSAR section 14.2.2.1. The MSLB mass and energy (M&E) analysis is provided in section 2.6.3.2, Mass and Energy Release Analysis for Secondary System Pipe Ruptures. The containment response is provided in section 2.6.1, Primary Containment Functional Design.

Description of Analyses and Evaluations

A detailed analysis using the RELAP5/MOD2-B&W computer code (Reference 1) was performed in order to determine the plant transient conditions following a main steam line break. The computer code uses a control volume approach to solve the time-dependent conservation equations for mass, energy and momentum over the steam and liquid phases. The Feedwater System model included the feedwater piping, pumps, valves, and feedwater heaters from the suction of the feedwater booster pumps to the inlet to the OTSGs. This allowed modeling of EFIC-initiated functions, including closure of various valves and trips of feedwater pumps, while also providing time-dependent solutions to complex phenomena such as transport of liquid into the OTSG due to flashing of liquid in the feedwater piping.

The analysis used the methodology defined in BAW-10193 (Reference 2). The RELAP5/MOD2-B&W code is used in conjunction with the methodology, which also incorporates conservative setpoints (consistent with current Improved Technical Specifications (ITS) Allowable Values) and capacities to arrive at a conservative result. The steam line break analysis takes both the pre-trip and post-trip response into consideration. The input assumptions for the pre-trip and post-trip evaluations differ in some cases, and are chosen to provide the most conservative response for the scenario in question. For example, the post-trip response evaluation models reactor trip at event initiation to maximize the overcooling. The pre-trip response evaluation uses conservative trip setpoints to evaluate the maximum power reached, conservatively delaying the time to reactor trip. Additional details regarding the input assumptions for both the pre-trip and post-trip evaluations are provided below.

The key input parameters and initial conditions used in the analysis of the main steam line rupture event are as follows:

- The core power level assumed for the post-trip reactivity response is 2966.8 MWt, which corresponds to the nominal power level when the Leading Edge Flow Meters (LEFM) are not available (ITS 3.3.1). The choice of power level is conservative for the post-trip response since the decay levels post-trip are reduced. The evaluation of the pre-trip power response assumes 3026.1 MWt (nominal EPU power level plus heat balance uncertainty).

- The initial OTSG mass inventory is maximized, reflecting full power operation at the EPU power level. Further, the downcomer level is assumed to be at the maximum allowed value. Due to the increase in power level, the initial OTSG mass inventory for the EPU analysis is increased as described above (see Introduction).

- The OTSG tube plugging level was assumed to be 0% to maximize the primary to secondary heat transfer.

- The evaluation of post-trip reactivity response modeled 0.9 times the ANS 1971 decay heat standard with no actinides, per Reference 2. The evaluation of the pre-trip power response modeled 1.0 times the ANS 1971 decay heat standard with actinides.

- Initial Reactor Coolant System flow is set to the minimum thermal design flow and complete thermal mixing in the downcomer and the reactor vessel lower plenum is assumed, consistent with Reference 2. The minimum thermal design flow (374,880 gpm) is increased for the EPU fuel cycle, taking credit for the presence of clean OTSGs.

- The initial RCS average temperature is 582°F, which reflects the increase in RCS temperature associated with the EPU.

- The initial RCS pressure is assumed to be 2170 psia at the hot leg tap. The pressurizer level was assumed to be 220 inches indicated. These parameters were set to their nominal values consistent with Reference 2, and are unchanged for the EPU.

- For the evaluation of the post-trip reactivity response at the EPU conditions, the reactor was tripped at the start of the event to minimize core heat addition to the reactor coolant. The evaluation of the pre-trip power response at the EPU conditions credited RPS trip functions (high neutron flux, low reactor coolant pressure). The high flux trip setpoint considered the effects of increased neutron attenuation due to colder water in the reactor vessel downcomer.

- The analysis of the main steam line break at the EPU conditions required an increase in the shutdown margin requirement from 1.0%Δk/k to 1.3%Δk/k, as described above (see Introduction). The control rod worth inserted on reactor trip is the minimum required to achieve a shutdown margin of 1.3%Δk/k at hot, zero power conditions with the highest worth control rod stuck out of the core.

- End-of-cycle reactivity parameters at the EPU conditions were assumed for the main steam line break analyses. The reactivity parameters for the pre- and post-trip evaluations were chosen to maximize the power response. The End of Cycle (EOC) MTC is increased for the EPU, as described above (see Introduction). Reactivity versus density curves were developed to model moderator feedback, based on -4.6×10^{-4} Δk/k/°F, which bounds the expected EOC conditions for the EPU.

- High pressure injection (HPI) was actuated upon event initiation for the post-trip response at the EPU conditions in order to maximize the overcooling of the RCS. An appropriate delay time to account for flushing of deborated injection fluid was assumed prior to crediting boron. The initiation of HPI does not impact the pre-trip evaluation since reactor trip occurs prior to initiation of engineered safeguards.

- Main feedwater isolation is initiated upon receipt of the EFIC low steam line pressure signal after an appropriate delay for signal processing. Once initiated, the analysis assumes that isolation of main feedwater flow occurs over a 20 second valve closure time. The valve closure time is less than assumed pre-EPU. The faster closure time is appropriate due to the installation, in conjunction with the EPU, of new MFW isolation valves with improved performance (refer to Appendix E).

- The temperature of the HPI fluid is assumed to be 40°F for the post-trip response, which represents the lowest allowable BWST temperature per the ITS.

- Offsite power is assumed to be available to maximize the heat removal via the OTSGs.

- The most limiting single failure, failure of the main feedwater pump on the affected OTSG to trip following receipt of a safety grade EFIC signal, is considered. This is the limiting failure because it maximizes the feedwater to the affected OTSG, causing the maximum cooling of the primary system and the greatest mass addition to the reactor building, as compared with other postulated single failures. The analyses performed for the EPU demonstrated that the limiting single failure continues to be the failure to trip the MFW pump on the affected loop, even when considering the faster isolation times for MFW flow.

- No operator actions are credited for the MSLB analyses.

The specific FSAR (Section 14.2.2.1.2) acceptance criteria applied by CR-3 for the MSLB event are as follows:

- The core shall remain intact for effective core cooling. This is met in the analysis by demonstrating that the reactor thermal power remains less than 112% of rated power.
- The reactor shall not return to a high power level due to a return to criticality following reactor trip. This is conservatively met by ensuring that the core does not return to critical following reactor trip. Additionally, a DNB analysis is performed that demonstrates the minimum DNBR ratio remains acceptable post-trip.
- The Reactor Coolant System pressure shall not exceed 2750 psig (110% of RCS design pressure).
- The accident doses shall be within 10 CFR 50.67 limits.
- The OTSG tubes shall not fail due to the loss of secondary side pressure and resultant temperature gradients.
- The reactor building pressure during a steam line rupture inside containment shall not exceed the reactor building design limit.

The discussion in the Results section demonstrates that all applicable acceptance criteria are met for the main steam line rupture event by CR-3 at the EPU conditions.

<u>Results</u>

For CR-3, the original licensing basis identifies the most limiting main steamline rupture case as a double-ended rupture of a main steam line inside containment with a failure of the main feedwater pump on the affected OTSG to trip following receipt of a safety grade EFIC signal. The evaluation of the pre-trip power response determined that the peak thermal power did not exceed 112 %. The calculated sequence of events for the most limiting case for post-trip core responses at the EPU conditions are shown in Table 2.8.5.1.2-1. Figures 2.8.5.1.2-1 through 2.8.5.1.2-4 illustrate the transient results for the most limiting CR-3 case at the EPU conditions with regard to the post-trip core response.

The break causes the secondary system to depressurize, resulting in a low steam line pressure EFIC actuation on the affected loop. This is followed by a low steam line pressure EFIC actuation on the unaffected loop. EFIC isolates the feedwater and steam systems from the OTSGs. However, the EFIC trip of the main feedwater pump on the affected loop is assumed to fail. As a result, feedwater continues to enter the affected OTSG prior to closure of the main feedwater pump suction valve. Following OTSG isolation, the affected OTSG continues to blow down until it is dry and depressurized.

The reactor trip and ESAS actuation of HPI are assumed coincident with break initiation to maximize the overcooling. Control rods insert to shut down the reactor. After 30 seconds, the unborated water in the system is flushed and boron from the HPI system begins to reach the cold legs. The primary system continues to cool and depressurize until the affected OTSG dries out and is depressurized. The cooldown causes a reactivity insertion and a reduction in core shutdown margin. The k_{EFF} approaches

1.0, but the core remains subcritical. The peak reactor power due to subcritical multiplication is 27 percent of 3014 MWt. Once the affected OTSG dries out and depressurizes, the core shutdown margin increases, the subcritical multiplication decreases, and the event is terminated.

The specific acceptance criteria listed are reviewed below:

- At no time did the core power exceed 112 % prior to reactor trip. Therefore, fuel pins will not experience DNB in the pre-trip core configuration since LCO offset limits provide protection for thermal power levels up to 112% of the EPU rated power level.
- Although the reactor remains subcritical following reactor trip, a DNB analysis of the limiting post-trip point (i.e., highest post-trip power) in the transient was performed at 27% of the EPU rated power level as described in Section 2.8.3, Thermal and Hydraulic Design. The results of the analysis showed that the DNBR remained above the limit.
- The steam line break accident results in a decrease in primary and secondary system pressures. Following dryout and depressurization of the affected OTSG, the primary system and unaffected OTSGs would repressurize to normal post-trip conditions. Consequently, system pressure limits are not challenged during a SLB accident.
- The evaluation of the dose consequences associated with a main steam line rupture is discussed in Section 2.9.2, Radiological Consequences Analyses.
- The structural integrity of the OTSG has been evaluated for the EPU conditions, as described in Section 2.2.2.5, Steam Generators and Supports. The results of the evaluation demonstrated that the steam generator pressure boundary and internal components continue to comply with the structural criteria of the ASME Code, Section III, Class 1, Subsection NB and NF.
- The reactor building pressure during a steam line rupture inside containment is evaluated in Section 2.6.1, Primary Containment Functional Design.

In addition to the FSAR acceptance criteria listed above, GDC-31 requires demonstration that the Reactor Coolant System behave in a nonbrittle manner and the probability of a rapidly propagating fracture is minimized for MSLB. Operating pressure-temperature limits are established to provide protection against brittle fracture. The RCS conditions predicted during the MSLB are well within the acceptable pressure-temperature limits of operation for the EPU. Therefore, brittle fracture concerns are precluded.

2.8.5.1.2.3 Conclusion

CR-3 has reviewed the analyses of the steam system piping failure events described above and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the reactor coolant pressure boundary pressure limits will not be exceeded, the reactor coolant pressure boundary will behave in a non-brittle manner, the probability of propagating fracture of the reactor coolant pressure boundary is minimized, and adequate core cooling will be provided. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.28, 1.4.29, 1.4.30, 1.4.32, 1.4.34, 1.4.35, 1.4.37, 1.4.41, 1.4.42, and 1.4.44 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Steam System Piping Failures Inside and Outside of Containment.

2.8.5.1.2.4 References

1. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary) and BAW-10164NP-A, Revision 6, June 2007 (Nonproprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

2. BAW-10193-P-A (Proprietary), RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors, Parece, M. V., January 2000.

Table 2.8.5.1.2-1

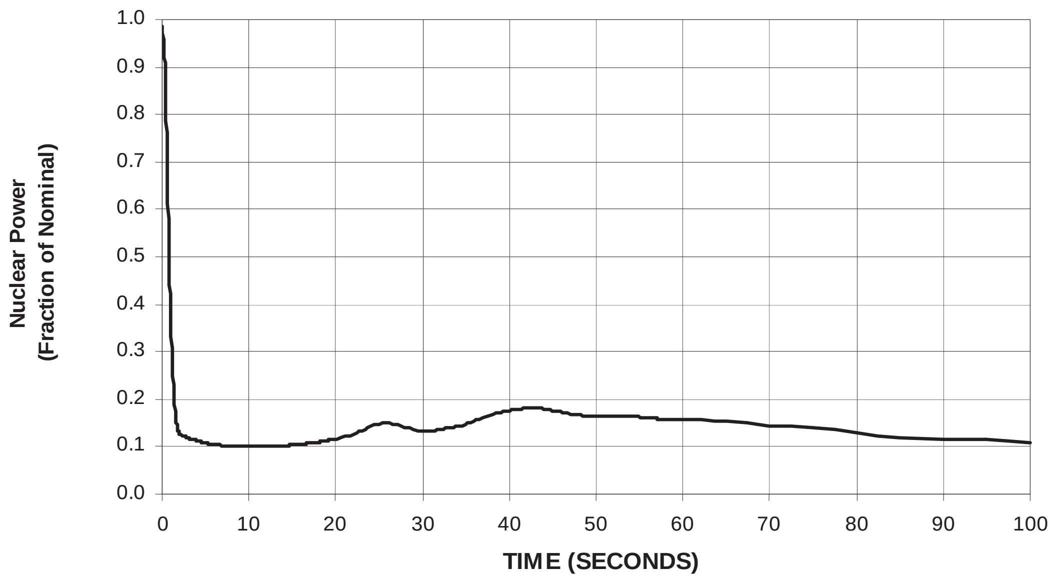
CR-3 Sequence of Events for Double-Ended Rupture of One Steam Line Inside Containment at EPU Conditions

(Failure of Main Feedwater Pump to Trip on Affected Loop)

Event	Time (seconds)
Break Initiated, Reactor Trip, and High pressure injection from two pumps begins.	0.0
Low steam line pressure EFIC setpoint reached in the affected loop (begins steam and feedwater isolation).	0.2
Main feedwater pump on the affected loop fails to trip (single failure)	N/A
Steam line isolation on the affected loop.	6.2
Low steam line pressure EFIC setpoint reached in the unaffected loop.	15.1
Main feedwater pump on the unaffected loop trips.	16.1
Steam line isolation on the unaffected loop.	21.1
Main feedwater pump suction valves are closed on the affected loop, isolating the feedwater line on the affected loop.	21.2
Boron from the High Pressure Injection reaches the cold legs.	30.0
Main feedwater isolation and startup block valves are closed on the affected loop.	31.2
Main feedwater pump suction valves are closed on the unaffected loop.	36.1
Maximum power due to subcritical multiplication is reached.	42.0
Main feedwater isolation, main feedwater pump suction, and startup block valves are closed on the unaffected loop.	46.1
Low load block valves are closed on the affected loop.	67.2
Low load block valves are closed on the unaffected loop.	82.1
Event terminated	100.0

Figure 2.8.5.1.2-1
CR-3 Double-Ended Rupture of One Steam Line Inside Containment at EPU Conditions

(Failure of Main Feedwater Pump to Trip on Affected Loop)
Nuclear Power and Core Heat Flux vs. Time



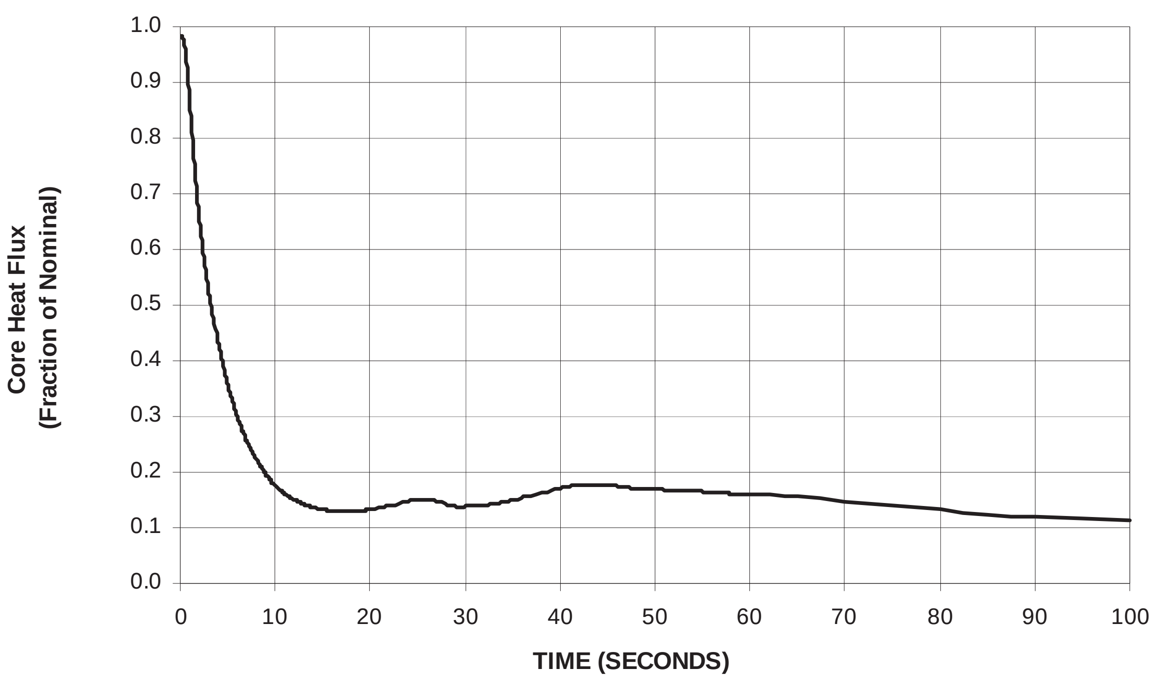
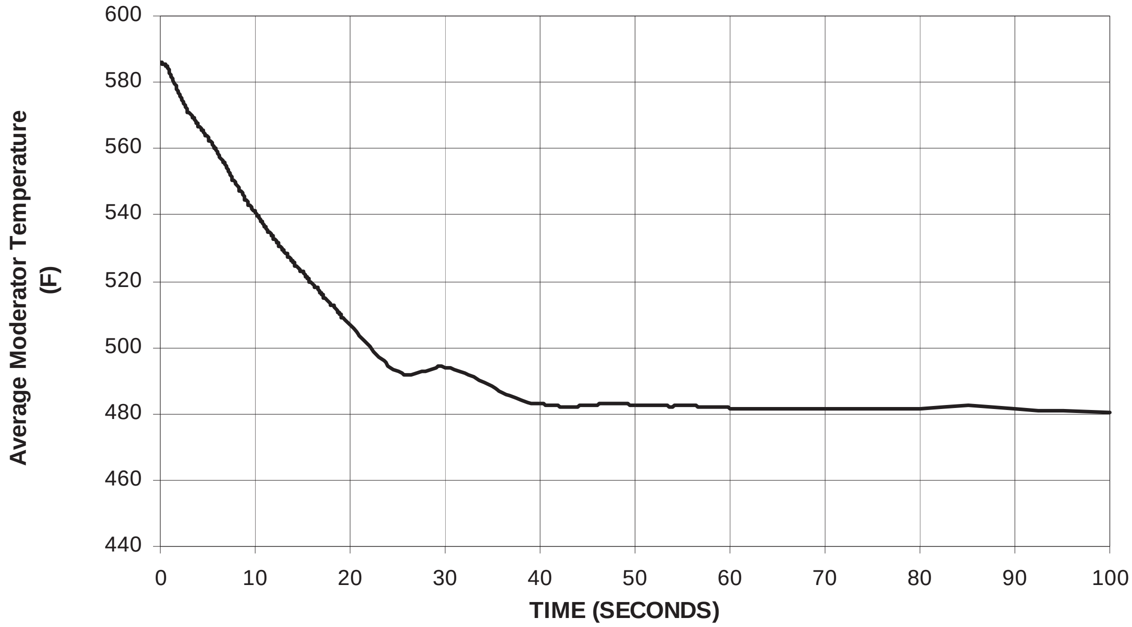


Figure 2.8.5.1.2-2
CR-3 Double-Ended Rupture of One Steam Line Inside Containment at EPU Conditions

(Failure of Main Feedwater Pump to Trip on Affected Loop)
Average Moderator Temperature and Cold Leg Fluid Temperature vs. Time



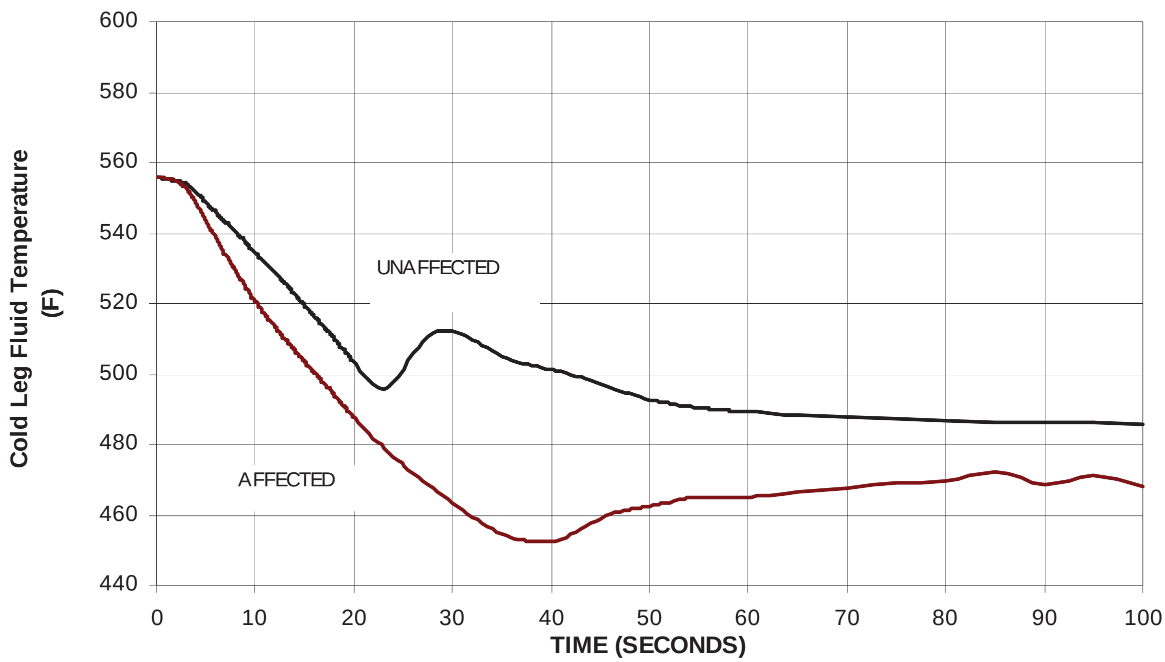
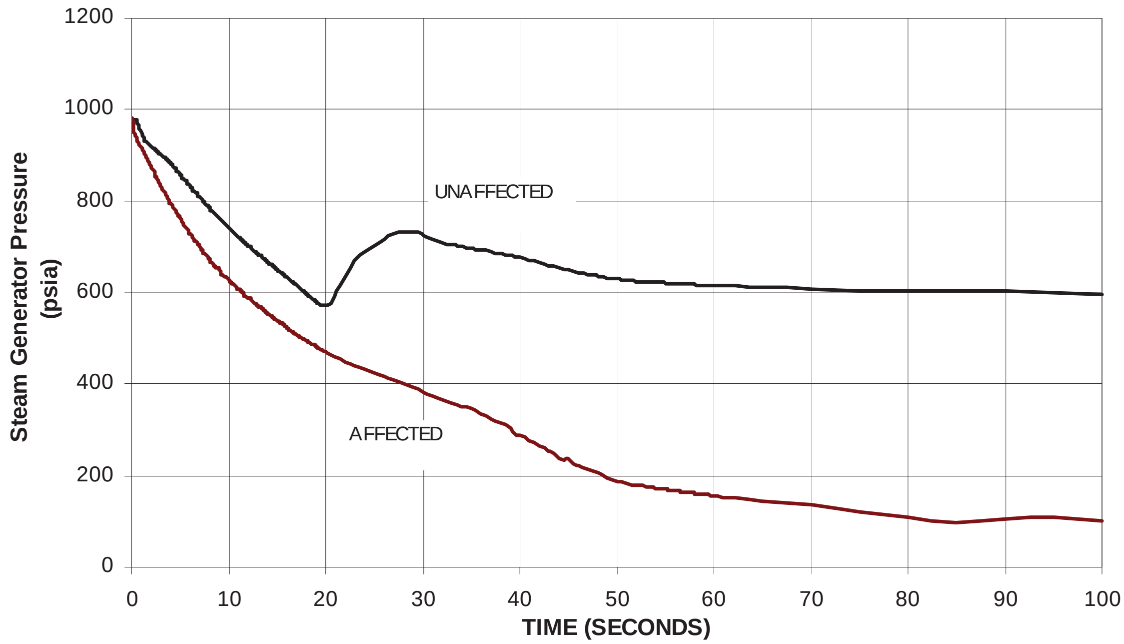


Figure 2.8.5.1.2-3
CR-3 Double-Ended Rupture of One Steam Line Inside Containment at EPU Conditions

(Failure of Main Feedwater Pump to Trip on Affected Loop)
Core Exit Pressure and OTSG Pressure vs. Time



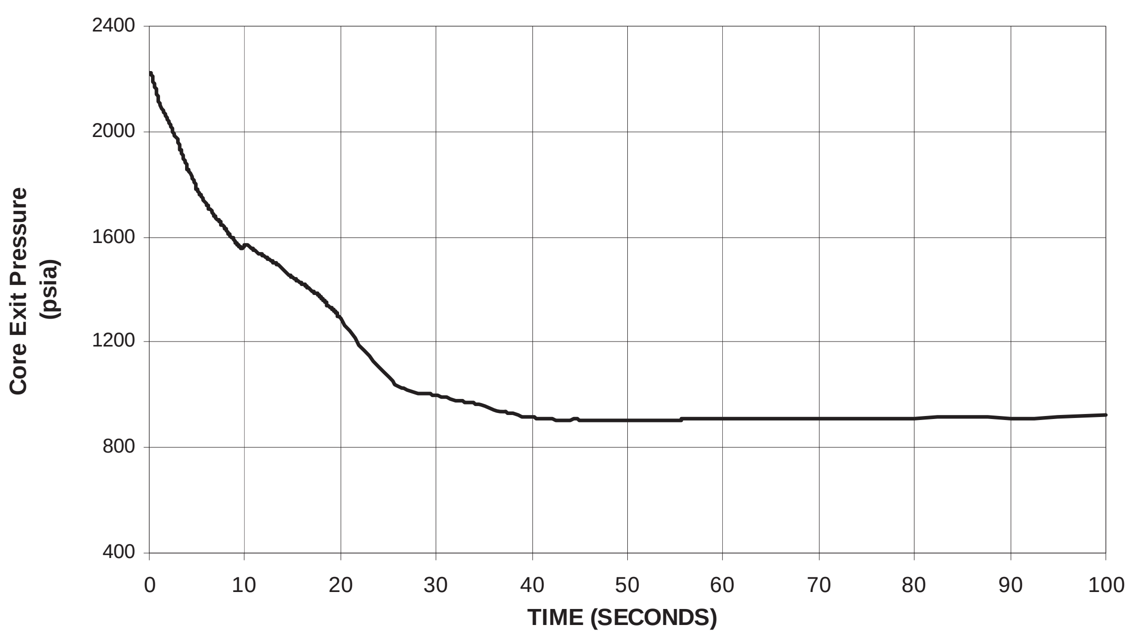
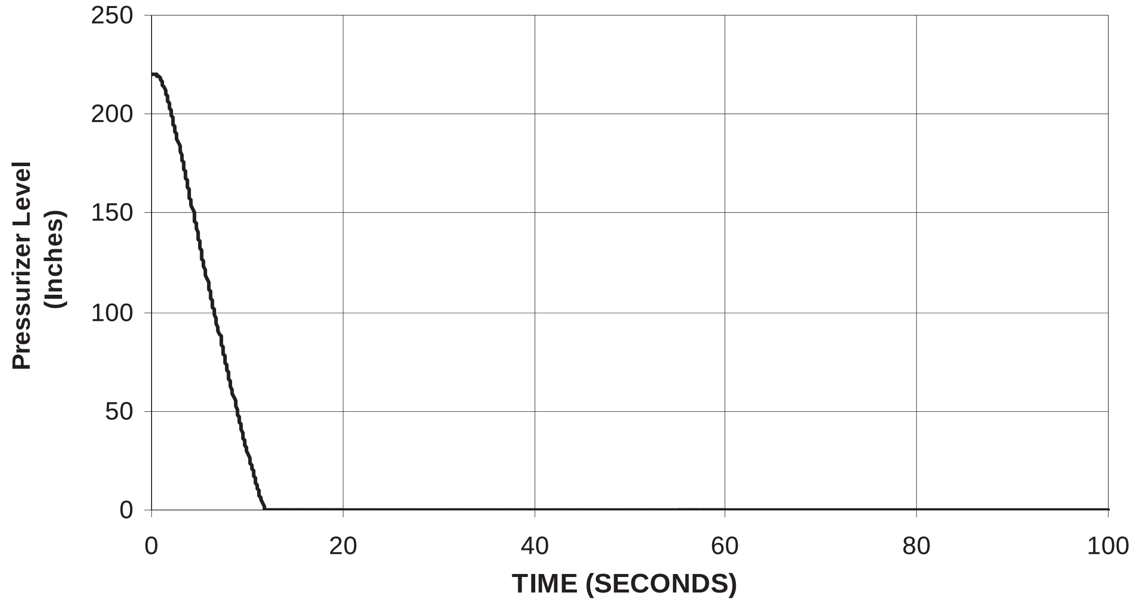
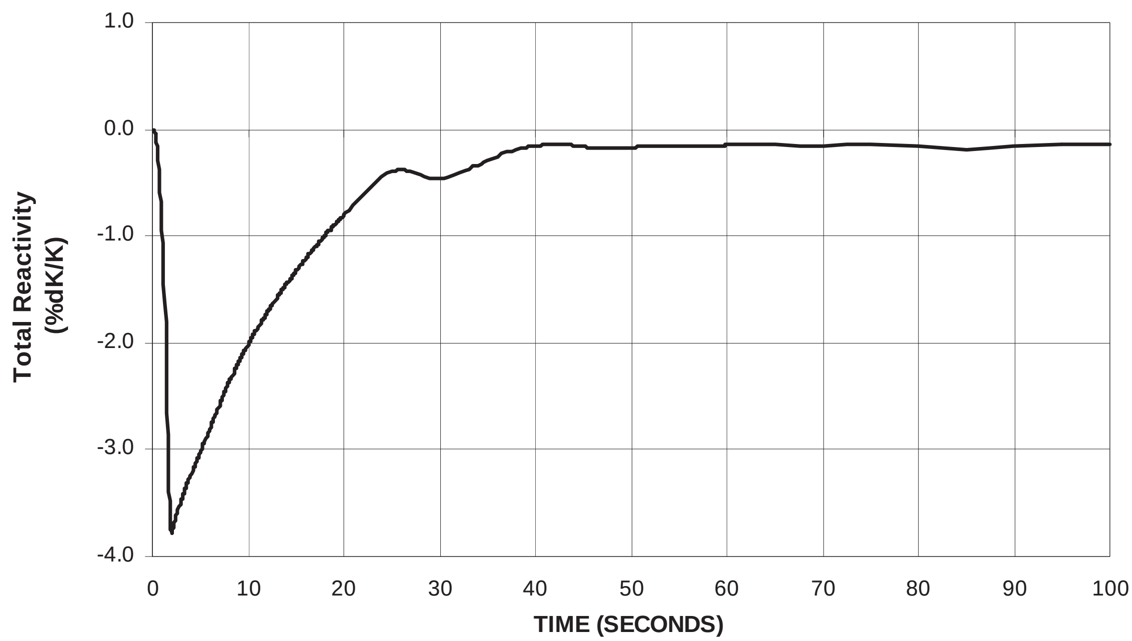


Figure 2.8.5.1.2-4
CR-3 Double-Ended Rupture of One Steam Line Inside Containment at EPU Conditions

(Failure of Main Feedwater Pump to Trip on Affected Loop)
Pressurizer Level and Total Reactivity vs. Time





2.8.5.2 Decrease in Heat Removal by the Secondary System

2.8.5.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure

2.8.5.2.1.1 Regulatory Evaluation

A number of initiating events may result in unplanned decreases in heat removal by the secondary system. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses.

The NRC's acceptance criteria are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operations occurrences (AOOs);
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operations;
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs. [GDC-10]
- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design of the RCPB are not exceeded during any condition of normal operations. [GDC-15];
- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity

changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded. [GDC-26].

2.8.5.2.1.2 Technical Evaluation

Introduction

As described in FSAR Section 14.1.2.8, the load rejection accident is a severance of or electrical disconnection from the unit transmission lines. The rejection results in a decrease in secondary heat removal and an increase in the secondary side steam pressure. A turbine trip from a power level > 45% will actuate the anticipatory reactor trip (ART). However, no credit is taken for anticipatory reactor trip on a turbine trip. A load rejection or turbine trip from full power will result in a reactor trip on RCS high pressure, which is assumed in the analysis of this event. The primary and secondary system pressure responses for a turbine trip event are bounding in comparison to loss of load, loss of condenser vacuum, and steam pressure regulator failure. Therefore, only the turbine trip event was explicitly analyzed for the EPU. The turbine trip event is limiting for secondary overpressurization, and therefore the objective of the analysis is to maximize secondary response. Other events, such as the Startup Accident (Section 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical pr Low Power Startup Condition) for overpressure, or the Loss of Coolant Flow (Section 2.8.5.3.1, Loss of Forced Reactor Coolant Flow) for Departure from Nucleate Boiling Ratio (DNBR), are more limiting for primary system effects.

The turbine trip event results in the fast closure of the turbine governor valves, causing the most rapid termination of steam flow from the steam generators, and for this reason the turbine trip is the most limiting of the loss of load events covered in this section. For a load rejection there is some possibility of an increase in RCS flow due to overspeed of the reactor coolant pumps. However, any overcooling and subsequent power response would be minimal such that events resulting in loss of reactor coolant flow would remain limiting for DNBR. The loss of steam flow produces a rapid increase in the secondary steam pressure and a decrease in secondary heat transfer. The secondary side pressure increase results in the opening of the main steam safety valves (MSSVs), which limits the peak secondary pressure, re-establishes steam flow, and consequently decreases the primary to secondary heat removal mismatch. The increase in the pressure on the primary side causes the high RCS pressure reactor trip setpoint to be reached. Following the reactor trip, the pressurizer safety valves (PSVs) may open if necessary to limit the peak pressure on the primary side. The reactor trip and opening of the PSVs further reduces the primary to secondary heat transfer mismatch until the heat removal is sufficient to reduce the RCS temperature. At this point, the turbine trip event is terminated.

Description of Analyses and Evaluations

The load rejection (turbine trip) analyses used the methodology defined in BAW-10193 (Reference 1). The RELAP5/MOD2-B&W code (Reference 2) was used in conjunction with the methodology, which also incorporated conservative setpoints and capacities to arrive at a conservative result. The key input parameters and initial conditions used in the analysis of the turbine trip accident are as follows:

- The initial core power level was set to 3026.1 MWt (100.4% of 3014 MWt), which is equivalent to the planned power level for the EPU with an allowance for heat balance uncertainty. Additionally, a conservative reactor coolant pump (RCP) heat of 16.4 MWt was also modeled.

- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent with the increase in T_{AVG} planned in conjunction with the EPU. The hot leg pressure was assumed to be 2170 psia.
- The RCPs were assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the minimum RCS flow rate (374,880 gpm).
- The initial pressurizer level was modeled as nominal minus uncertainty (200 inches) to maximize the secondary peak pressure.
- Two PSVs were modeled with a nominal lift setpoint of 2514.7 psia, plus 3% lift tolerance, and 0% accumulation. A blowdown of 4% was also used. The PSV capacity was modeled as 317,973 lbm/hr/valve at 2764.7 psia.
- Pressurizer spray was modeled with a design flow of 190 gpm. Pressurizer heaters were not modeled.
- The pressurizer power operated relief valve (PORV) was not credited.
- The main feedwater flow was linearly ramped down to zero flow over 3 seconds following reactor trip.
- Reactor trip was modeled to occur on a nominal high RCS pressure setpoint plus uncertainty (2400 psia).
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- The tripped rod worth assumed for the analysis is based on a minimum shutdown margin of 1.0 %Δk/k. This is less than the minimum Modes 1 and 2 shutdown margin required for the EPU (i.e., 1.3 %Δk/k as detailed in a separate attachment associated with Improved Technical Specifications (ITS) changes).
- A Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) and moderator coefficient (0.0 Δk/k/°F), typical of beginning-of-cycle conditions, were used since they yield the maximum rate of power increase.
- The MSSVs were modeled to lift at the nominal setpoints plus 3% lift tolerance and 3% accumulation. A nominal blowdown value of 5% was used.
- The actions of Emergency Feedwater Initiation and Control (EFIC), including Emergency Feedwater EFW flow, are not credited since the peak pressures occur prior to EFW delivery to the once through steam generators (OTSGs).
- OTSG tube plugging of 0% was modeled.
- Offsite power was available, which is consistent with the plant licensing basis.

- No single failures were considered since there is no single failure that would produce a more limiting event consequence.
- No operator actions were considered since peak pressures occur before any actions are postulated.
- No beneficial integrated control system (ICS) actions were credited (i.e., power runback).
- No credit is taken for anticipatory reactor trip on turbine trip.
- No credit is taken for the turbine bypass valves or the atmospheric dump valves.

The specific FSAR acceptance criteria applied by CR-3 for the load rejection event were:

- Fuel damage shall not occur from an excessive power-to-flow ratio.
- The reactor coolant system pressure shall not exceed code pressure limits. This criterion is met by ensuring that the peak RCS pressure remains below 110% of the design pressure of the RCS. With a RCS design pressure of 2514.7 psia, the peak pressure shall remain below 2764.7 psia.
- The accident doses shall be within 10 CFR 50.67 limits.
- The OTSG and steam line piping shall not exceed code pressure limits. This criterion is met by ensuring that the steam line pressure remains below 110% of the steam line design pressure of 1064.7 psia (1169.7 psia) and that the OTSG pressure remains below 110% of the OTSG design pressure of 1164.7 psia (1279.7 psia).

With the current plant configuration, the early reactor trip during the turbine trip event coupled with full RCS flow precludes fuel damage from an excessive power-to-flow ratio. Further, plant operating limits are set each refueling cycle to provide steady-state DNBR protection for power levels up to 112%, assuming minimum RCS flow rates and normal operating pressure and temperature. Therefore, the DNBR is not calculated for the turbine trip event. The reactor trip also reduces the amount of steam relieved to the atmosphere and ensures that the dose consequences of the turbine trip event are bounded by those generated by the steam line break event (Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment). Consequently, with the current plant configuration, only the acceptance criteria related to pressure are postulated to be challenged during the limiting load rejection event (turbine trip). Specifically, the limiting load rejection event was analyzed to challenge the secondary side overpressure acceptance criteria. The RCS pressure response for the load rejection event (turbine trip) is bounded by the Startup Accident (Section 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition), and therefore the inputs for the load rejection (turbine trip) analysis were biased to maximize the secondary side pressure response.

Results

The sequence of events for the turbine trip accident is listed in Table 2.8.5.2.1-1 and the calculated results are tabulated in Table 2.8.5.2.1-2. Figures 2.8.5.2.1-1 through 2.8.5.2.1-5 show transient plots of the significant plant parameters following a turbine trip accident.

The RCS pressure peak was reached at 5 seconds and remained below the limit of 2764.7 psia. The peak steam line pressure was reached at 8 seconds and the peak OTSG pressure was reached at 8.5 seconds. The steam line maximum pressure was less than the corresponding limit of 1169.7 psia and the OTSG peak pressure was below the limit of 1279.7 psia.

The acceptance criteria pertaining to fuel damage and dose consequences are not challenged for this accident. This is confirmed as the power remained at or below the initial value of 3026.1 MWt, forced flow and subcooling are maintained, and RCS pressure remains within limits. The remaining acceptance criteria pertain to the RCS and secondary side (OTSG and steam line) pressures remaining below code pressure limits. From the results provided in Table 2.8.5.2.1-2, there is ample margin (i.e., more than 100 psi) between the RCS and OTSG peak pressures and their respective limits. The peak steam line pressure maintained a minimum margin of 16.7 psi. Based on these results, it can be concluded that, for a turbine trip with reactor power at the EPU conditions, the MSSVs in conjunction with the high RCS pressure reactor trip provide sufficient overpressure protection to maintain OTSG and steam line pressures below the ASME code pressure limits. Therefore, the design basis limits for fission product barriers are not exceeded or altered.

2.8.5.2.1.3 Conclusion

CR-3 has reviewed the analyses of the decrease in heat removal events described above and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that it has been demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB limits will not be exceeded as a result of these events. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the turbine trip event.

2.8.5.2.1.4 References

1. BAW -10193PA-00, “RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors”.

2. BAW-10164PA-06, “RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis”.

Table 2.8.5.2.1-1: Sequence of Events for Full Power Turbine Trip

Event	Time (sec)
Transient begins TSVs close	0
MSSVs begin to lift	1.5
RCS hot leg pressure reaches the RPS setpoint of 2400 psia	2.8
Maximum RCS pressure is reached	5.0
MFW flow reaches 0	6.5
Maximum Steam Line pressure is reached	8.0
Maximum Steam Generator pressure is reached	8.5

Table 2.8.5.2.1-2: Results for Full Power Turbine Trip

Parameter	Analysis Result	Acceptance Criteria
Maximum RCS pressure (psia)	2569.83	2764.7
Maximum steam line pressure (psia)	1152.99	1169.7
Maximum OTSG pressure (psia)	1166.58	1279.7

Figure 2.8.5.2.1-1: Turbine Trip Core Power versus Time

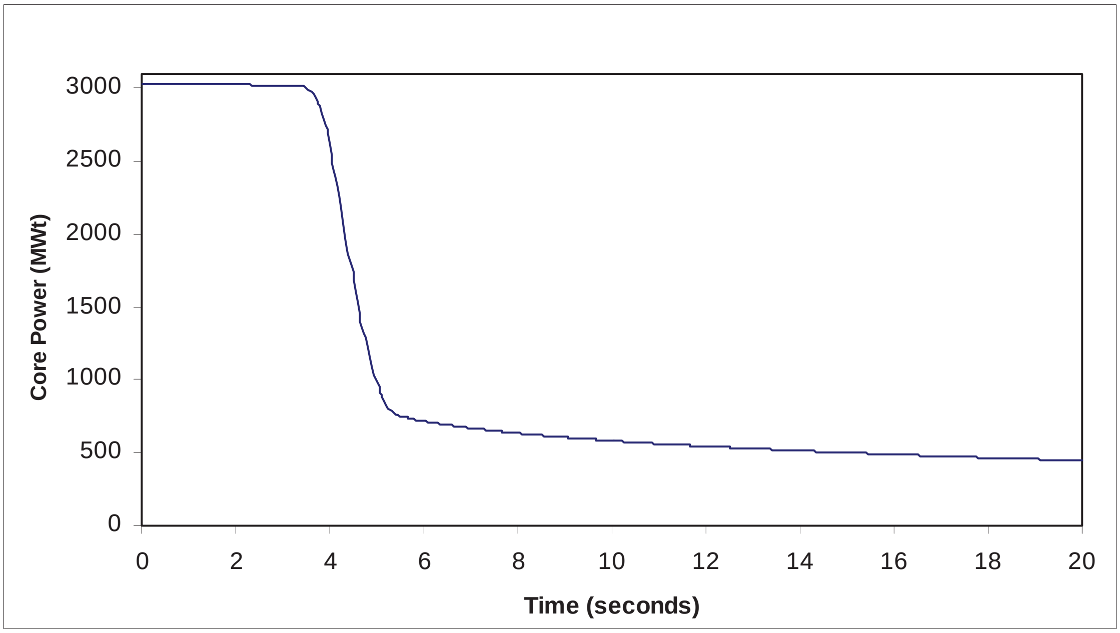


Figure 2.8.5.2.1-2: Turbine Trip RCS Pressure versus Time

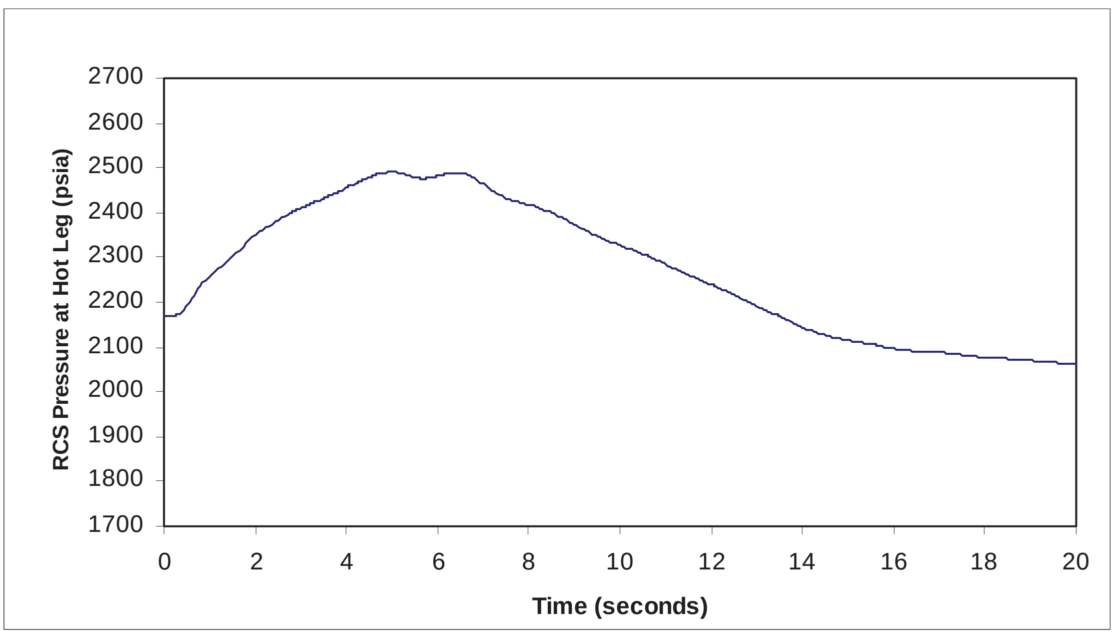


Figure 2.8.5.2.1-3: Turbine Trip RCS Average Temperature versus Time

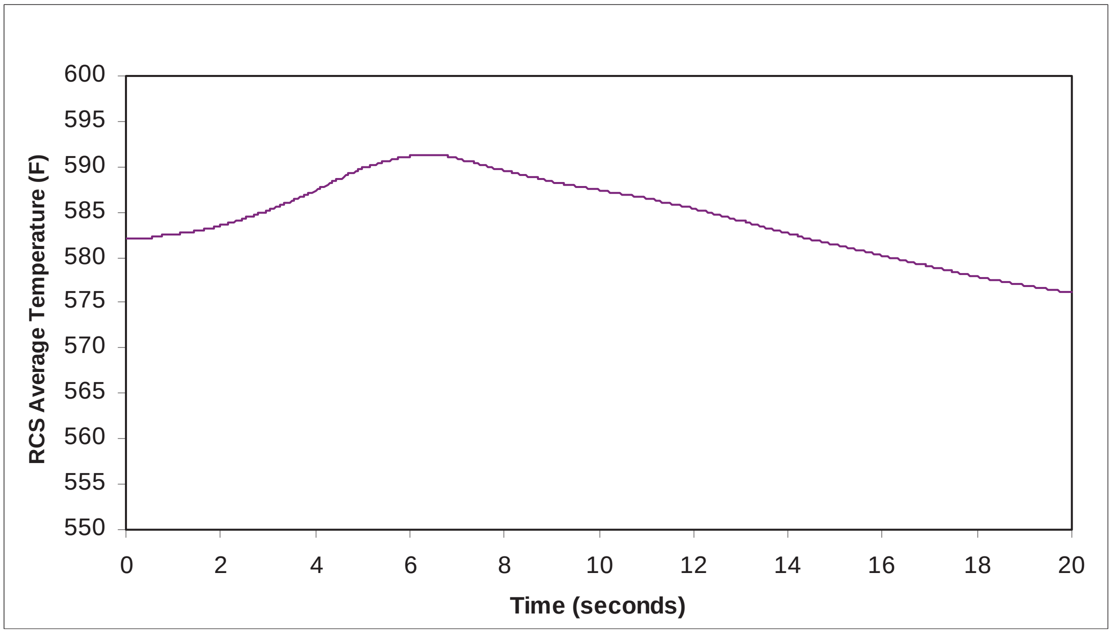


Figure 2.8.5.2.1-4: Turbine Trip Pressurizer Level versus Time

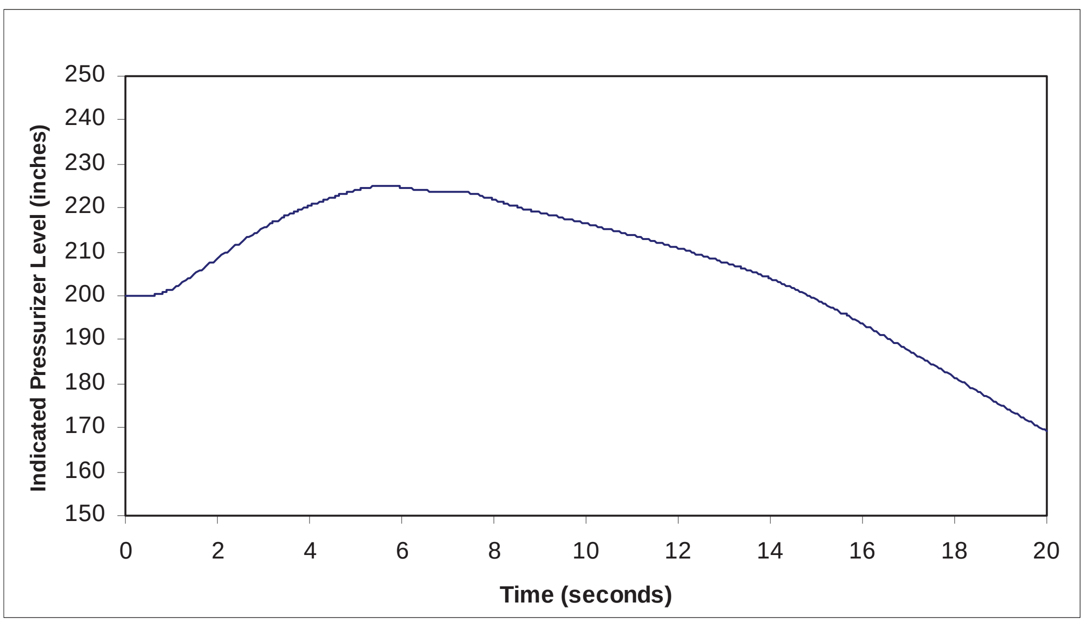
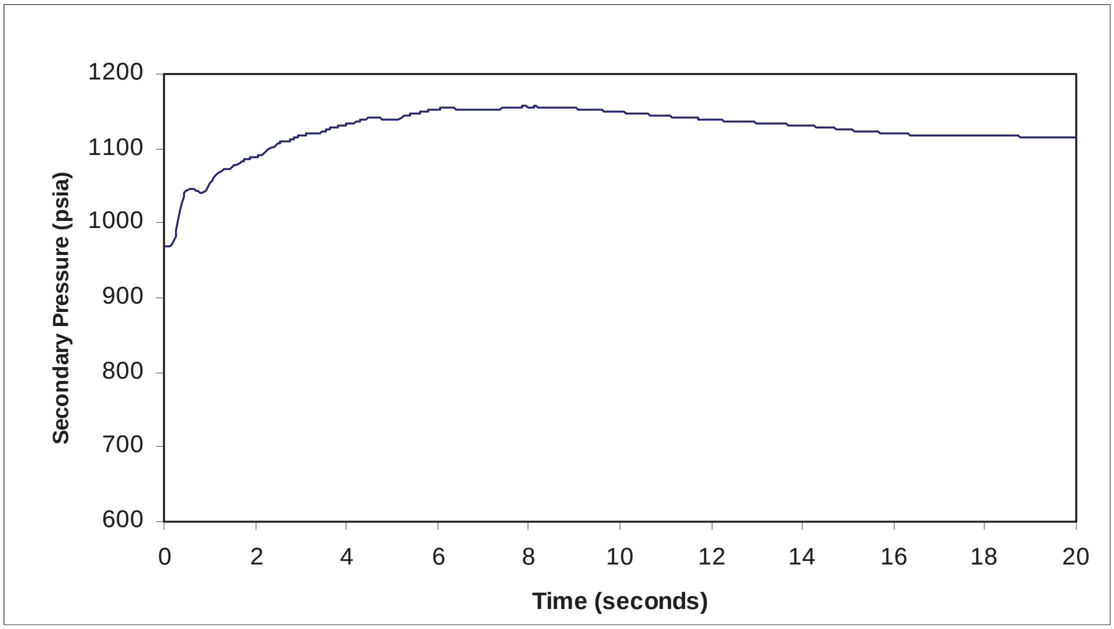


Figure 2.8.5.2.1-5: Turbine Trip Steam Line Pressure versus Time



2.8.5.2.2 Loss of Non-Emergency AC Power to the Station Auxiliaries

2.8.5.2.2.1 Regulatory Evaluation

The Loss of Non-Emergency AC Power to the Station Auxiliaries is assumed to result in the loss of all power to the station auxiliaries and the simultaneous tripping of all reactor coolant circulation pumps. This causes a flow coastdown as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. For this event, the scope of the review was satisfied by comparison to other, bounding analyses as described in the Technical Evaluation.

The NRC's acceptance criteria for the Loss of Non-Emergency AC Power to the Station Auxiliaries are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs);

- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation; and

- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs. [GDC-10]

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operations. [GDC-15]

- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded. [GDC-26]

2.8.5.2.2.2 Technical Evaluation

<u>Introduction</u>

For the Loss of Non-Emergency AC Power to Station Auxiliaries event, the CR-3 FSAR does not contain: (1) a sequence of events, (2) any analytical model used for analyses, and (3) any results associated with the event. CR-3 is not proposing to include this event to the Licensing basis, following implementation of EPU. However, the following qualitative evaluation is provided for the EPU conditions. The Loss of Non-Emergency AC Power to Station Auxiliaries is assumed to result in the loss of all power to the station auxiliaries. This situation, which could be the result of a complete loss of either the external (offsite) grid or the onsite AC Distribution System, is different from the loss of load condition, because in the latter case, AC power remains available to operate the stations auxiliaries. The major difference is that in the Loss-of-AC-power transient, all the reactor coolant pumps (RCPs) are tripped simultaneously by the initiating event, resulting in a flow coast-down as well as a decrease in heat removal by the secondary system. For the Loss-of-AC event, the control rods are also inserted quickly due to the loss of power.

Within a few seconds, the turbine trips, primary to secondary heat transfer is greatly reduced, and the pressure and temperature of the reactor coolant increase. The diesel generators start automatically and provide electric power to the vital loads. The sensible and decay heat loads are handled by actuation of the steam relief systems, and Emergency Feedwater (EFW) System.

The Loss of Non-Emergency AC Power to Station Auxiliaries event is a non-limiting event at CR-3, and can be shown to be bounded by other design basis events.

<u>Description of Analyses and Evaluations</u>

The Loss of Forced Reactor Coolant Flow, discussed in Section 2.8.5.3.1, Loss of Forced Reactor Coolant Flow, bounds the Loss of Non-Emergency AC Power to Station Auxiliaries event at the EPU conditions. The first few seconds after the Loss of Non-Emergency AC Power to the RCPs, the flow transient closely resembles the complete loss of flow incident, where core damage due to rapidly increasing core temperature is prevented by the reactor trip due to the loss of AC power. For a Loss of Non-Emergency AC Power to Station Auxiliaries event, the departure from nucleate boiling ratio (DNBR) results are less limiting since the reactor trips immediately due to the loss of AC power, and simultaneously with the initiation of RCP coastdown.

The Loss of Normal Feedwater Flow, discussed in Section 2.8.5.2.3, Loss of Normal Feedwater Flow, bounds the Loss of Non-Emergency AC Power to Station Auxiliaries event at the EPU conditions. The Loss of Non-Emergency AC power to Station Auxiliaries event is similar to the Loss of Normal Feedwater Flow in that the secondary side pressurizes following the turbine trip, and the RCS pressurizes due to the degradation in the secondary side heat sink. For the Loss of Non-Emergency AC Power to Station Auxiliaries event, the peak RCS and secondary side pressures are less limiting since the reactor will have already tripped prior to the pressurization, and the RCP heat is minimized following the RCP coastdown.

As described in Section 2.8.5.2.3, Loss of Normal Feedwater Flow, the Loss of Normal Feedwater Flow event is also the limiting transient in establishing the minimum EFW flow requirements, which assures that adequate feedwater is available to remove core decay heat, RCS stored energy, and heat associated with RCP operation. In addition, natural circulation cooldown demonstration for CR-3 is bounded by the 10 CFR Part 50 Appendix R cooldown analysis, as discussed in Section 2.5.1.4, Fire Protection.

Results

The Loss of Non-Emergency AC Power to Station Auxiliaries event is a non-limiting event at CR-3. The CR-3 evaluation has shown the Loss of Non-Emergency AC Power to the Station Auxiliaries event to be bounded by the Loss of Forced Reactor Coolant Flow, the Loss of Normal Feedwater Flow events and natural circulation cooldown.

2.8.5.2.2.3 Conclusion

CR-3 has reviewed the Loss of Non-Emergency AC Power to Station Auxiliaries event at the EPU conditions and concludes that the evaluations have adequately accounted for operation of the plant at the proposed power level and that the Loss-of-AC event is bounded by other analyzed events. CR-3 further concludes that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB limits will not be exceeded as a result of this event. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Loss of Non-Emergency AC Power to Station Auxiliaries event.

2.8.5.2.2.4 References

None.

2.8.5.2.3 Loss of Normal Feedwater

2.8.5.2.3.1 Regulatory Evaluation

A Loss of Normal Feedwater Flow could occur from pump failures, valve malfunctions, or a loss of offsite power. Loss of Normal Feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a Loss of Normal Feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The CR-3 review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses.

The NRC's acceptance for Loss of Normal Feedwater Flow criteria are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs);
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with margin sufficient to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operations; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs [GDC-10];
- FSAR Section 1.4.9, Reactor Core Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the pressure boundary will not be breached during normal operations including AOOs [GDC-15]; and
- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity

changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded. [GDC-26].

In addition, this event is a basis for establishing the pressurizer water level upper limit that is listed in ITS Section 3.4.8.

2.8.5.2.3.2 Technical Evaluation

<u>Introduction</u>

A Loss of Normal Flow through the secondary system results in a reduction in secondary heat removal and may be due to a Loss of Feedwater (LOFW) or a main feedwater line break. A LOFW accident is a complete loss of forced flow through the secondary system. A LOFW event can result from the inadvertent closure of a feedwater isolation valve, failure of a control valve, failure of a feedwater pump, or loss of offsite power.

A Loss of Normal Feedwater reduces the ability to remove heat generated by the core from the RCS. The Emergency Feedwater (EFW) System is provided to ensure that adequate feedwater is available to remove core decay heat, RCS stored energy, and heat associated with reactor coolant pump (RCP) operation. The LOFW accident is the limiting transient in terms of establishing the minimum EFW flow requirements.

<u>Description of Analyses and Evaluations</u>

The LOFW event was analyzed using the RELAP5/MOD2-B&W computer code (Reference 1). The pre-EPU LOFW analysis is described in FSAR Section 14.2.2.9. Two separate sets of initial conditions are considered for the EPU LOFW event. Condition A reflects the Chapter 14 safety analysis cases to meet FSAR 14.2.2.9 criteria. Condition B uses selected nominal inputs in order to confirm the pressurizer water level upper limit that is listed in ITS Section 3.4.8.

Condition A - The Safety analysis RCS overpressure and pressurizer overfill events:

The LOFW analysis was performed for the EPU conditions using the parameters listed at the end of this section. These parameters ensure a greater energy addition to the reactor coolant; therefore, they provide a conservative prediction for the LOFW events. Feedwater flow was assumed to decrease from 100 to 0% of full flow in 3.2 seconds. The 3.2-second coastdown of main feedwater flow to the steam generators (SGs) simulates the closing of one or more of the FW control valves..

To maximize the power response to the core temperature increase, beginning-of-life core conditions were assumed. The reduction in secondary heat removal will cause the average core moderator temperature and fuel temperatures to increase. Beginning of life core conditions, coupled with the increase in core average temperature and fuel temperatures, result in the least power decrease possible. With the least power decrease possible, the maximum amount of energy was transferred to the cladding and the reactor coolant.

When power to the control rod drive mechanisms (CRDMs) is interrupted following a reactor trip, a signal is sent to the Electro-Hydraulic Control System (EHCS). Upon receiving this signal, the EHCS trips the turbine. This turbine trip feature was modeled in the computer code. Once the turbine was tripped, the secondary pressure increased. The main steam safety valves (MSSVs) were modeled to control secondary pressure.

The power operated relief valve (PORV) is a non-safety grade component; therefore, it is not usually modeled in safety analyses. However, in the case of a LOFW, actuation of the PORV to control RCS pressure would aggravate the liquid insurge to the pressurizer by venting steam from the pressurizer at a lower pressure than would the pressurizer safety valves (PSVs). Similarly, pressurizer spray is a non-safety grade pressure control system. However, actuation of pressurizer spray flow worsens the pressurizer liquid level response during the event by condensing the pressurizer steam bubble.

Two events (RCS overpressure event and pressurizer overfill event) were analyzed. The RCS overpressure event was performed to confirm that the peak RCS pressure is less than the acceptance criterion (i.e., less than 110% of the design pressure). For the RCS overpressure event, neither the PORV nor the pressurizer spray was modeled. The pressurizer overfill event was performed to confirm that the pressurizer does not go liquid solid during the transient. For the pressurizer overfill event, the pressurizer spray, and the PORV were modeled. The PSVs are safety grade component and were modeled for both cases. Note that, the rated PSV capacity (90% of benchmark capacity) is used for the RCS overpressure event and the larger benchmark PSV capacity is used for the pressurizer overfill event. In addition, the pressurizer overfill model uses the surge line loss coefficient which is typical for 177 FA plant. For the RCS overpressure event a CR-3 plant-specific surge line loss coefficient is used. This loss coefficient is larger than the loss coefficient for the typical 177 FA plant and is therefore, appropriate for the RCS overpressure calculation. The smaller loss coefficient typical of 177 FA plants is conservative for the overfill scenario as it will allow higher insurge into the pressurizer an increase the calculated pressurizer level.

The LOFW analyses used the methodology defined in BAW-10193 (Reference 2). The RELAP5/MOD2-B&W code (Reference 1) was used in conjunction with the methodology, which also incorporated conservative setpoints and capacities to arrive at a conservative result.

The key input parameters and initial conditions used in the analysis of the LOFW accidents are as follows:

- The initial core power level was set to the nominal EPU power level plus heat balance uncertainty, or 3026.1 MWt (100.4% of 3014 MWt). A conservative RCP heat of 16.4 MWt was also included.
- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent with the increase in T_{AVG} planned in conjunction with the EPU. The Reactor Coolant System pressure is assumed to be the nominal value of 2170 psia, as measured at the hot leg tap.
- The RCPs were assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the minimum thermal design flow rate (374,880 gpm).
- The initial pressurizer level modeled was nominal indicated water level plus uncertainty (240 inches).

- Two PSVs were modeled with a nominal lift setpoint of 2514.7 psia, plus 3% lift tolerance, and 0% accumulation. A blowdown of 4% was also considered.
- Pressurizer spray was only modeled for the LOFW overfill event that maximized pressurizer level response.
- The PORV was only modeled for the LOFW overfill event that maximized pressurizer level response.
- The main feedwater flow was linearly reduced to zero over a period of 3.2 seconds at the start of the transient.
- Reactor trip was modeled to occur on a nominal high RCS pressure setpoint plus uncertainty (2400 psia).
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- A Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) and moderator coefficient (0.0 Δk/k/°F), typical of beginning-of-cycle conditions, were used since they yield the maximum rate of power increase.
- The MSSVs were modeled to lift at the nominal setpoints plus 3% lift tolerance and 3% accumulation. A nominal blowdown value of 5% was used. One of the lowest setpoint MSSVs on each SG was conservatively considered out of service.
- A minimum EFW flow of 660 gpm (330 gpm to each SG) was modeled with a 40-second delay after the low SG level initiation setpoint was reached. The minimum EFW flow is increased for the EPU (from 550 gpm), and the delay time assumption is decreased for the EPU (from 60 seconds). The new assumptions are required in order to ensure that the overheating is mitigated quickly such that the pressurizer does not become liquid solid during the overheating event. EFW temperature is assumed to be 120°F. Section 2.5.4.5 discusses the EFW modification.
- Steam generator tube plugging of 5% was modeled.
- Offsite power was available allowing the RC pumps to remain active and contribute to the severity of the overheating event.
- A single failure of one train of Emergency Feedwater Initiation and Control (EFIC) was assumed such that EFW flow was not initiated automatically in one train. Therefore, only one of the two EFW pumps was assumed available to provide flow to the SGs.
- No operator actions were credited.
- No Integrated Control System (ICS) actions were credited.

The specific FSAR acceptance criteria applied by CR-3 for this event were:

- The peak RCS pressure shall remain below 110% of the design pressure of the RCS (i.e., 2764.7 psia).
- Fuel pins will not experience departure from nucleate boiling by demonstrating that the minimum departure from nucleate boiling ratio (DNBR) remains above the applicable limit. This acceptance criterion is not analyzed for specifically. Since forced circulation is maintained throughout the event, the DNBR considerations are bounded by the loss-of-flow events.
- Offsite dose consequences remain less than the limits specified in 10 CFR 50.67. This also is not analyzed for specifically since the fission product barrier remains intact and releases are bounded by the steam generator tube rupture.

An additional acceptance criterion, that the pressurizer does not become water solid, was also used for this event. This criterion established that the minimum EFW flow assumed is adequate to prevent a liquid-solid pressurizer and subsequent relief of liquid through the PSVs. Demonstrating this ensures that the event will not evolve into a worse event, namely a small break loss of coolant accident (LOCA).

Condition B - The nominal RCS overpressure and pressurizer overfill event:

The nominal LOFW RCS overpressure and the pressurizer overfill events were evaluated for the EPU conditions. The methods and the computer code used for Condition A remain the same.

The key input parameters and the initial conditions used for Condition A analyses and discussed above in this section remain the same except the nominal value for the following parameters were used:

- The initial pressurizer level of 290 inches (compared to 240 inches for Condition A) was modeled.
- The initial core power level was set to the nominal EPU power level of 3014 MWt (compared to an uncertainty adjusted power of 3026.1 MWt for Condition A).
- The nominal setpoint for the Reactor Protection System (RPS) High RCS Pressure of 2369.7 psia (compared to 2400 psia for Condition A) was used.
- The nominal setpoint for the Emergency Feedwater Initiation and Control (EFIC) System Low Steam Generator (SG) Liquid Level function was used to provide a realistic time for delivery of EFW to the SGs. The nominal EFIC Low SG Liquid Level setpoint for the EPU conditions is 9.34 inches of water (compared to zero inches for Condition A) above the lower tap of Low Range instrument string.
- The nominal EFW fluid temperature was set to 90°F (compared to 120ºF for Condition A) to provide realistic energy removal by the SGs.

Cases analyzed with Condition B demonstrate that the acceptance criteria are met with an initial pressurizer level of 290 inches (indicated) using selected nominal inputs and therefore that this current ITS limit remains applicable to the EPU conditions.

Results

The results of the analyses showed that the worst-case peak pressurizer liquid level is obtained when credit is taken for pressurizer spray and PORV. The limiting case for peak RCS pressure control occurred when only the PSVs were modeled.

Condition A:

The sequence of events for the LOFW is listed in Table 2.8.5.2.3-1 and the calculated results are tabulated in Table 2.8.5.2.3-2. Figures 2.8.5.2.3-1 through 2.8.5.2.3-6 show transient plots of the significant plant parameters following a LOFW.

Initially, the EFW flow rate provided insufficient heat removal to match core decay heat and pump heat. Therefore, the reactor coolant continued to expand until the heat removal by EFW and heat absorption in the primary system metal matched decay heat and pump heat. Subsequently, the RCS temperature decreased, and RCS pressure decreased as the reactor coolant contracted. As the RCS pressure continues to decrease, depending on the case, the PSV and PORV reseat and the pressurizer spray is terminated. At no time in either case did the pressurizer liquid level exceed the elevation of the PSV and PORV inlet nozzles (Figure 2.8.5.2.3-2). This precluded any water relief though the valves.

The peak RCS pressure occurred in the lower downcomer region of the reactor vessel. Peak RCS pressure did not exceed 110% of the design pressure (i.e., 2764.7 psia) for either event (Figure 2.8.5.2.3-3). Since the RCPs remained operating, the fluid remained subcooled, and core power remained less than 112% throughout the analysis. Therefore, it is concluded that the minimum DNBR would remain above the correlation limit. All acceptance criteria listed in Section 2.8.5.2.3.1 were met for the CR-3 EPU LOFW accident analyses.

Condition B:

The sequence of events for the LOFW is listed in Table 2.8.5.2.3-3 and the calculated results are tabulated in Table 2.8.5.2.3-4. Figures 2.8.5.2.3-4 through 2.8.5.2.3-6 show transient plots of the significant plant parameters following a LOFW.

The peak pressurizer liquid level (40.42 ft) did not exceed the elevation of the PSV and PORV inlet nozzles (41.65 ft) and the peak RCS pressure (2733.57 psia) did not exceed 110% of the design pressure (i.e., 2764.7 psia). These results indicate that the LOFW events initiated from the nominal conditions with an initial pressurizer level of 290 inches (indicated) neither cause the peak RCS pressure to exceed 2764.7 psia nor fill the pressurizer i.e. at no time in either event will the pressurizer liquid level exceed the elevation of the PSV and PORV inlet nozzles. Therefore, the current ITS limit of 290 inches (indicated) remains applicable to the EPU conditions.

2.8.5.2.3.3 Conclusion

CR-3 has reviewed the analyses of the Loss of Normal Feedwater Flow event and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 also concludes that the evaluation has demonstrated that the plant will continue to meet the requirements for establishing the pressurizer water level upper limit that is listed in ITS Section 3.4.8, following implementation of the proposed EPU. CR-3 further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of the Loss of Normal Feedwater Flow. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Loss of Normal Feedwater Flow event.

2.8.5.2.3.4 References

1. BAW-10164PA-06, “RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis”.

2. BAW -10193PA-00, “RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors”.

Table 2.8.5.2.3-1: Sequence of Events for Loss of Feedwater– Condition A

Events	Overpressure Event (sec)	Overfill Event (sec)
MFW decrease initiated	0.0	0.0
MFW flow ends	3.2	3.2
PZR spray starts	N/A	8.5
High RCS pressure trip actuated	15.47	15.95
Control rods begin to fall, turbine stop valves (TSVs) begin to close	16.08	16.56
PORVs open (first time)	N/A	17.24
PSVs open (first time)	18.86	19.53
Peak RCS pressure occurs	19.49	19.9
Low SG level (EFW initiation setpoint) reached	66.83	65.65
EFW flow initiated to both SGs	106.83	105.65
Peak RCS temperature occurs	227.89	210.71
Peak PZR liquid level occurs	311.65	393.62
Transient terminated	800	800

Table 2.8.5.2.3-2: Results for Loss of Feedwater – Condition A

Parameters	Overpressure Event	Overfill Event	Acceptance Criterion
Peak RCS pressure (psia)	2750.63	2699.73	≤ 2764.7
Peak PZR liquid level (ft)	38.3	40.71	≤41.65 [1]

(1) The acceptance criterion applied is that the liquid level remains below the elevation of the PSV inlet.

Table 2.8.5.2.3-3: Sequence of Events for Loss of Feedwater – Condition B

Events	Overpressure Event (sec)	Overfill Event (sec)
MFW decrease initiated	0.0	0.0
MFW flow ends	3.2	3.2
PZR spray starts	N/A	8.14
High RCS pressure trip actuated	13.86	14.20
Control rods begin to fall, turbine stop valves (TSVs) begin to close	14.48	14.82
PORVs open (first time)	N/A	15.67
PSVs open (first time)	16.71	~18.0
Peak RCS pressure occurs	17.91	17.91
Low SG level (EFW initiation setpoint) reached	53.27	52.33
EFW flow initiated to both SGs	93.27	92.33
Peak RCS temperature occurs	220.97	229.59
Peak PZR liquid level occurs	317.89	276.67
Transient terminated	800.0	800.0

Table 2.8.5.2.3-4: Results for Loss of Feedwater – Condition B

Parameters	Overpressure Event	Overfill Event	Acceptance Criterion
Peak RCS pressure (psia)	2733.57	2700.0	≤ 2764.7
Peak PZR liquid level (ft)	38.53	40.42	≤ 41.65 [1]

(1) The acceptance criterion applied is that the liquid level remains below the elevation of the PSV inlet.

Figure 2.8.5.2.3-1: LOFW RCS Pressures – Condition A

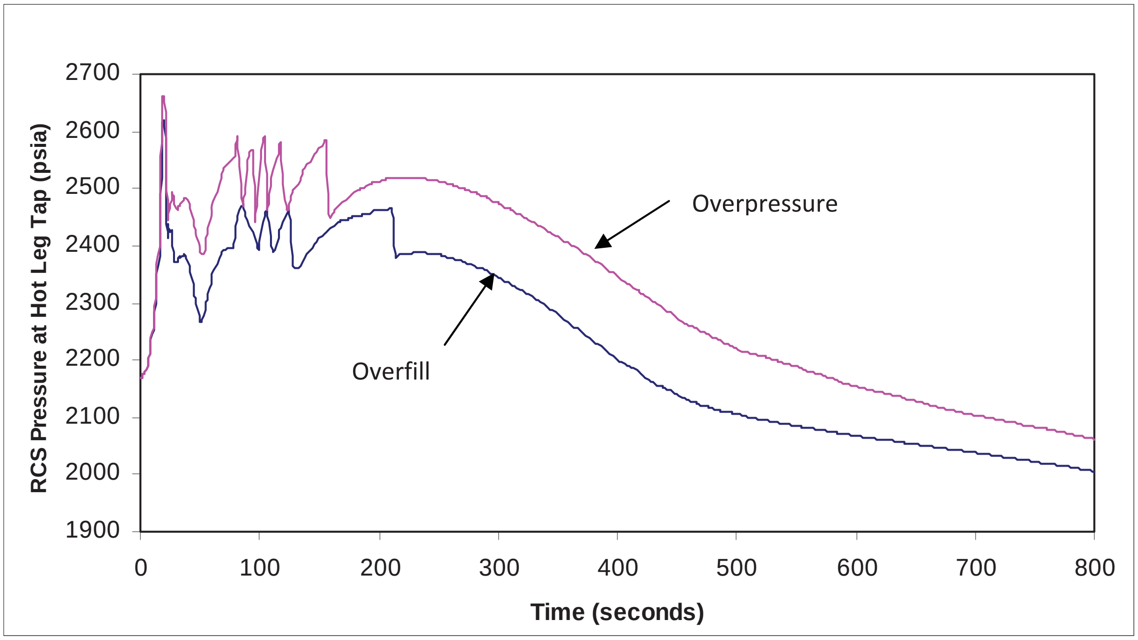


Figure 2.8.5.2.3-2: LOFW Pressurizer Level – Condition A

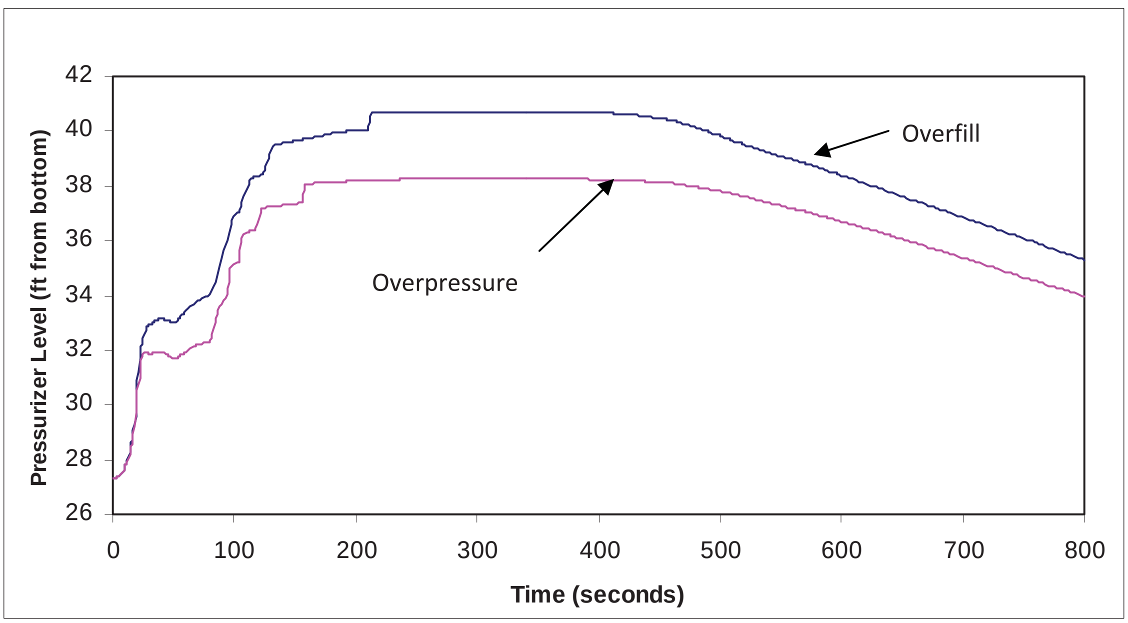


Figure 2.8.5.2.3-3: LOFW (Overpressure) Peak RCS Pressure – Condition A

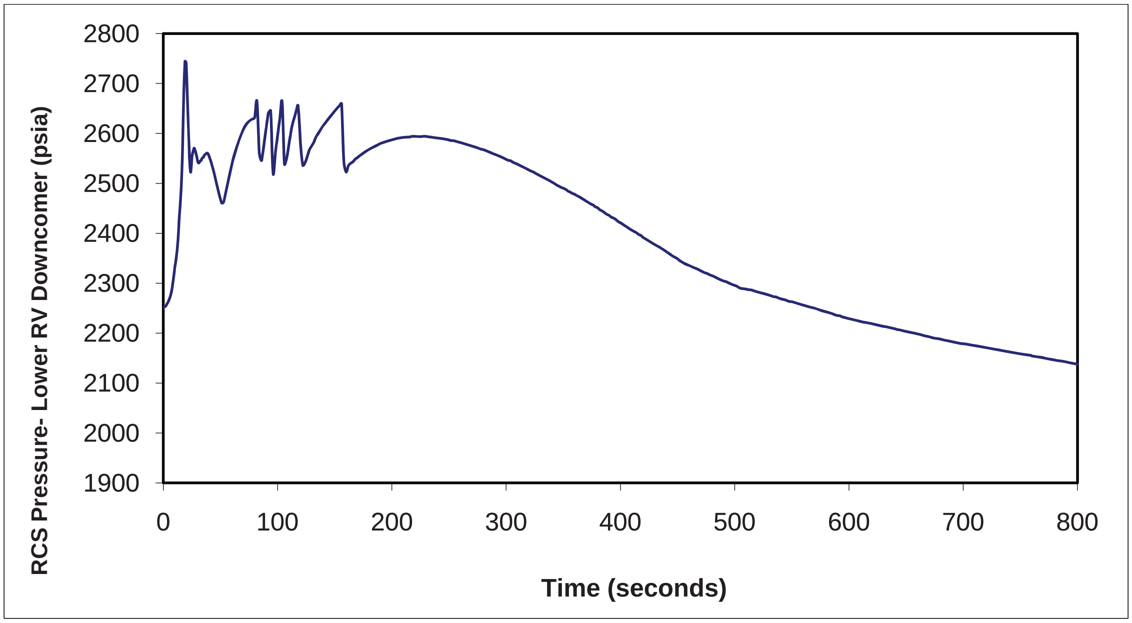


Figure 2.8.5.2.3-4: LOFW RCS Pressures – Condition B

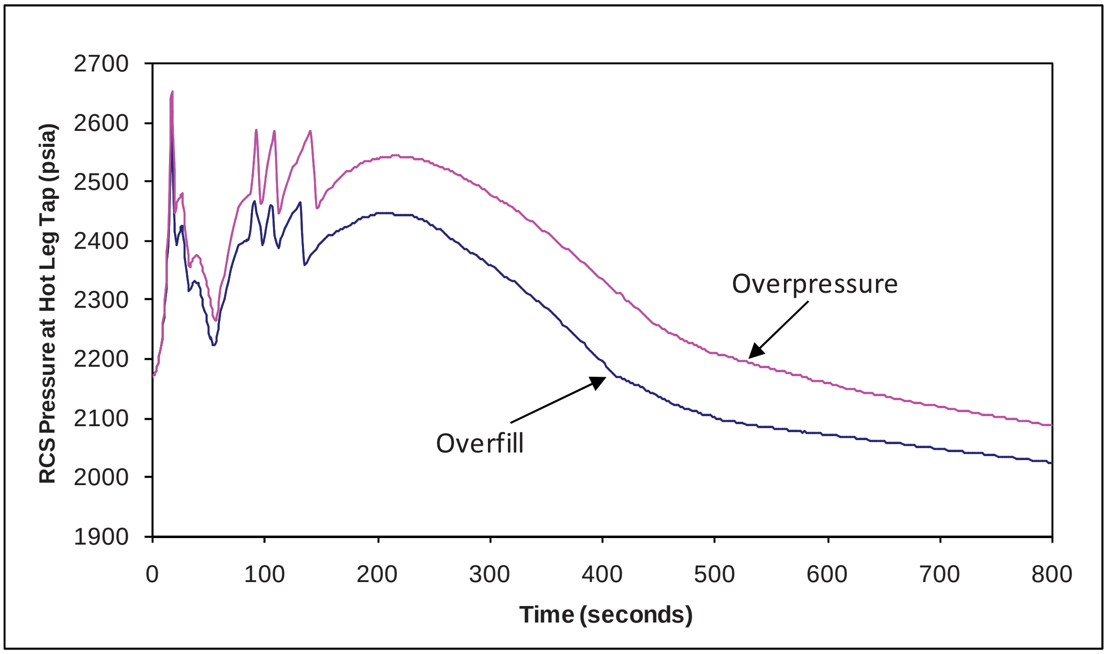


Figure 2.8.5.2.3-5: LOFW Pressurizer Level – Condition B

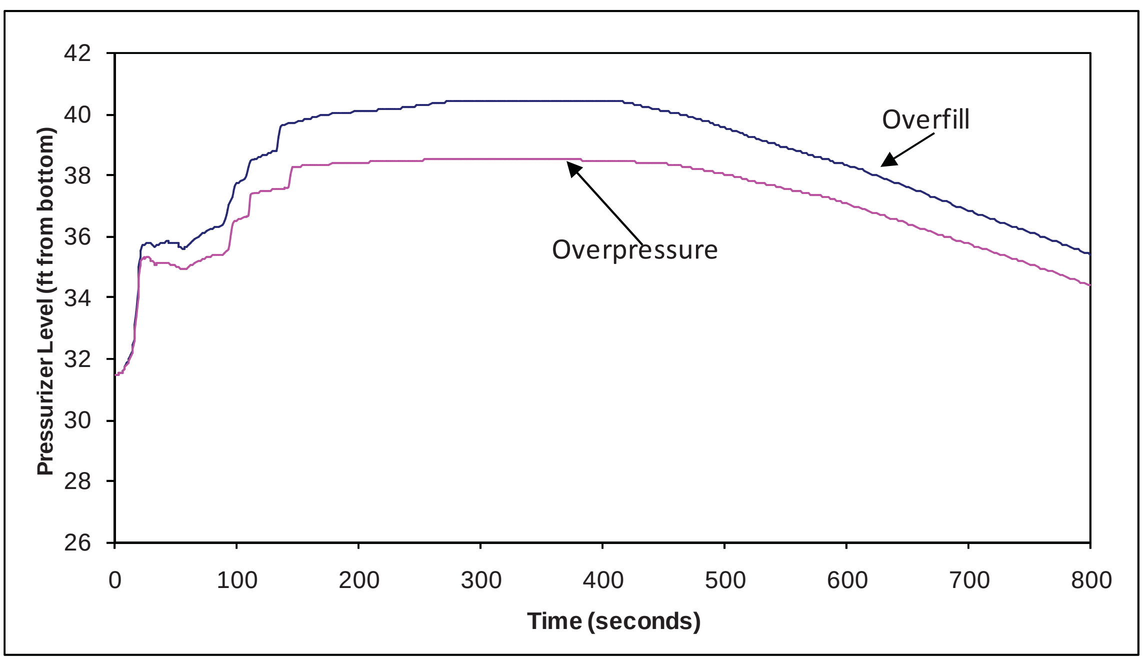
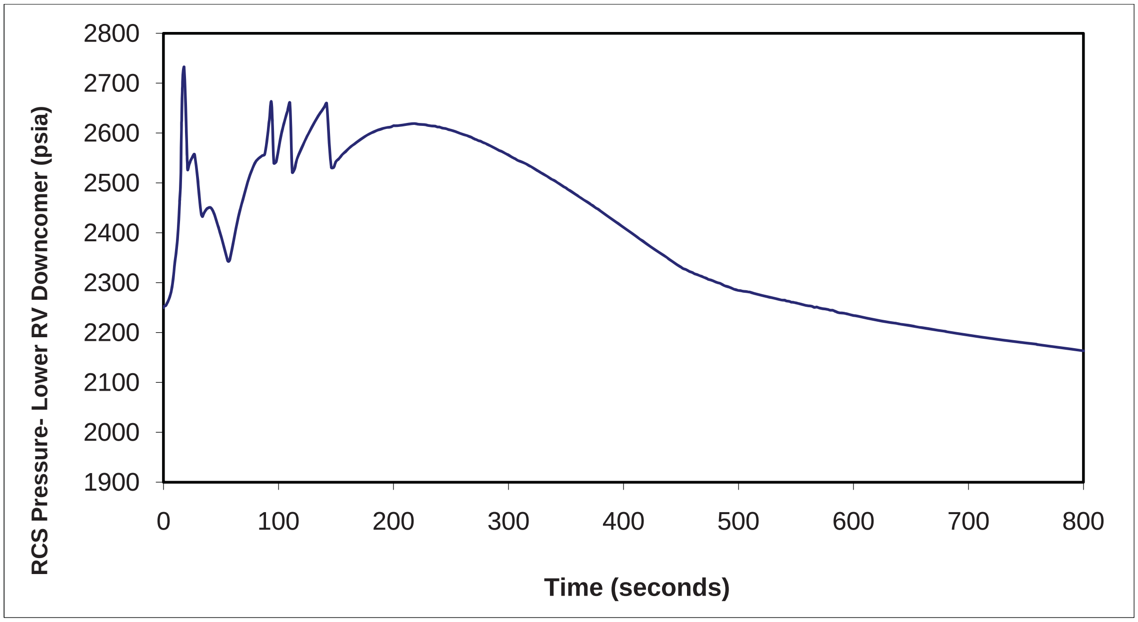


Figure 2.8.5.2.3-6: LOFW (Overpressure) Peak RCS Pressure – Condition B



2.8.5.2.4 Feedwater System Pipe Breaks Inside and Outside Containment

2.8.5.2.4.1 Regulatory Evaluation

Depending upon the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a Reactor Coolant System (RCS) cooldown (by excessive energy discharge through the break) or a RCS heatup (by reducing feedwater flow to the affected steam generator). In either case, reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered (1) postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed response of the reactor coolant and auxiliary systems, (5) the functional and operational characteristics of the Reactor Protection System, (6) operator actions, and (7) the results of the transient analyses.

The NRC's acceptance criteria for Feedwater System Pipe Breaks Inside and Outside Containment are based on:

- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by Emergency Core Cooling System (ECCS), of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;
- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the Reactor Coolant Pressure Boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core;
- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a non brittle manner and the probability of a rapidly propagating fracture is minimized; and
- GDC-35, insofar as it requires the Reactor Cooling System (RCS) and associated auxiliaries be designed to provide abundant emergency core cooling.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.28, Reactivity Hot Shutdown Capability, FSAR Section 1.4.29, Reactivity Shutdown Capability, and FSAR Section 1.4.30, Reactivity Holddown Capability - insofar as these criteria require that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained.. [GDC-27]
- FSAR Section 1.4.32, Maximum Reactivity Worth of Control Rods, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the Reactor Coolant Pressure Boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. [GDC-28]
- FSAR Section 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention, and FSAR Section 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention - insofar as these criteria require that the RCPB be design with sufficient margin to assure that, under specified conditions, it will behave in a non brittle manner and the probability of a rapidly propagating fracture is minimized. [GDC-31]
- FSAR Section 1.4.37, Engineered Safety Features Basis for Design, FSAR Section 1.4.41, Engineered Safety Features Performance Capability, FSAR Section 1.4.42, Engineered Safety Features Components Capability, and FSAR Section 1.4.44, Emergency Core Cooling Systems Capability - insofar as these criteria require the Reactor Cooling System and associated auxiliaries be designed to provide abundant emergency core cooling. [GDC-35]

2.8.5.2.4.2 Technical Evaluation

Introduction

A feedwater line break (FWLB) event is the result of a break in the piping of the Main Feedwater (MFW) System. The pre-EPU FWLB analysis is described in FSAR Section 14.2.2.9. The initial and boundary conditions for the event are such that the results are bounding for feedwater line breaks inside and outside of containment. While it is noted in Section 2.8.5.2.4.1, that the feedwater line break may result in either a heatup or cooldown event, for CR-3 the most limiting event is the double-ended rupture of a main feedwater line which results in heatup of the RCS. The resulting heatup transient is summarized below.

Description of Analyses and Evaluations

The main FWLB event was analyzed for the EPU conditions using the RELAP5/MOD2-B&W computer code (Reference 1).

The FWLB event is sufficiently severe that the pressurizer fills. As a result, the Pressurizer Safety Valves (PSVs) begin to pass single-phase liquid instead of single-phase steam or a two-phase mixture. The PSVs of the type installed at CR-3 achieve satisfactory performance for fluid temperatures greater than ~550°F. Based on this, an additional protection criterion is required to show that the PSV fluid inlet temperature remains greater than 600°F. Under these conditions, the PSVs have sufficient relieving capacity to prevent the system pressure from exceeding the accident acceptance criterion regardless of the inlet fluid conditions. The PSVs continue to cycle until the heat removal capability of the Emergency Feedwater (EFW) fluid is adequate to remove core decay heat, RCS stored energy, and heat associated with Reactor Coolant Pump (RCP) operation. At this time the RCS temperature begins to decrease and the FWLB event analysis is terminated.

Two separate cases to analyze the RCS overpressurization were performed, one with pressurizer spray and one without. EFW was modeled at a flow rate of 550 gpm following a 60 second delay time after the Steam Generator (SG) low level setpoint was reached. The EFW flow modeled is conservatively less than the minimum that will be available for the EPU operation, and the delay time for initiation of EFW flow is conservatively longer than what will be credited for the EPU plant configuration.

The FWLB analyses used the methodology defined in BAW-10193 (Reference 2). The RELAP5/MOD2-B&W code was used in conjunction with the methodology, which also incorporated conservative setpoints and capacities to arrive at a conservative result. The key input parameters and initial conditions used in the analysis of the FWLB event are as follows:

- The initial core power level is consistent with the targeted EPU power level of 3026.1 MWt (100.4% of 3014 MWt). A conservative RCP heat of 16.4 MWt was also included.
- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent the increase in T_{AVG} that is planned in conjunction with the EPU. The initial hot leg pressure of 2170 psia was assumed in the analysis.
- The RCPs were assumed to continuously operate throughout the transient providing a constant reactor coolant volumetric flow equal to the minimum flow rate (374,880 gpm).
- The initial pressurizer level modeled was nominal indicated water level plus uncertainty (240 inches).
- Two PSVs were modeled with a nominal lift setpoint of 2500 psig, plus 3% lift tolerance, and 0% accumulation. A blowdown of 4% was also considered.
- Pressurizer heaters were not modeled for the FWLB analysis. Pressurizer spray was not modeled in the FWLB analysis for peak RCS pressure. However, the spray was included for the FWLB case that confirmed the PSV fluid inlet temperature remained above 600°F.
- The Pilot Operated Relief Valve (PORV) was not modeled for the FWLB analysis.

- Reactor trip was modeled to occur on a nominal high RCS pressure setpoint plus uncertainty (2445.45 psia), which includes the effects of elevated pressure that may exist inside containment post-FWLB.
- A minimum shutdown margin of 1.0 %Δk/k at hot zero power was assumed. This is conservative in comparison to the Modes 1 and 2 minimum shutdown margin of 1.3 %Δk/k that is planned for the CR-3 EPU.
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- A Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) and a moderator coefficient (0.0 Δk/k/°F), typical of beginning-of-cycle conditions, were used since they yield the maximum rate of power increase.
- The Main Steam Safety Valves (MSSVs) were modeled to lift at the nominal setpoints plus 3% lift tolerance and 3% accumulation. A nominal blowdown value of 5% was used. One of the lowest setpoint MSSVs on each SG was conservatively considered out of service.
- A minimum EFW flow of 550 gpm (total) was modeled with a 60-second delay after the low SG level initiation setpoint was reached. Each of these values was conservatively chosen as described above. EFW temperature was 120°F. The CR-3 Emergency Feedwater Initiation and Control (EFIC) System contains feed-only-good generator logic, thus all EFW was provided to the unaffected SG.
- Steam generator tube plugging of 5% was modeled.
- Offsite power was available allowing the RC pumps to remain active and contribute to the severity of the overheating event.
- A single failure of one train of EFIC was assumed such that EFW flow was not initiated automatically in one train. Therefore, only one of the two EFW pumps was assumed available to provide flow to the SGs.
- No operator actions were credited.
- No Integrated Control System (ICS) actions were credited.

The FWLB is considered a limiting fault event per the CR-3 FSAR (Section 14.2.2.9.2). The FSAR acceptance criteria for a limiting fault include the following restrictive criteria:

a. RCS pressure does not exceed 110% of design pressure (2500 psig), or 2750 psig.

b. Fuel pins will not experience departure from nucleate boiling by demonstrating departure from nucleate boiling ratio (DNBR) remains above the applicable limit.

However, for EPU CR-3 Licensing Basis is being revised to reflect an RCS pressure limit of 120% of design pressure, or 3000 psig. This is consistent with acceptance criteria specified in the Standard Review Plan (SRP) 15.2.8 for the FWLB event.

The FWLB will result in filling the pressurizer and passing liquid through the PSVs. To ensure that the PSVs will remain functional, the analysis will show that liquid passed by the PSVs remains above 600°F.

Results

The sequence of events for the FWLB accident is listed in Table 2.8.5.2.4-1 and the calculated results are tabulated in Table 2.8.5.2.4-2. Plots that demonstrate the transient response following a FWLB are provided as Figures 2.8.5.2.4-1 through 2.8.5.2.4-5.

Following initiation of the FWLB, the blowdown of the affected SG resulted in a reduction in the secondary heat removal and actuated EFIC on low SG level. The mismatch between energy addition to the reactor coolant and the secondary heat removal caused the reactor coolant to heat up and pressurize. As a result of the increasing pressure, the reactor tripped on high RCS pressure. After reactor trip, the RCS pressure continued to increase until the PSVs lifted. EFW initiation occurred within 70 seconds after the start of the event with a flow rate of 550 gpm provided to the unaffected SG.

The results of the two FWLB overpressurization cases show that the effect of pressurizer spray was minimal on the timing and magnitude of several key event parameters as shown in Table 2.8.5.2.4-2. The main parameter of interest for the case with pressurizer spray was the upper pressurizer fluid temperature which was predicted to remain above 600°F for the analysis duration. The FWLB case without pressurizer spray produced the limiting peak RCS pressure.

The peak RCS pressure occurred in the lower downcomer region of the reactor vessel and did not exceed 120% of the design pressure of 2500 psig (3000 psig) for either FWLB case (Figure 2.8.5.2.4-1). RCS peak pressure below 120% of design pressure is identified in the SRP as acceptance criteria for very low probability events like double-ended guillotine breaks. Because of the change in the current licensing basis acceptance criteria, an additional evaluation of the impact of the increased peak pressure limit was performed. Consistent with the guidance in SRP 15.2.8, RCPB instrumentation and RCP seals were evaluated. The evaluation concluded that the increase in the peak pressure limit did not represent a challenge to the integrity of these components, nor their ability to perform active safety functions. The Steam Generator was also assessed for higher peak RCS pressures resulting from the FWLB, consistent with the Regulatory Guide 1.121 requirements for maintaining steam generator tube integrity. This assessment resulted in minor adjustment to the allowable tube flaw sizes, as discussed in LAR, Section 2.2.2.5 Steam Generators and Supports.

Since the acceptance criterion for peak pressure is not exceeded, and since the heatup event does not result in a potential for brittle fracture, it is assured that the RCPB will behave in a nonbrittle manner and the probability of propagating fracture of the RCPB is minimized. Since the RCPs remained operating, the RCS fluid remained subcooled, and core power remained less than 112% throughout the analysis, it is concluded that the minimum DNB ratio would remain above the applicable correlation limit, demonstrating abundant core cooling. All acceptance criteria were met for the CR-3 EPU FWLB event.

2.8.5.2.4.3 Conclusion

CR-3 has reviewed the analyses of Feedwater System Pipe Breaks Inside and Outside of Containment and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a nonbrittle manner, the probability of propagating fracture of the RCPB is minimized, and abundant core cooling will be provided. Based on this, CR-3 concludes that the plant will continue to meet the CR-3 current licensing basis with respect to the requirements of FSAR Sections 1.4.28, 1.4.29, 1.4.30, 1.4.32, 1.4.34, 1.4.35, 1.4.37, 1.4.41, 1.4.42, and 1.4.44 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Feedwater System Pipe Breaks Inside and Outside of Containment.

2.8.5.2.4.4 References

1. BAW-10164P-A-06, “RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis”.

2. BAW-10193P-A-00, “RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors”.

Table 2.8.5.2.4-1: Sequence of Events for Feedwater Line Break

Event	Overpressure Event No PZR Spray (sec)	Overpressure Event PZR Spray Operating (sec)
Transient initiated	0.0	0.0
MFW to both SGs interrupted	1. 0E-6	1. 0E-6
PZR spray begins	N/A	~3
EFIC actuated on Low SG-B Level	6.515	6.35
Peak thermal power occurs	6.522	6.518
RPS high RCS pressure trip actuated	8.802	8.948
Control rods begin to insert TSVs begin to close	9.415	9.56
Initial PSV lift occurs	~12	~12
Peak RCS pressure occurs	13	12.968
Affected SG depressurization complete	~36	~36
EFW flow begins	~68	~68
Peak T_{AVG} occurs	346	412
Final PSV closure occurs	352	226
PZR spray ends	N/A	~1016
Transient terminated	600	2000

Table 2.8.5.2.4-2: Results for Feedwater Line Break

Parameter	Overpressure Event No PZR Spray	Overpressure Event PZR Spray Operating
Peak RCS pressure (psia)	2896.20	2885.23
Peak thermal power (%RTP)	100.49	100.49
Peak T_{AVG} (F)	623.56	624.09

Figure 2.8.5.2.4-1: FWLB RCS Peak Pressure versus Time

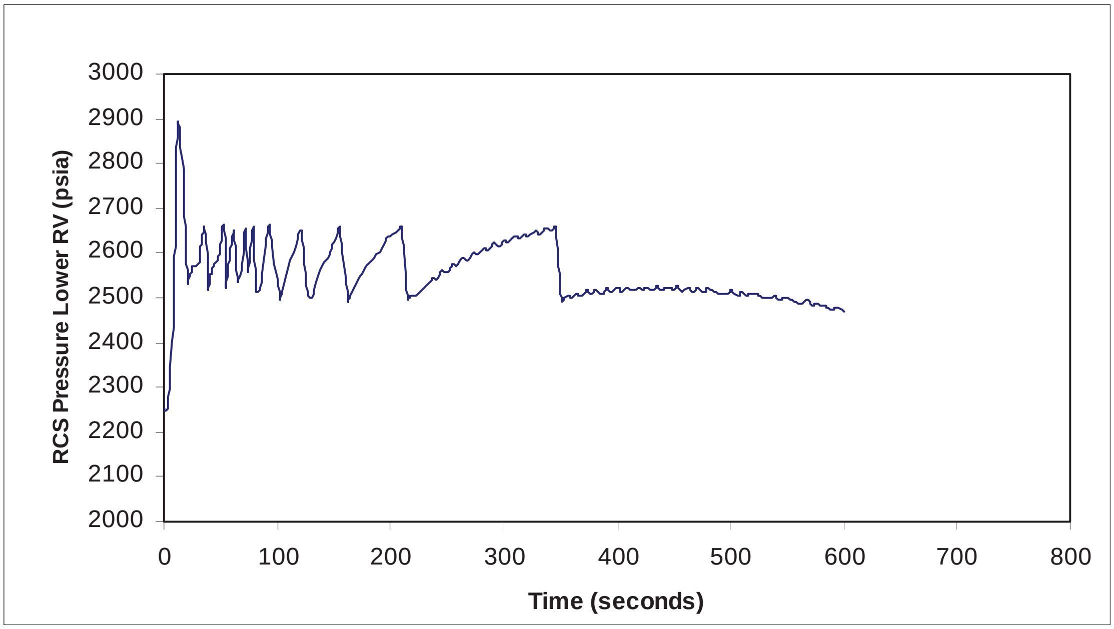


Figure 2.8.5.2.4-2: FWLB RCS Temperatures versus Time

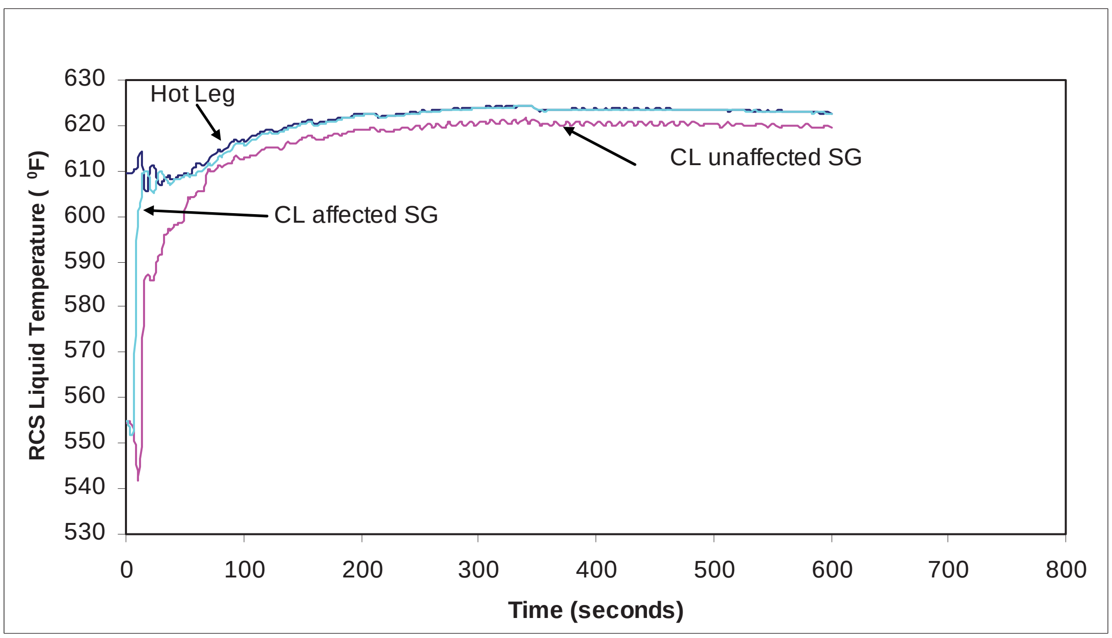


Figure 2.8.5.2.4-3: FWLB Pressurizer Level versus Time

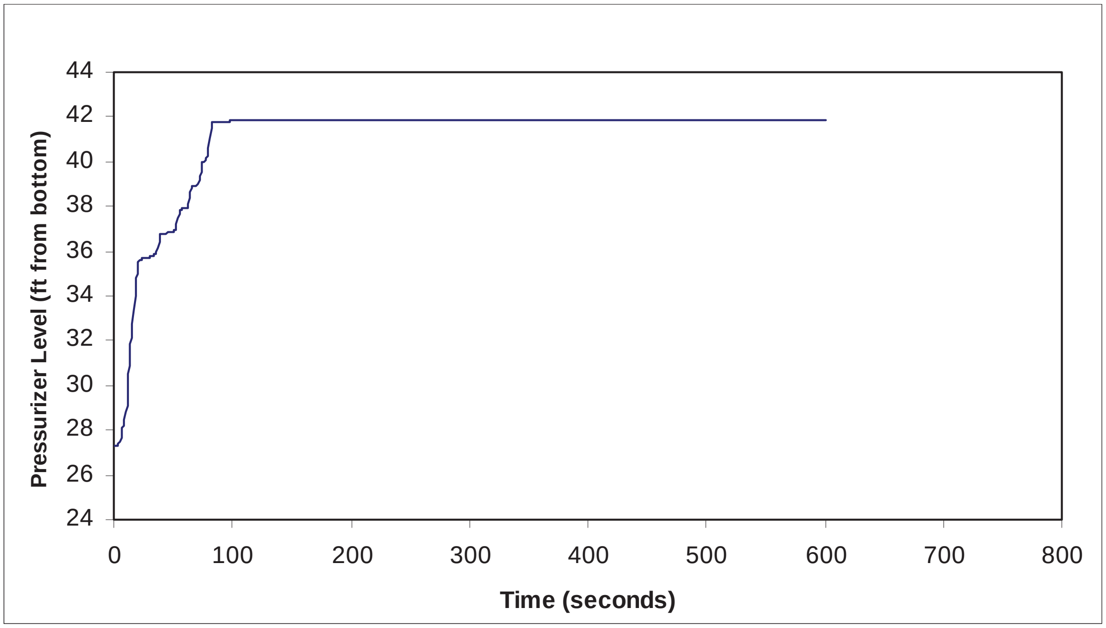


Figure 2.8.5.2.4-4: FWLB SG Pressures versus Time

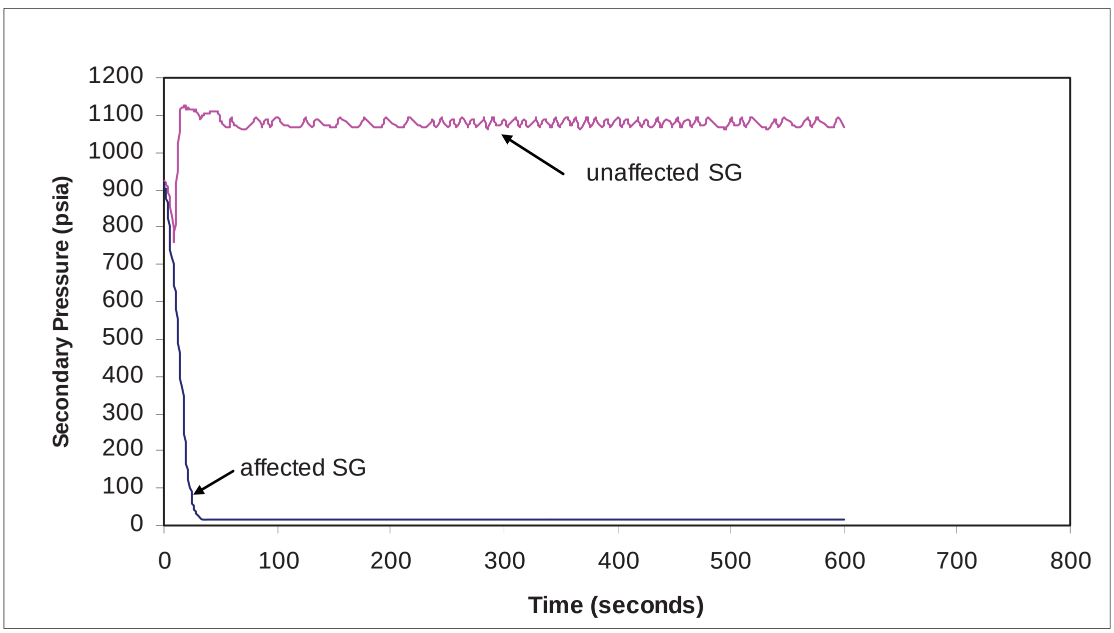
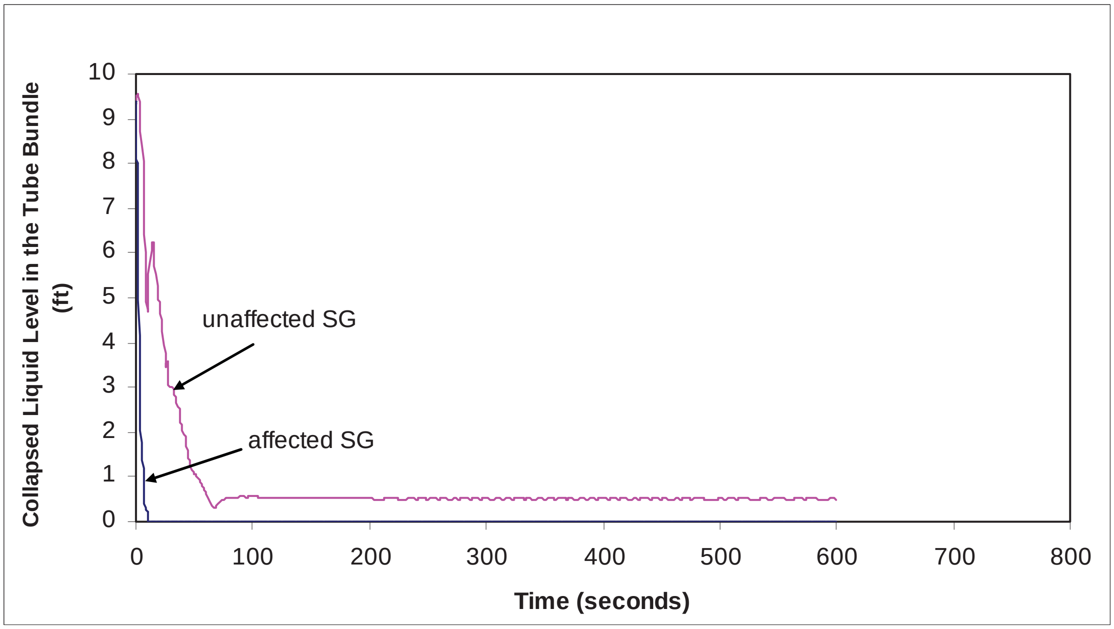


Figure 2.8.5.2.4-5: FWLB SG Levels versus Time



2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

2.8.5.3.1.1 Regulatory Evaluation

A Decrease in Reactor Coolant Flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if specified acceptable fuel design limits (SAFDLs) are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal-hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of the reactor system components, (5) the functional and operational characteristics of reactor protection system, (6) the operator actions, and (7) the results of the transient analyses.

The NRC’s acceptance criteria for a Decrease in Reactor Coolant Flow are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations including anticipated operational occurrences (AOOs);
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operations; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs; [GDC-10]
- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design of the RCPB are not exceeded during any condition of normal operations; [GDC-15] and

- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded [GDC-26].

2.8.5.3.1.2 Technical Evaluation

Introduction

As described in FSAR Section 14.1.2.6, a loss-of-coolant-flow (LOCF) accident is the Loss of or Reduction in Forced Flow through the Reactor Coolant System. The Loss of or Reduction in Forced Flow may be due to a mechanical failure in the reactor coolant pump(s) (RCP(s)) or a loss of electrical power to the RCP(s). A LOCF event results in a reduction in the heat removal capability of the reactor coolant.

The LOCF analyses used the methodology in Reference 1 to provide the thermal-hydraulic forcing functions to the departure from nucleate boiling (DNB) analyses. The RELAP5/MOD2-B&W code (Reference 2) was used in conjunction with the methodology, which also incorporated conservative setpoints and capacities to arrive at a conservative result. The key input parameters and initial conditions used in the thermal-hydraulic analysis of the LOCF events are as follows:

- The initial core power level was set to 3026.1 MWt, which is equivalent to the planned EPU power level plus heat balance uncertainty. Additionally, a conservative RCP heat of 16.4 MWt was also included.
- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent with the planned increase in T_{AVG} planned in conjunction with EPU.
- The RCS initial pressure is the nominal value of 2155 psig (2169.7 psia) in the reactor coolant hot leg.
- An initial RCS flow equal to the minimum thermal design flow rate was used (374,880 gpm).
- The initial pressurizer level modeled was the nominal value (220 inches) for the 1- and 4-pump coastdown events.
- Pressurizer heaters were not modeled for the LOCF analyses since a lower pressure is conservative with respect to DNB calculations.
- Pressurizer spray (190 gpm) was modeled since pressurizer spray would keep the pressure lower, which is conservative for DNB calculations.
- The main feedwater flow was isolated coincident with reactor trip by linearly reducing the flow to zero over 3.0 seconds.
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- For the 4- Pump Coast Down (PCD) event, the RPS will initiate a reactor trip using the power/pump monitors trip function of the RPS. The trip is initiated when all power is lost to the RCPs in one or more RCS loops. A conservative delay time for the power/pump monitor trip function was then assumed prior to control rod insertion.
- Reactor trip is initiated by the Nuclear Overpower RCS Flow and Measured Axial Power Imbalance (Power/Imbalance/Flow or PIF) trip function, specifically the flux-to-flow portion of the trip. The flux-to-flow trip occurs when the ratio of the percent full power determined by

the measured neutron flux divided by the percent flow exceeds the setpoint. A flux-to-flow setpoint for CR-3 of 1.13 %full power/%flow was used, which accounts for instrument uncertainties in order to protect the allowable values contained in the core operating limits report. An appropriate response time is modeled prior to control rod insertion.

- A Doppler temperature coefficient (-1.30 x 10^{-5} Δk/k/°F) and moderator coefficient (0.0 Δk/k/°F), typical of beginning-of-cycle conditions, were used since they yield the maximum rate of power increase.
- Steam generator tube plugging of 5% was modeled. Consideration of tube plugging is conservative due to the increased resistance to flow within the RCS, and 5% tube plugging bounds actual tube plugging amounts.
- Offsite power was available in accordance with the licensing basis for the event.
- No single failure was assumed in accordance with the licensing basis for the event. There are no credible single failures that could result in more limiting results.
- No operator actions were credited due to the short timeframe of the event.

The CR-3 FSAR acceptance criteria for the coastdown events are that departure from nucleate boiling ratio (DNBR) shall remain above the limit, which is 1.45 for the applicable EPU fuel type.

<u>Description of Analyses and Evaluations</u>

The 4 pump coastdown event results in a complete loss of forced flow in the RCS. This results in a rapid decrease in RCS flow with power level remaining relatively constant. The DNB ratios decrease until the reactor trips on the power pump monitor trip at which point core power decreases and the DNB recovers. The thermal-hydraulic analysis was performed using RELAP5/MOD2-B&W. Upon event initiation, the affected RCPs begin to coast down. As a result, RCS flow rate decreases with the core power level remaining relatively constant. The pump monitor trip signal is generated immediately, and control rod insertion begins 1.60 seconds after event initiation. The thermal-hydraulic analysis is continued for a short period of time beyond when the DNB transient is mitigated (minimum DNBR reached) upon control rod insertion. The sequence of events is provided in Table 2.8.5.3.1-1, and the results of the analysis are depicted on Figures 2.8.5.3.1-1 through 2.8.5.3.1-3.

The 1 pump coastdown is the loss of a single RCP due to loss of power or mechanical failure while the reactor is in operation at full power. All other RCPs remain operating. The 1 pump coastdown results in the decrease of RCS flow while the core power remains relatively constant. The DNB ratio is thus decreased and reaches a minimum near the time flux-to-flow trip setpoint is reached of control rod insertion due to the reactor trip on flux-to-flow ratio. The thermal-hydraulic analysis was performed using RELAP5/MOD2-B&W. Upon event initiation, the affected RCP begins to coast down. The flux-to-flow trip setpoint is reached 5.23 seconds after event initiation. The control rods begin to insert 7.41 seconds after event initiation. The DNB transient is mitigated (minimum DNBR reached) upon control rod insertion. The sequence of events is provided in Table 2.8.5.3.1-2, and the results of the analysis are depicted on Figures 2.8.5.3.1-4 through 2.8.5.3.1-6.

The 4 pump coastdown and 1 pump coastdown analyses are performed by inputting the transient flow and power curves calculated by RELAP5/MOD2-B&W for each event into a LYNXT model based on the steady state DNB analysis. Per BAW-10179 (Reference 3), the power curve for each event is adjusted to discount the favorable effects of the moderator temperature coefficient, which would cause a slight decrease in power prior to the control rods entering the core due to the increased coolant temperature. The models are initiated from 100% of nominal core power and use design radial and axial peaking. The heat balance error is not included in the initial core power, since it is included within the statistical core design methodology for calculating DNBR. The analyses calculate the DNBR at various time steps within the transient, with greater resolution in the period between the beginning of the flow coastdown and the decrease in power due to control rod insertion. The minimum DNBR for the 4 pump coastdown and 1 pump coastdown transients are the lowest DNBR calculated at any time step throughout each event.

Results

As detailed in Section 2.8.3, the minimum DNBR is adequately high for the 4 pump coastdown and 1 pump coastdown transients, indicating that no DNB will occur during the 1 pump and 4 pump coastdown transients. The 4 pump coastdown resulted in minimum DNBR of 1.54, and the 1 pump coastdown resulted in a minimum DNBR of 1.62, versus the thermal design limit of 1.45. The more limiting of these two transients – the 4 pump coastdown – is used to calculate the operating limit maximum allowable peaking (MAP) limits which are used by the maneuvering analysis to set alarm functions.

2.8.5.3.1.3 Conclusion

CR-3 has reviewed the analyses of the Loss of Forced Reactor Coolant Flow events and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLS and the RCPB limits will not be exceeded as a result of this event. Based on this, CR-3 concludes that the plant will continue to meet the requirements of CR-3 FSAR Sections 1.4.6, 1.4.9, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Loss of Forced Reactor Coolant Flow event.

2.8.5.3.1.4 References

1. BAW-10193PA-00 (Proprietary), “RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors.”

2. BAW-10164PA-06 (Proprietary), “RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis.”

3. BAW-10179PA-007 (Proprietary), “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses.”

Table 2.8.5.3.1-1: Sequence of Events for 4 Pump Coastdown

Event	Time (s)
Event Initiated	0.0
RPS Reactor Pump Monitor Trip Signal Received	0.01
Control Rods Begin to Insert	1.60
Minimum DNBR	2.50
Transient Ends	4.00

Table 2.8.5.3.1-2: Sequence of Events for 1 Pump Coastdown

Event	Time (s)
Event Initiated	0.0
RPS Trip Signal Received (Flux/Flow > 1.13)	5.23
Control Rods Begin to Insert	7.41
Minimum DNBR	7.80
Transient Ends	12.00

Figure 2.8.5.3.1-1: CR-3 EPU 4 Pump Coastdown Analysis

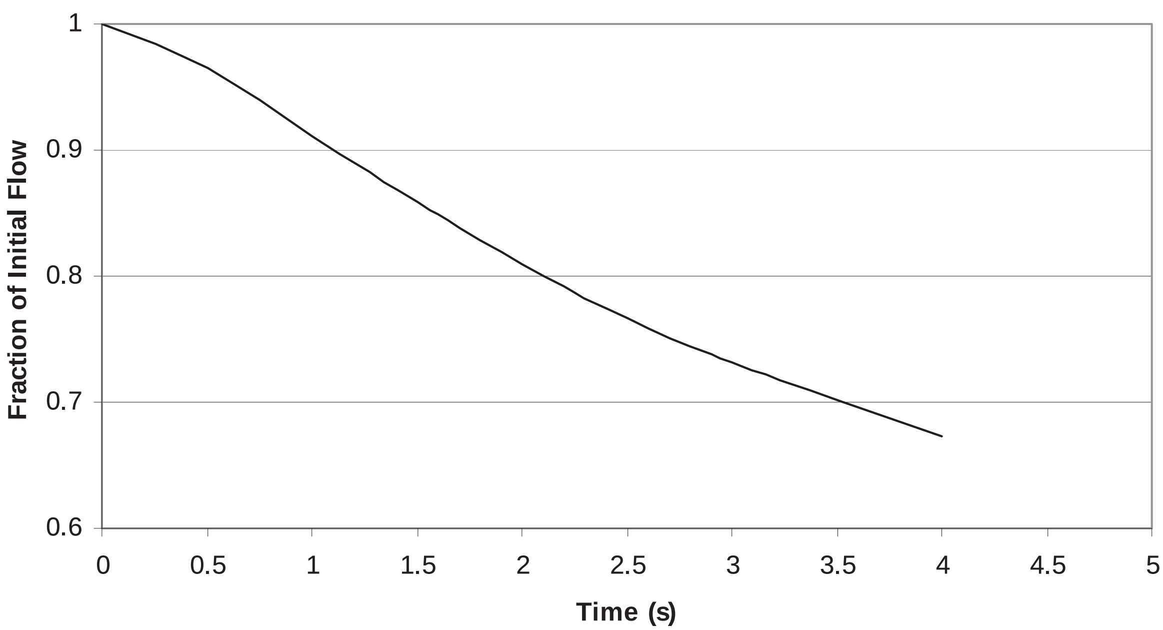
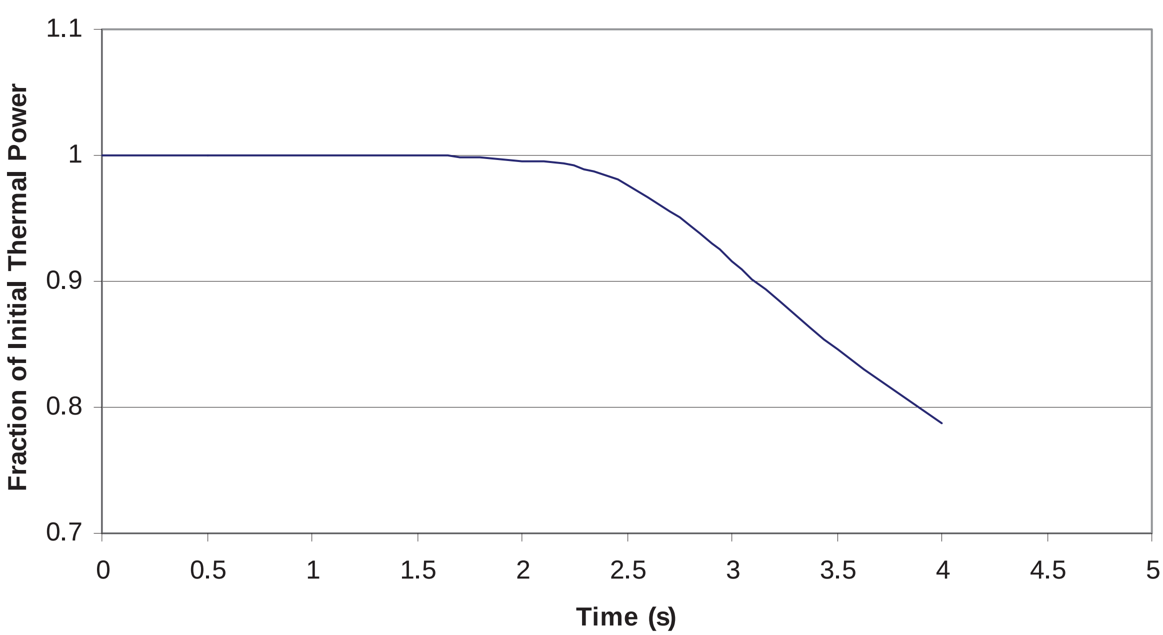


Figure 2.8.5.3.1-2: CR-3 EPU 4 Pump Coastdown Analysis



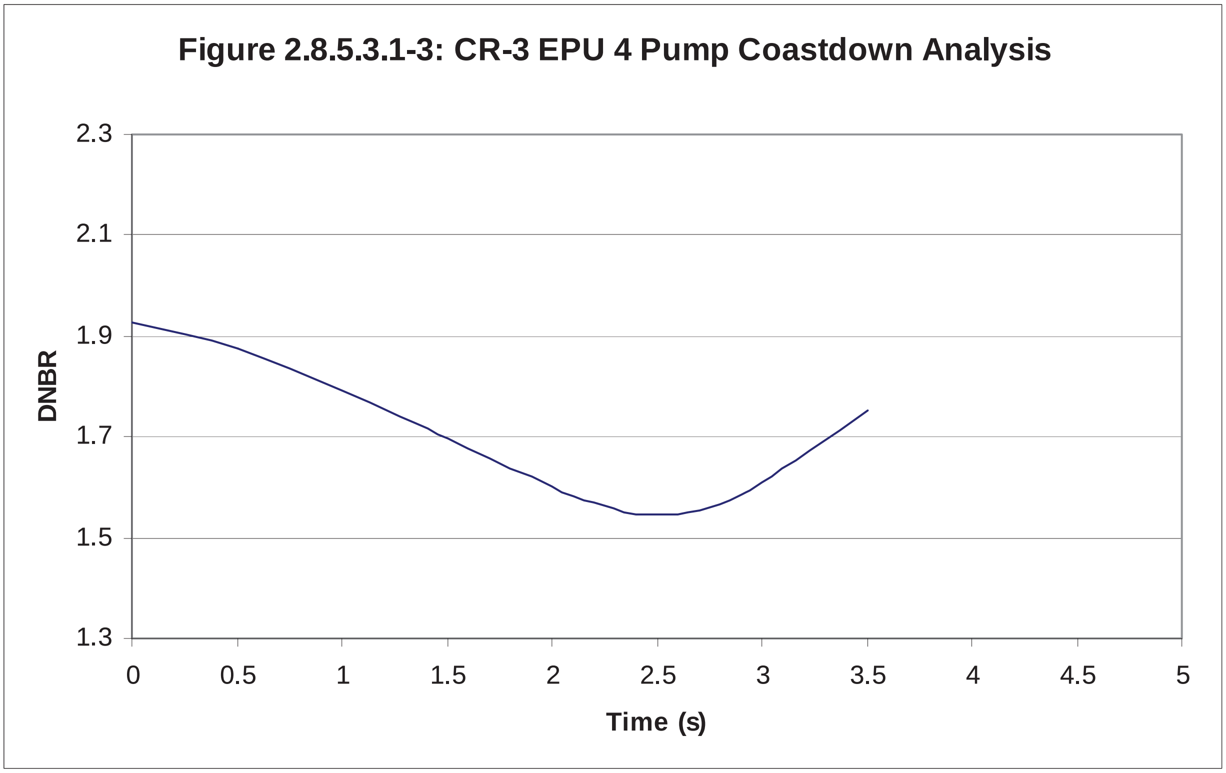


Figure 2.8.5.3.1-3: CR-3 EPU 4 Pump Coastdown Analysis
DNBR
2.3
2.1
1.9
1.7
1.5
1.3
0
0.5
1
1.5
2
2.5
3
3.5
4
4.5
5
Time (s)

Figure 2.8.5.3.1-4: CR-3 EPU 1 Pump Coastdown Analysis

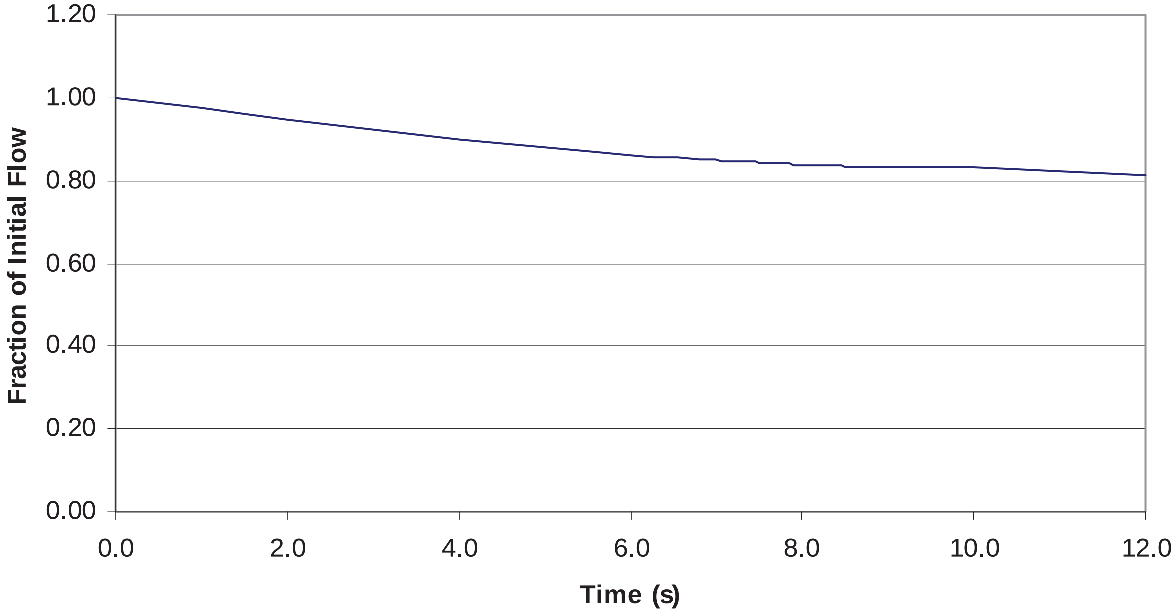


Figure 2.8.5.3.1-5: CR-3 EPU 1 Pump Coastdown Analysis

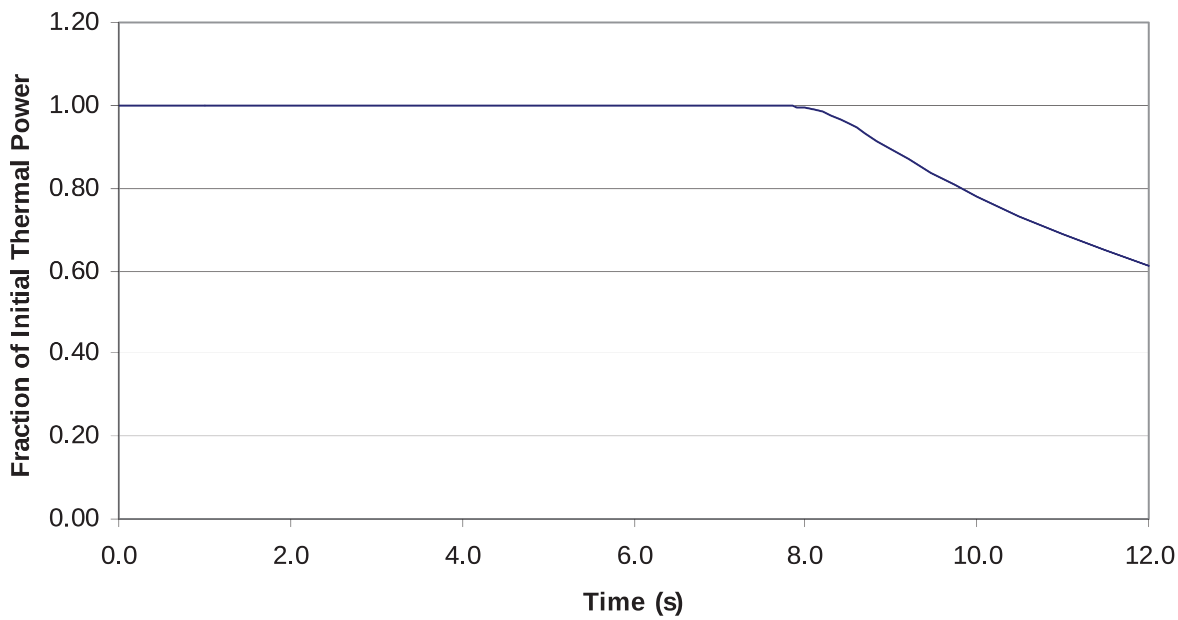
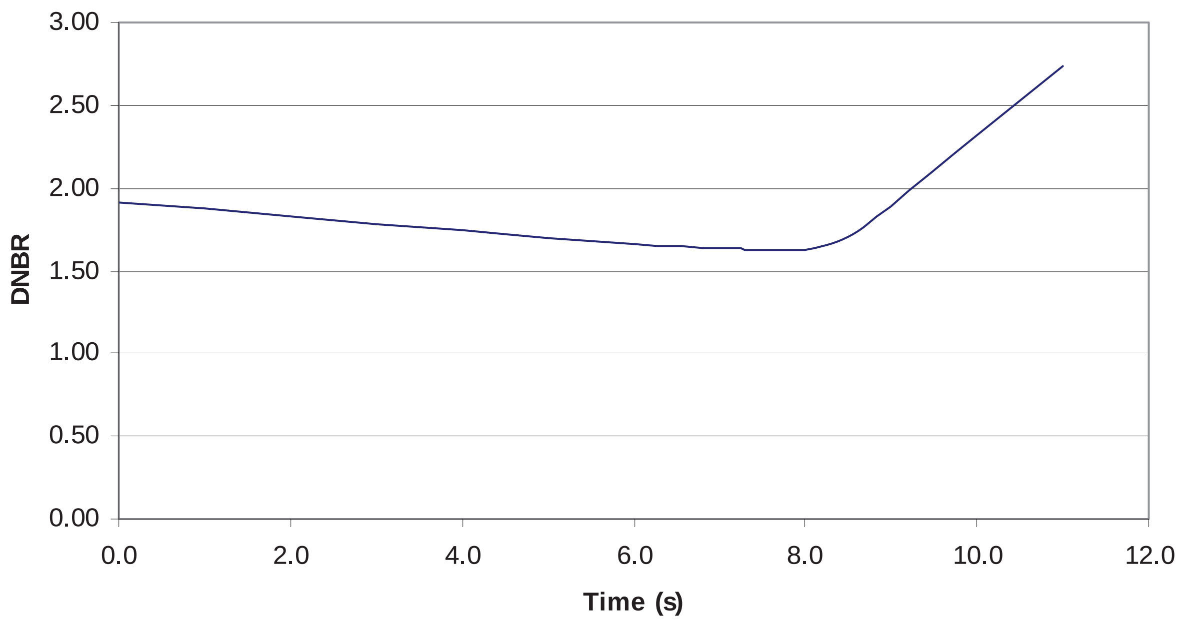


Figure 2.8.5.3.1-6: CR-3 EPU 1 Pump Coastdown Analysis



2.8.5.3.2 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

2.8.5.3.2.1 Regulatory Evaluation

The events postulated are an instantaneous seizure of the rotor or break of the shaft of a reactor coolant pump (RCP). Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer, which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The CR-3 review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of the reactor system components, (5) the functional and operational characteristics of Reactor Protection System, (6) operator actions, and (7) the results of the transient analyses.

The NRC's acceptance criteria for Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break are based on:

- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;

- GDC 28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and,

- GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

CR-3 Current Licensing Basis

As noted in CR-3 FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in CR-3 FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Sections 1.4.28, Reactivity Hot Shutdown Capability, 1.4.29, Reactivity Shutdown Capability, and 1.4.30, Reactivity Holddown Capability, insofar as these criteria require that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained [GDC-27];
- FSAR Section 1.4.32, Maximum Reactivity Worth of Control Rods, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core [GDC-28]; and,
- FSAR Sections 1.4.34, Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention, and 1.4.35, Reactor Coolant Pressure Boundary Brittle Fracture Prevention, insofar as these criteria require that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized [GDC-31].

2.8.5.3.2.2 Technical Evaluation

Introduction

As described in FSAR Section 14.1.2.6, a loss-of-coolant-flow (LOCF) accident is the loss of or reduction in forced flow through the Reactor Coolant System (RCS). The Loss of or Reduction in Forced Flow may be due to a mechanical failure in the reactor coolant pump (s) (RCP(s)) or a loss of electrical power to the RCP(s). A LOCF event results in a reduction in the heat removal capability of the reactor coolant. The event that results from seizure of an RCP or a shaft break is referred to as a locked rotor for CR-3.

The locked rotor event occurs when the rotor of a RCP seizes. When the rotor seizes, forced flow is no longer provided by the affected RCP. The locked rotor event results in a more rapid reduction in RCS flow than the four pump coastdown event. Since the results of each LOCF event are primarily controlled by the rate of reduction in RCS flow, the locked rotor event is the limiting LOCF event. The locked rotor transient results in a more severe reduction in flow initially, while an RCP shaft break would allow greater reverse flow, minimizing the flow through the active fuel. The analyzed transient for CR-3 bounds both scenarios by modeling an immediate flow reduction, and also allowing unrestricted reverse flow (i.e., no additional resistance due to locked rotor).

The locked rotor accident will not initiate a more serious accident. There is nothing in the design of the motor that could possibly cause the rotating elements to come to an instantaneous stop, thereby imposing the initial forces into the piping and resistance of the RCS.

Reactor protection for the locked rotor event is provided by the flux/flow trip that is part of the Nuclear Overpower RCS Flow and Measured Axial Power Imbalance Reactor Protection System trip function.

Input Parameters, Assumptions, and Acceptance Criteria

The LOCF analyses used the methodology defined in BAW-10193P-A (Reference 1) to provide the thermal-hydraulic forcing functions to the departure from nucleate (DNB) analyses. The RELAP5/MOD2-B&W code (Reference 2) was used in conjunction with the methodology, which also incorporated conservative setpoints and capacities to arrive at a conservative result. The key input parameters and initial conditions used in the thermal-hydraulic analysis of the LOCF events are as follows:

- The initial core power level was set to 3026.1 MWt, which is equivalent to the proposed EPU power level plus heat balance uncertainty. Additionally, a conservative RCP heat of 16.4 MWt was also included.
- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent with the planned increase in T_{AVG} planned in conjunction with the EPU.
- The RCS initial pressure is the nominal value of 2170 psia in the reactor coolant hot leg.
- An initial RCS flow equal to the minimum thermal design flow rate was used (374,880 gpm).
- The initial pressurizer level modeled was the nominal value plus uncertainty (240 inches) for the locked rotor event.
- Pressurizer heaters were not modeled for the locked rotor analyses since a lower pressure is conservative with respect to DNB calculations.
- Pressurizer spray (190 gpm) was modeled since pressurizer spray would keep the pressure lower, which is conservative for DNB calculations.
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- Reactor trip is initiated by the Nuclear Overpower RCS Flow and Measured Axial Power Imbalance (Power/Imbalance/Flow or PIF) trip function, specifically the flux-to-flow portion of the trip. The flux-to-flow trip occurs when the ratio of the percent full power determined by the measured neutron flux divided by the percent flow exceeds the setpoint. A flux-to-flow setpoint for CR-3 of 1.13 %full power/%flow was used, which accounts for instrument uncertainties in order to protect the allowable values contained in the core operating limits report. An appropriate response time is modeled prior to control rod insertion.
- A Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) and moderator temperature coefficient (0.0 Δk/k/°F), typical of beginning-of-cycle conditions, were used since they yield the maximum rate of power increase.
- Steam generator tube plugging of 5% was modeled. Consideration of tube plugging is conservative due to the increased resistance to flow within the RCS, and 5% tube plugging bounds actual tube plugging amounts.
- Offsite power was available in accordance with the licensing basis for the event.

- No single failure was assumed in accordance with the licensing basis for the event. There are no credible single failures that could result in more limiting results.

- No operator actions were credited due to the short timeframe of the event.

The CR-3 acceptance criteria for the event are related to maintaining DNB ratio (DBNR) above the thermal design limit for the fuel. For the locked rotor event, due to the event classification (Condition IV Event - in accordance with the definitions provided in Section 2.8.5.0, Non-LOCA Analysis Introduction), fuel failure is allowed, provided offsite dose limits of 10CFR50.67 are not exceeded. The possibility for fuel failure has not been considered previously and was not required since DNBR results for the locked rotor event have demonstrated that there are no fuel failures. However, for the EPU locked rotor event, the minimum DNBR calculated is not sufficiently high to preclude DNB in the event of locked rotor transient. The radiological consequences of the event are examined in Section 2.9.2, Radiological Consequence Analyses. The analysis is described further in the Results section.

Description of Analyses and Evaluations

The locked rotor event occurs when the rotor of a RCP seizes, and forced flow immediately ceases in the affected loop. The reduction in the RCS flow, coupled with relatively constant reactor power causes the DNBR to decrease. The DNBR reaches a minimum near the time of control rod insertion due to the reactor trip on flux-to-flow ratio. The thermal-hydraulic analysis was performed using RELAP5/MOD2-B&W. Upon event initiation, the affected RCP is stopped instantaneously. As a result, RCS flow rate decreases with the core power level remaining relatively constant. The flux-to-flow trip setpoint is reached 0.36 seconds after event initiation, and reverse flow in the affected loop is established after 0.5 seconds. The control rods begin to insert 2.55 seconds after event initiation. The thermal-hydraulic event is continued for a short period of time, but the DNB transient is mitigated (minimum DNBR reached) upon control rod insertion.

The locked rotor DNB analysis is performed by inputting the transient flow and power curves calculated by RELAP5/MOD2-B&W into a LYNXT model based on the steady state DNB analysis for four pump operations. Per BAW-10179 (Reference 3) the power curve is adjusted to discount the favorable effects of the moderator temperature coefficient, which would cause a slight decrease in power prior to the control rods entering the core due to the increased coolant temperature. The core exit pressure is conservatively held constant at its initial value throughout the event. The core inlet temperature is also held constant at its initial value. Due to the short duration of the transient, cold leg temperatures do not vary during the event. The thermal-hydraulic analysis results support this assumption. The model is initiated from 100% core power and uses design radial and axial peaking. The heat balance error is not included in the initial core power, since it is included within the statistical core design methodology for calculating DNBR. The analysis calculates the DNBR at various time steps within the transient, with greater resolution in the period between the beginning of the flow coastdown and the decrease in power due to control rod insertion.

Results

The transient results are provided in Table 2.8.5.3.2-1, and Figures 2.8.5.3.2-1 through 2.8.5.3.2-3. For the locked rotor event, the minimum DNBR was calculated to be 1.40 at 2.60 seconds after transient initiation. The locked rotor event analysis calculates a minimum DNBR which is not sufficiently high to preclude DNB in the event of that transient. As mentioned previously, the radiological consequences of the event are examined in Section 2.9.2. The locked rotor results are acceptable based on the results of the dose evaluation.

In addition, it was determined that a set of locked rotor maximum allowable peaking (MAP) limits can be employed which protect the core from DNB. These locked rotor MAP limits do not significantly impact the operating limits of the plant, as other analyses are setting the operating limits for CR-3. Therefore, the locked rotor MAP limits are used by the maneuvering analysis to protect the core and no failed fuel is expected in the CR-3 core during a locked rotor transient.

The CR-3 FSAR does not include an acceptance criterion related to RCS pressure. The results of the analyzed event demonstrated that maximum RCS pressure remains below the pressurizer code safety valve setpoints, thus RCPB limits were not challenged. The assumptions of the thermal-hydraulic analysis were not geared toward maximizing RCS pressure response, however, the challenge to the RCPB would be greater for the Startup Accident, as described in Section 2.8.5.4.1, Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition. The locked rotor event does not challenge RCPB limits.

Another consideration for the event is that the RCPB be designed with sufficient margin to assure that, under specified conditions, it will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. The licensing basis for the event does not include acceptance criteria to address this aspect of the event. However, as stated above the maximum RCS pressure does not challenge the RCPB. Also, since the event is an overheating event, Reactor Coolant System temperatures do not approach values that would be of concern for brittle fracture.

2.8.5.3.2.3 Conclusion

CR-3 has reviewed the analyses of the sudden decrease in core coolant flow events and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the evaluation has demonstrated that the reactor protection and safety systems will continue to ensure that the ability to insert control rods is maintained, the RCPB pressure limits will not be exceeded, the RCPB will behave in a non-brittle manner, the probability of propagating fracture of the RCPB is minimized, and adequate core cooling will be provided. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.28, 1.4.29, 1.4.30, 1.4.32, 1.4.34, and 1.4.35 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.3.2.4 References

1. BAW-10193PA-00 (Proprietary), "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors."

2. BAW-10164PA-06 (Proprietary), "RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

3. BAW-10179PA-007 (Proprietary), "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses."

Table 2.8.5.3.2-1: Sequence of Events for Locked Rotor Event

Event	Time (s)
Event Initiated	0.0
RPS Trip Signal Received (Flux/Flow > 1.13)	0.36
Reverse Flow Begins in Affected Loop	0.50
Control Rods Begin to Insert	2.55
Minimum DNBR	2.60
Transient Ends	5.00

Figure 2.8.5.3.2-1: CR-3 EPU Locked Rotor Analysis

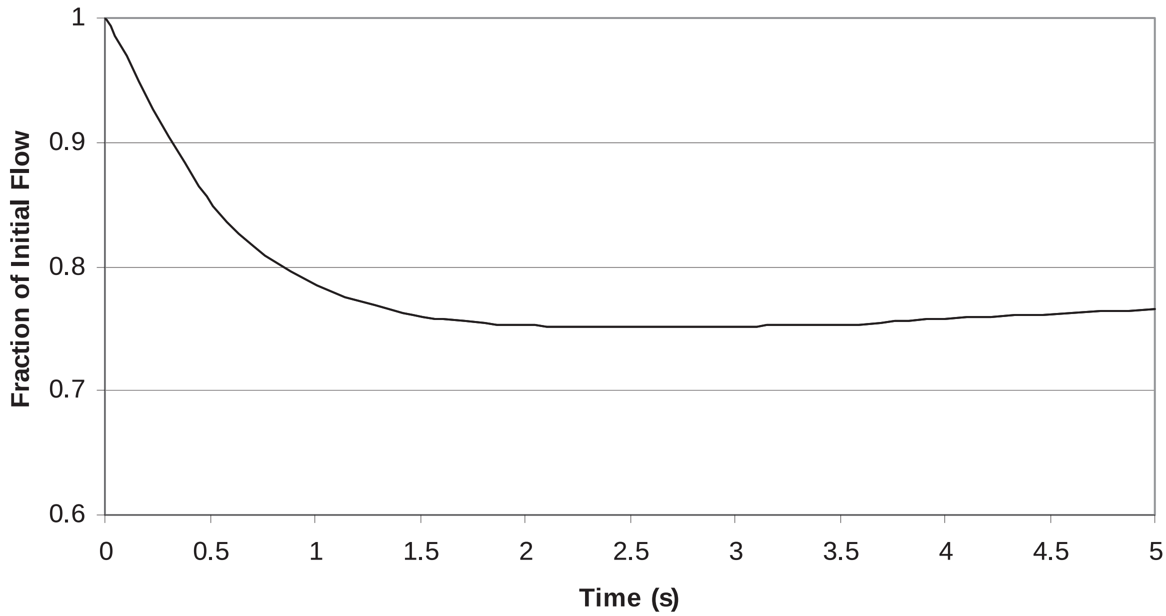


Figure 2.8.5.3.2-2: CR-3 EPU Locked Rotor Analysis

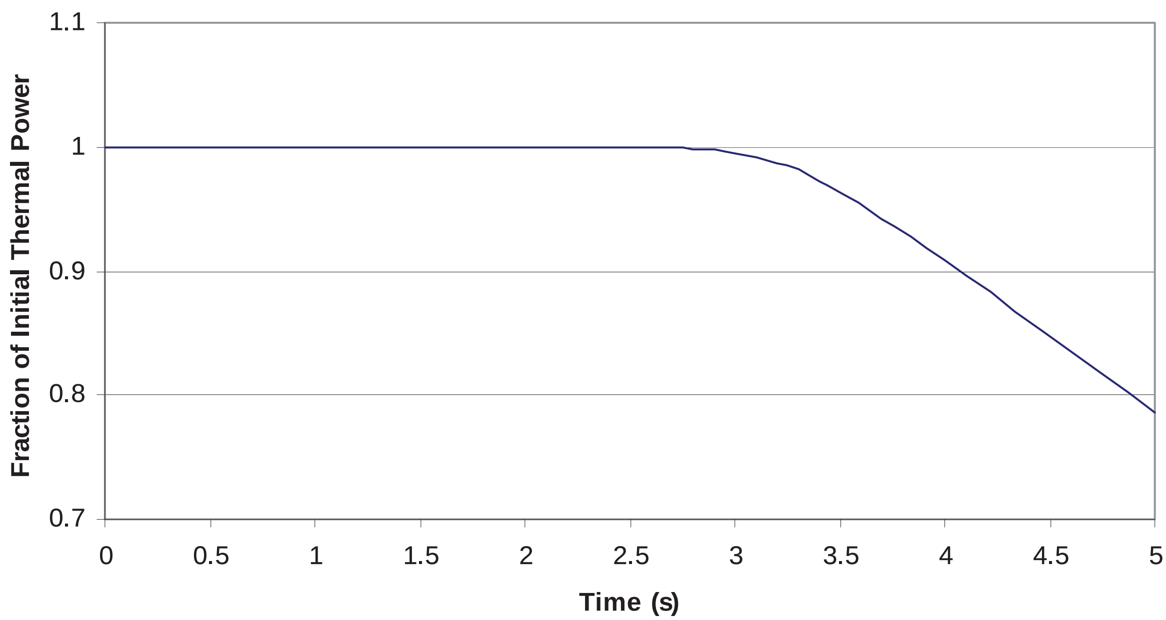
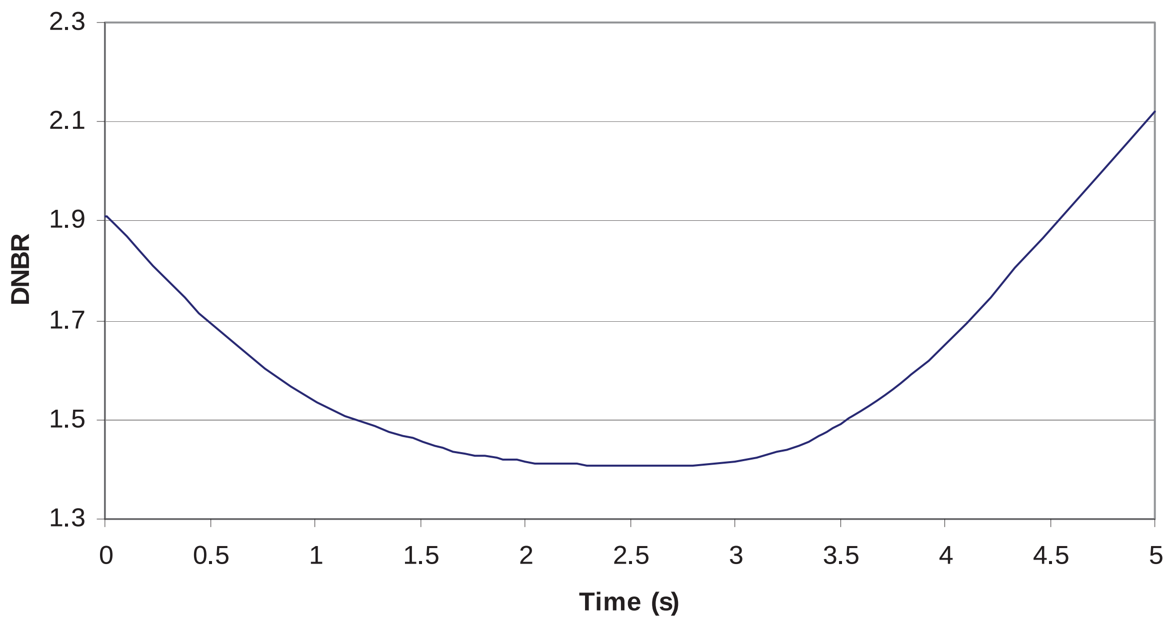


Figure 2.8.5.3.2-3: CR-3 EPU Locked Rotor Analysis



2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

2.8.5.4.1.1 Regulatory Evaluation

An Uncontrolled Control Rod Assembly Withdrawal From Subcritical or Low Power Startup Condition may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The CR-3 review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the transient analyses.

The NRC's acceptance criteria for the Uncontrolled Rod Withdrawal From a Subcritical or Low Power Startup Condition are based on

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including anticipated operational occurrences (AOOs),
- GDC-20, insofar as it requires that the Reactor Protection System (RPS) be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs, and
- GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; [GDC-10]
- FSAR Sections 1.4.14, Core Protection Systems, insofar as it requires that the reactor protection system be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and [GDC-20]
- FSAR Section 1.4.31, Reactivity Control Systems Malfunction, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. [GDC-25]

2.8.5.4.1.2 Technical Evaluation

Introduction

During a typical startup, reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level by Control Rod Assembly (CRA) withdrawal and by reducing the core boron concentration. CRA motion results in much faster changes in reactivity than can result from changing boron concentration.

The rods are physically prevented from withdrawing in other than their respective banks. During startup the safety rod groups are withdrawn first, enabling withdrawal of the regulating control groups. The sequence allows operation of only one regulating rod group at a time except where reactivity insertion rates are low (first and last 20%-30% of stroke), at which time two adjacent groups are operated simultaneously in overlapped fashion. The motor, lead screw, and power supply are designed to provide a uniform rate of speed for rod travel. The reactivity change is dependent upon rod group size. A range of reactivity insertion rates are analyzed in the detailed plant analysis.

During the Startup Accident (FSAR Section 14.1.2.2), the neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the initial power increase results from a fast negative fuel temperature feedback (Doppler effect) and is of prime importance during a startup accident since it limits the power to an acceptable level prior to protection system action. The startup accident is terminated by either the Nuclear Overpower or RCS High Pressure trip functions. Due to the additional protection provided at the lower modes, the at-power (i.e., hot zero power (HZP)) events are more limiting.

Description of Analyses and Evaluations

The startup accident was analyzed for the EPU using RELAP5/MOD2-B&W. The RELAP5/MOD2-B&W code has been approved by the NRC for use in non-loss of coolant accident (LOCA) transient analyses (Reference 1). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The Reactor Coolant System model provides for heat transfer from the core, transport of the coolant to the once-through steam generators (OTSGs), and heat transfer to the OTSGs. The secondary model includes a detailed depiction of the Main Steam System, including steam relief to the atmosphere through the main steam safety valves and simulation of the turbine stop valves. The secondary model also includes the delivery of feedwater, both main and emergency, to the OTSGs. The analyses were performed in compliance with methodology that has been approved by the NRC for B&W plant non-LOCA transient analysis (Reference 2).

The startup accident is initiated at a very low power level to maximize the power excursion required to reach the nuclear overpower reactor trip setpoint. In order to analyze the startup accident, a range of reactivity insertion rates are considered. The control rod travel and speed are fixed, resulting in a constant reactivity insertion rate (RIR) for a given control rod worth being withdrawn. The analysis considers a range of RIRs that exceeds the maximum RIRs predicted for the EPU cycle. A lower RIR causes neutron flux to increase slowly, which results in increases in core power and RCS pressure that remain closely coupled with reactor trip occurring on high RC pressure. As the RIR is increased, the neutron flux response is the more dominant effect, and at some reactivity insertion rate, reactor trip will occur on nuclear overpower, and thereafter the RCS response is less limiting. The RELAP5/MOD2-B&W analysis of the startup accident models a spectrum of RIR and determines the results in terms of RCS pressure and core thermal power response. The startup accident is typically the limiting event in terms of RCS overpressure, and can place limitations on the core reload design (to limit the maximum RIR).

The key input parameters and initial conditions used in the analysis of the startup accident are as follows:

- The initial Reactor Coolant System pressure is assumed to be the nominal value of 2169.7 psia at the hot leg pressure tap.
- The initial pressurizer level (240 inches indicated) is conservatively set to the nominal full power level, and is increased to account for measurement uncertainty.
- A least-negative Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) typical of beginning-of-cycle conditions is used since it yields the maximum rate of power increase. The startup accident will result in increased fuel temperature, and so a least-negative Doppler coefficient minimizes the negative feedback as the fuel temperature increases.
- A most-positive moderator temperature coefficient (+0.75 Δk/k/°F) is used since this yields the maximum rate of power increase. The use of the fixed value is conservative, since as the transient progresses, the power increase is such that the moderator temperature coefficient (MTC) would be less positive than assumed. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the nuclear flux response time constant. The most-positive MTC is consistent with the proposed HZP technical specification value for the EPU (see Attachment 2, Operating License and Technical Specification Changes (Markup)). This change is being made in conjunction with the EPU. As shown Table 2.8.2-2, the proposed HZP MTC value bounds predicted the EPU fuel cycles and thus is acceptable.
- The analysis assumes the reactor to be at hot zero power conditions with a nominal hot zero power temperature of 532°F. Small variations in the initial temperature are not significant to the event results.
- The reactor is assumed to be critical at the time the transient is initiated.
- A range of reactivity insertion rates is considered, from that associated with a single rod to an RIR that exceeds the maximum expected RIR due to withdrawal of an entire rod group.

- The analysis models reactor trip to occur on either high RCS pressure (2400 psia) or nuclear overpower (112% of full power). The setpoints modeled include allowances for instrument error and setpoint error and are bounded by the Improved Technical Specifications (ITS) RPS Allowable Values. Also, appropriate delays for signal processing and control rod assembly release are modeled. While both these trip functions were modeled, for all cases analyzed the reactor tripped on high RCS pressure.

- The maximum reactivity insertion rate analyzed is greater than achievable for the EPU conditions considering the sequencing of bank withdrawals. For the EPU, the maximum reactivity insertion rate due to rod withdrawal is predicted to be 17 pcm/second. The analysis considers RIRs that bound this value. Analysis results are reported for the bounding RIR of 20 pcm/second.

- The initial power level is set to a conservatively low value ($1x10^{-9}$% of EPU full power).

- The analysis is initiated at hot zero power, at which point all reactor coolant pumps (RCPs) are running. Minimum thermal design RCS flow rates are assumed in accordance with approved methodology (Reference 2).

The acceptance criteria for the event, as described in the FSAR Chapter 14, are as follows:

- Peak Reactor Coolant System pressure shall remain below 110% of the RCS design pressure, based upon ASME code.

- Reactor thermal power shall remain below 112% of the uprated power level. Since DNB and centerline fuel melt calculations assume a maximum power level of 112% of the uprated power level, meeting this criterion ensures departure from nucleate boiling (DNB) and centerline fuel melt (CFM) limits are preserved, and precludes the need for transient-specific DNB and CFM calculations for the startup accident.

Results

As the RIR was increased, the peak RCS pressure increased. The absolute power associated with the nuclear overpower trip setpoint increases due to the EPU. As a result, the nuclear overpower trip is less effective in mitigating the event and the RIRs modeled resulted in RCS pressure approaching the peak RCS pressure limit. For the EPU, the peak one-group rod withdrawal rate is 17 pcm/second. The analysis considered maximum RIRs that bound this value and results are reported for the case that models a RIR of 20.0 pcm/second, providing margin above the maximum expected RIR for the EPU. The sequence of events for this RIR is reported in Table 2.8.5.4.1-1 and in Figures 2.8.5.4.1-1 to 2.8.5.4.1-4. Figures 2.8.5.4.1-1 and 2.8.5.4.1-2 show total power and thermal power responses, respectively. The total power includes neutron and decay power, while the thermal power is indicative of heat transferred to the reactor coolant. The thermal power response lags the total power response because the neutron power increases quickly, but the transfer of heat to the coolant is slower because heat must be transferred from the fuel, across the gap and cladding, and then into the reactor coolant. The peak pressure that results from this RIR is 2746.3 psia, leaving margin to the acceptance criterion of 2764.7 psia. The total power does not exceed 112% of the EPU rated thermal power and, therefore all acceptance criteria for the event are met.

2.8.5.4.1.3 Conclusion

CR-3 has reviewed the analyses of the Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition and concludes that the analyses have adequately accounted for the changes in core design necessary for operation of the plant at the EPU power level. CR-3 also concludes that the analyses were performed using acceptable analytical models. CR-3 further concludes that the reactor protection and safety systems will continue to ensure the SAFDLs are not exceeded. Based on this, CR-3 concludes that the plant will continue to meet the requirements of CR-3 FSAR Sections 1.4.6, 1.4.14, and 1.4.31 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Uncontrolled Control Rod Assembly Withdrawal From a Subcritical or Low Power Startup Condition.

2.8.5.4.1.4 References

1. BAW-10164PA-06 (Proprietary), “RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis.”

2. BAW-10193PA-00 (Proprietary), “RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors.”

Table 2.8.5.4.1-1: Sequence of Events for Startup Accident (RIR = 20.0 pcm/second)

Parameter	Time, sec
Control rod withdrawal initiated	0.0
High flux trip actuated	N/A
High RCS pressure trip actuated	36.087
Control Rods begin to fall,	36.698
Peak Thermal Power occurs	37.559
Pressurizer Safety Valve initial lift	~39.1
Peak RCS pressure occurs	39.342
Transient terminated	60.0

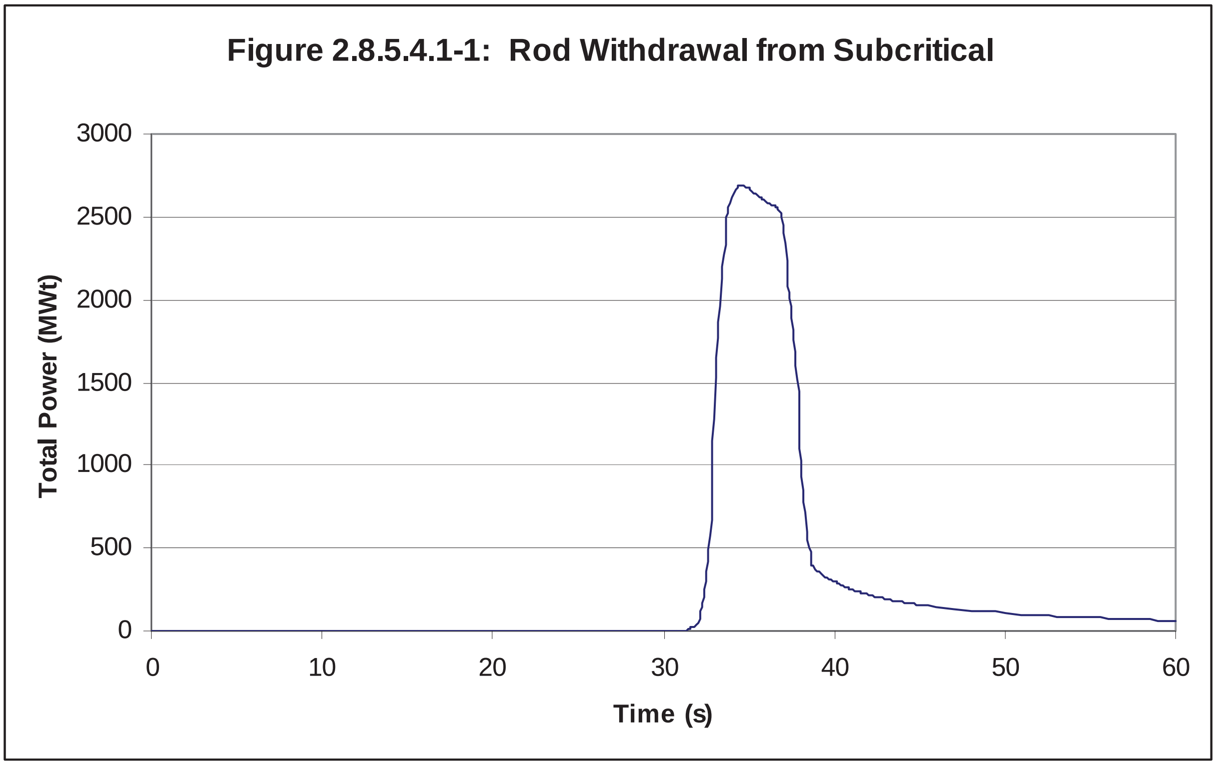


Figure 2.8.5.4.1-1: Rod Withdrawal from Subcritical
Total Power (MWt)
3000
2500
2000
1500
1000
500
0
0
10
20
30
40
50
60
Time (s)

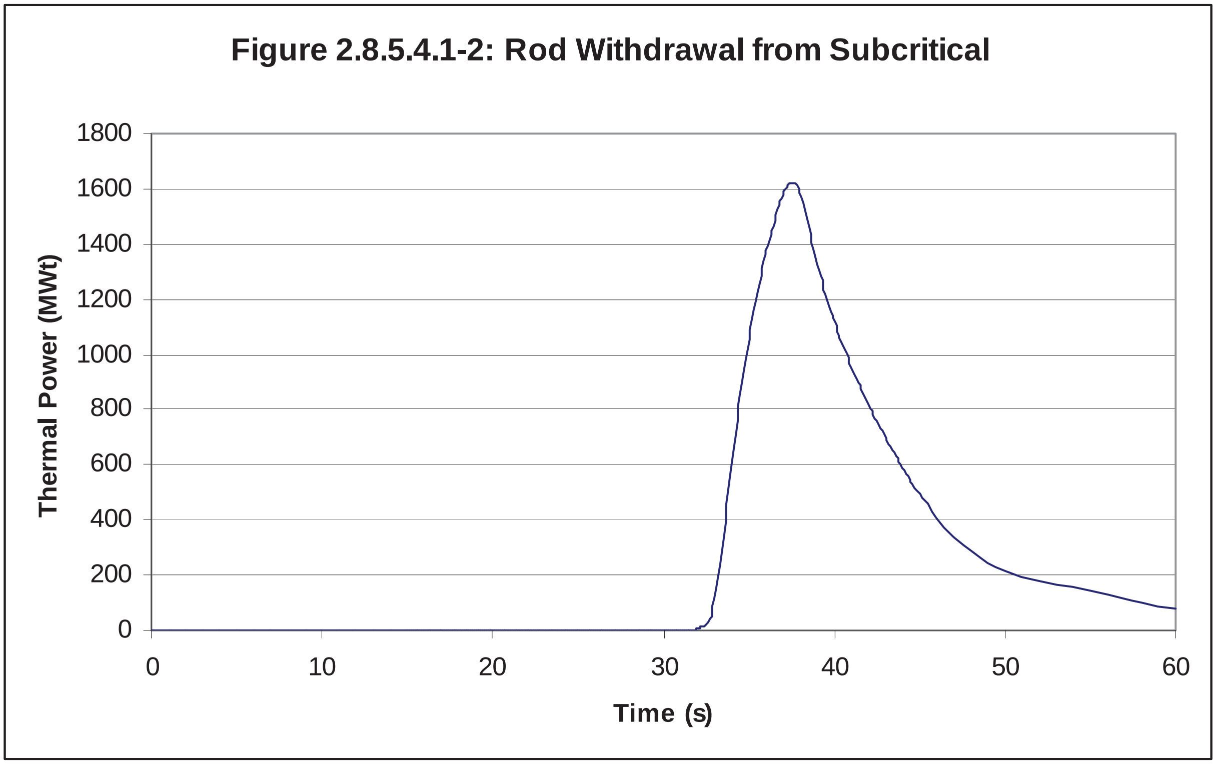


Figure 2.8.5.4.1-2: Rod Withdrawal from Subcritical
Thermal Power (MWt)
1800
1600
1400
1200
1000
800
600
400
200
0
0
10
20
30
40
50
60
Time (s)

Figure 2.8.5.4.1-3: Rod Withdrawal from Subcritical

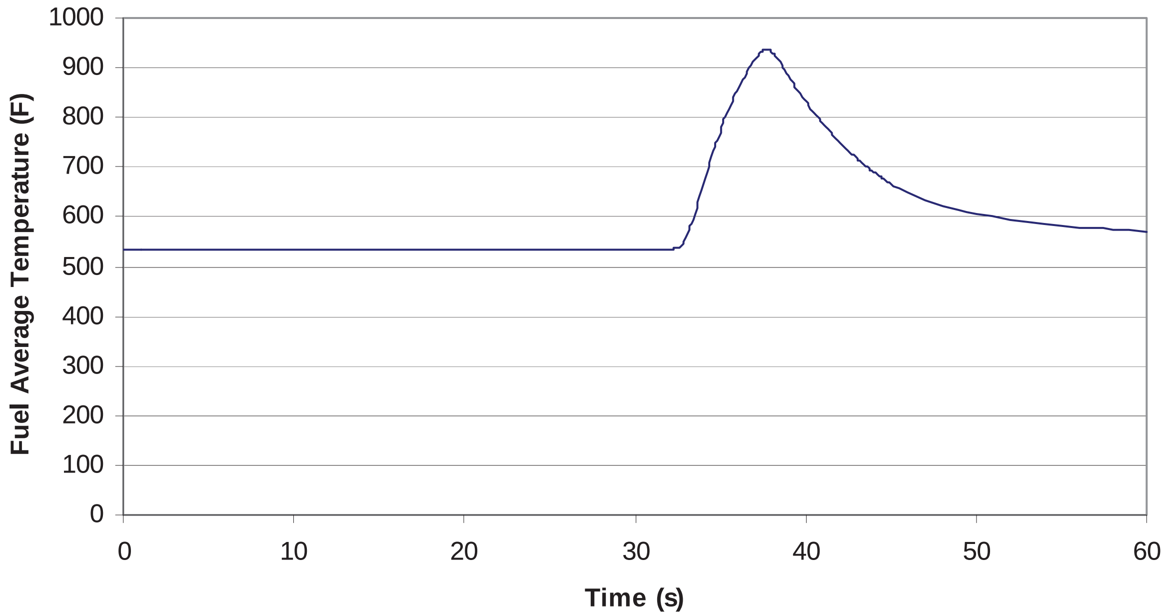
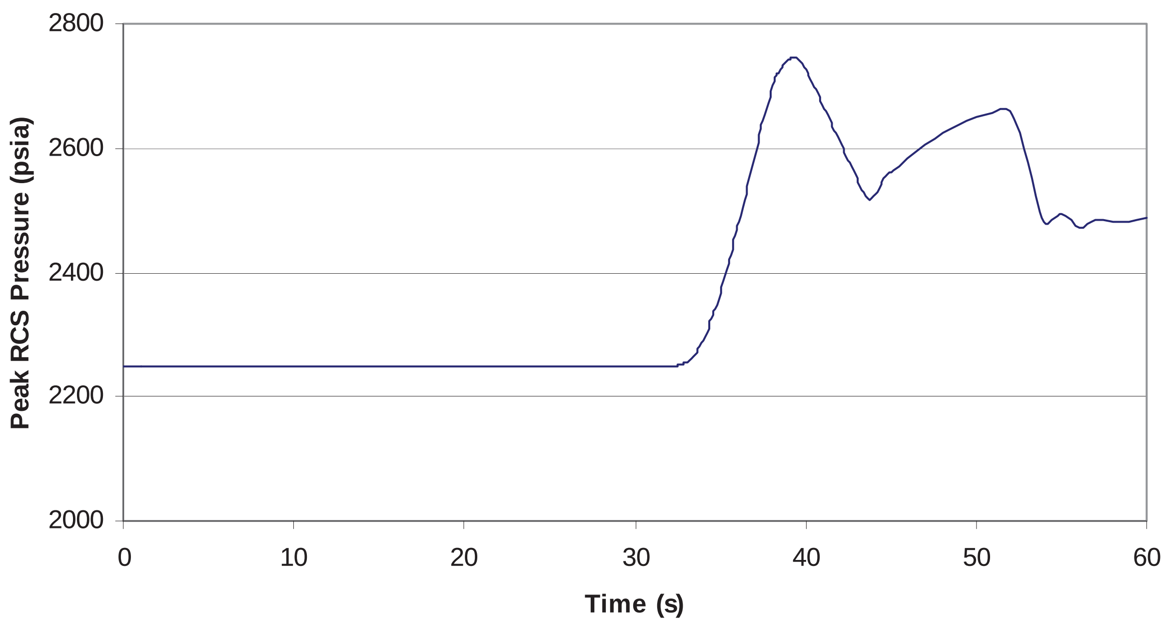


Figure 2.8.5.4.1-4: Rod Withdrawal from Subcritical



2.8.5.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

2.8.5.4.2.1 Regulatory Evaluation

An Uncontrolled Rod Control Assembly Withdrawal at Power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The CR-3 review covered (1) the description of the causes of the anticipated operational occurrence (AOO) and the description of the event itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses.

The NRC's acceptance criteria for an Uncontrolled Rod Control Assembly Withdrawal at Power are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations, including AOOs;
- GDC-20, insofar as it requires that the Reactor Protection System (RPS) be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and
- GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs; [GDC-10]
- FSAR Sections 1.4.14, Core Protection Systems, and 1.4.15, Engineered Safety Features Protection Systems, insofar as it requires that the Reactor Protection System be designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded as a result of AOOs; and [GDC-20]
- FSAR Section 1.4.31, Reactivity Control Systems Malfunction, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. [GDC-25]

2.8.5.4.2.2 Technical Evaluation

Introduction

An Uncontrolled Control Rod Assembly (CRA) Withdrawal at Power is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of CRAs resulting in a power excursion. Such a transient could be caused by a malfunction of the Control Rod Drive Control System. The uncontrolled CRA withdrawal at power is classified as an ANS Condition II event of moderate frequency, per the event classifications presented in Section 2.8.5.0, Non-LOCA Analysis.

The rods are physically prevented from withdrawing in other than their respective banks. At power the safety rod groups would be fully withdrawn, and two of the three regulating control rod groups would be withdrawn. The motor, leadscrew, and power supply designs are designed to provide a uniform rate of speed for rod travel. A range of reactivity insertion rates are analyzed in the detailed plant analysis.

During the CRA withdrawal at power (FSAR Section 14.1.2.3), the neutron flux response to a continuous reactivity insertion is characterized by a very fast flux increase terminated by the reactivity feedback effect of the negative Doppler coefficient. This self-limitation of the initial power increase results from a fast negative fuel temperature feedback (Doppler Effect) and is of prime importance during a rod withdrawal transient since it limits the power to an acceptable level prior to protection system action. The CRA withdrawal at power is terminated by either the Nuclear Overpower or RCS High Pressure trip functions.

Input Parameters, Assumptions, and Acceptance Criteria

The CRA withdrawal at power is initiated at the full nominal power level, plus heat balance uncertainty. In order to analyze the CRA withdrawal at power, a range of reactivity insertion rates are considered. The control rod travel and speed are fixed, resulting in a constant reactivity insertion rate (RIR) for a given control rod worth being withdrawn. The analysis considers a range of RIRs that exceed the maximum RIRs predicted for the EPU cycle. A lower RIR causes neutron flux to increase slowly, which results in increases in core power and RCS pressure that remain closely coupled with reactor trip occurring on high RCS pressure. As the RIR is increased, the neutron flux response is the more dominant effect, and at some reactivity insertion rate, reactor trip will occur on nuclear overpower, and thereafter the RCS response is less limiting. The RELAP5/MOD2-B&W analysis of the CRA withdrawal at power models a spectrum of RIRs and determines the results in terms of RCS pressure and core thermal power response.

The key input parameters and initial conditions used in the analysis of the CRA withdrawal at power are as follows:

- The initial power level is assumed to be 3026.1 MWt, which includes the nominal EPU power level of 3014 MWt, plus heat balance uncertainty. Additionally, total heat input from the reactor coolant pumps of 16.4 MWt is included.
- The initial RCS pressure is assumed to be the nominal value of 2169.7 psia at the hot leg pressure tap.
- The initial pressurizer level (240 inches indicated) is conservatively set to the nominal full power level, and is increased to account for measurement uncertainty.

- The pressurizer code safety valves are modeled to lift at the nominal setting, plus setpoint tolerance of 3%. The total relief capacity of the valves is 635,946 lbm/hr at 2764.7 psia.

- A least-negative Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) typical of beginning-of-cycle conditions is used since it yields the maximum rate of power increase. The CRA withdrawal accident will result in increased fuel temperature, and so a least-negative Doppler coefficient minimizes the negative feedback as the fuel temperature increases.

- A most-positive hot full power (HFP) moderator temperature coefficient of 0.0 Δk/k/°F (i.e., zero) is used since this yields the maximum rate of power increase. The use of the fixed value is conservative, since as the transient progresses, the power increase is such that the moderator temperature coefficient (MTC) would be less positive than assumed. The contribution of the moderator temperature coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and moderator is much longer than the nuclear flux response time constant.

- The analysis assumes the reactor to be at HFP conditions with a nominal HFP average temperature of 582°F. The choice of T_{AVG} is consistent with the planned increase in conjunction with the EPU.

- A range of reactivity insertion rates is considered which bounds the maximum expected RIR due to rod withdrawal at power.

- The analysis models reactor trip to occur on either high RCS pressure (2400 psia) or nuclear overpower (112% of full power). The setpoints modeled include allowances for instrument error and setpoint error and are bounded by the Improved Technical Specification RPS Allowable Values. Also, appropriate delays for signal processing and control rod assembly release are modeled.

- Once an RPS setpoint is reached, reactor trip is modeled with tripped rod worth based on a minimum shutdown margin of 1.0 %Δk/k, which is conservatively less than the Modes 1 and 2 shutdown margin of 1.3 %Δk/k for the EPU.

- Minimum thermal design RCS flow rates are assumed in accordance with approved methodology (Reference 1).

The acceptance criteria for the event, as described in the FSAR Chapter 14, are as follows:

- Peak RCS pressure shall remain below 110% of the RCS design pressure, based upon ASME code.

- Reactor thermal power shall remain below 112% of the uprated power level. Since departure from nucleate boiling (DNB) and centerline fuel melt calculations assume a maximum power level of 112% of the uprated power level, meeting this criterion demonstrates DNB and centerline fuel melt (CFM) limits are preserved, and precludes the need for transient-specific DNB and CFM calculations for the CRA withdrawal at power accident.

Description of Analyses and Evaluations

The Uncontrolled CRA Withdrawal at Power accident was analyzed for the EPU using RELAP5/MOD2-B&W. The RELAP5/MOD2-B&W code has been approved by the NRC for use in non-loss of coolant accident (LOCA) transient analyses (Reference 2). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The RCS model provides for heat transfer from the core, transport of the coolant to the once-through steam generators (OTSGs), and heat transfer to the OTSGs. The secondary model includes a detailed depiction of the main steam system, including steam relief to the atmosphere through the main steam safety valves and simulation of the turbine stop valves. The secondary model also includes the delivery of feedwater, both main and emergency, to the OTSGs. The analyses were performed in compliance with methodology that has been approved by the NRC for B&W plant non-LOCA transient analysis (Reference 1).

A spectrum of RIRs was considered, from that associated with withdrawal of a single control rod assembly up to an RIR that is equivalent to withdrawing the worth of all rod groups simultaneously.

Results

For all RIRs analyzed, the calculated peak RCS pressures and peak thermal power levels remained within the acceptance criteria limits of 2764.7 psia and 112%, respectively. The highest predicted peak RCS pressure was 2673.95 psia, as measured in the lower reactor vessel where RCS pressure is greatest. The highest predicted peak thermal power was 110.140%. The limiting RIRs resulted in near simultaneous initiations of the RPS high RCS pressure and high nuclear power trips. Increased RIRs result in rapid RPS initiations on nuclear overpower and make the results less limiting. The sequence of events is shown in Table 2.8.5.4.2-1 for a RIR that produces limiting results. The transient responses of various plant parameters are shown in Figures 2.8.5.4.2-1 through 2.8.5.4.2-6 for the same RIR of 4.62 pcm/s.

2.8.5.4.2.3 Conclusion

CR-3 has reviewed the analyses of the Uncontrolled Control Rod Assembly Withdrawal at Power event and concludes that the analyses have adequately accounted for the changes in core design necessary for plant operation at the proposed power level. CR-3 also concludes that the analyses were performed using acceptable analytical models. CR-3 further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the SAFDLs are not exceeded. Based on this, CR-3 concludes that the plant will continue to meet the requirements in FSAR Sections 1.4.6, 1.4.14, 1.4.15, and 1.4.31 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Uncontrolled Control Rod Assembly Withdrawal at Power.

2.8.5.4.2.4 References

1. BAW-10193PA-00 (Proprietary), "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors."

2. BAW-10164PA-06 (Proprietary), "RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

Table 2.8.5.4.2-1: Sequence of Events for Rod Withdrawal at Power (RIR = 4.62 pcm/s)

Parameter	Time, sec
Control Rod Withdrawal initiated	0.0
High RCS pressure trip actuated	34.254
High flux trip actuated	34.467
Control rods begin to insert	34.865
TSV closure starts	34.866
Peak thermal power	34.971
MFW flow decreases to zero	37.87
Peak RCS Pressure	38.124
Initial PSV lift occurs	38.36
Final PSV closure occurs	40.16
Transient analysis ends	120.0

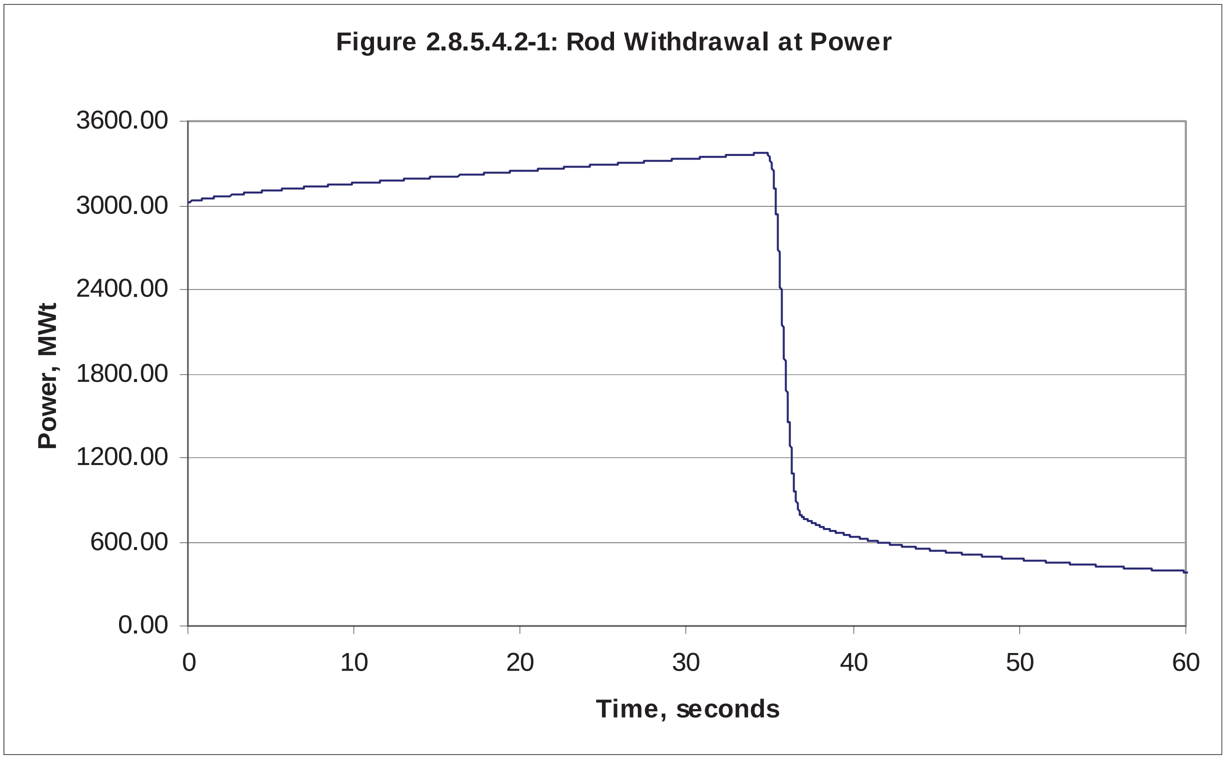


Figure 2.8.5.4.2-1: Rod Withdrawal at Power
3600.00
3000.00
2400.00
1800.00
1200.00
600.00
0.00
Power, MWt
0
10
20
30
40
50
60
Time, seconds

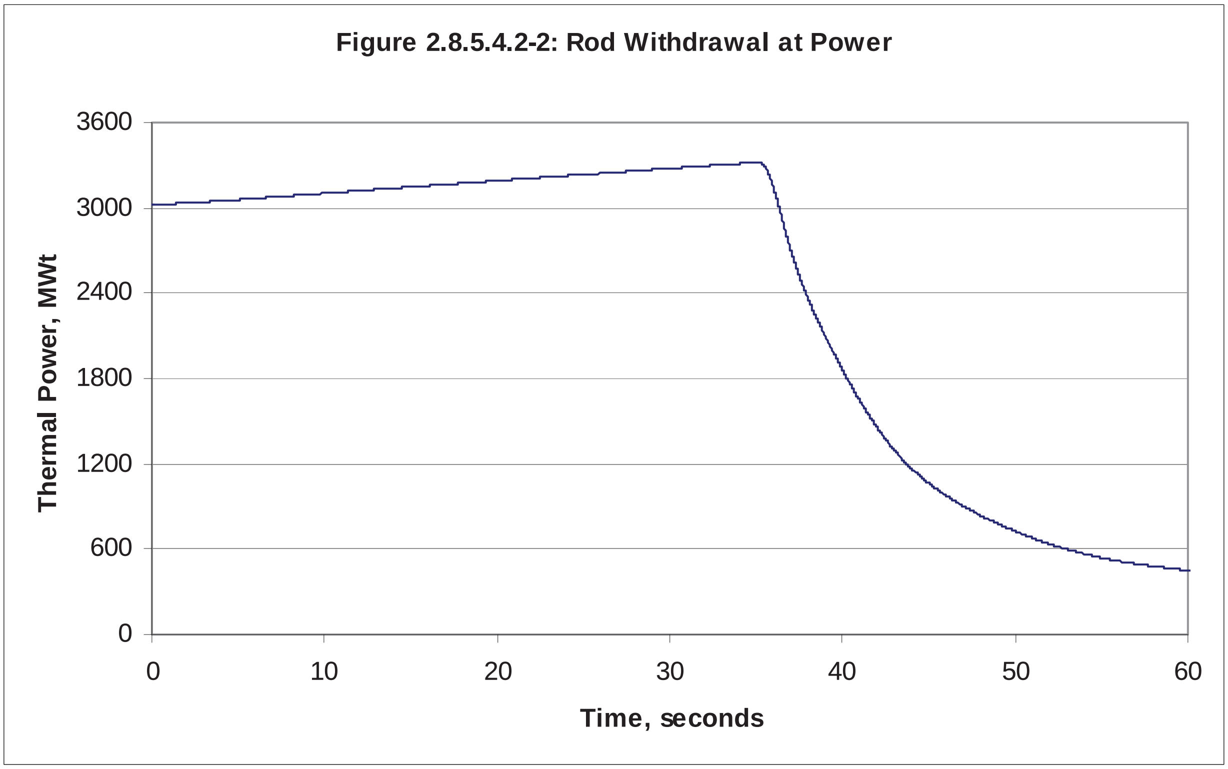


Figure 2.8.5.4.2-2: Rod Withdrawal at Power
3600
3000
2400
1800
1200
600
0
Thermal Power, MWt
0
10
20
30
40
50
60
Time, seconds

Figure 2.8.5.4.2-3: Rod Withdrawal at Power

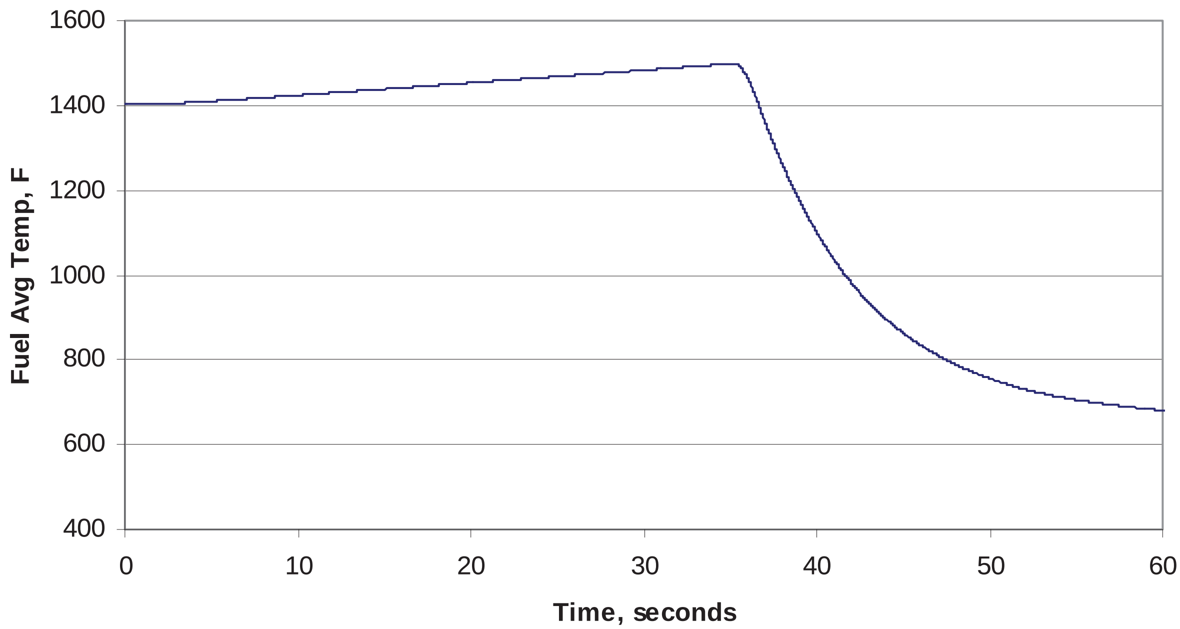


Figure 2.8.5.4.2-4: Rod Withdrawal at Power

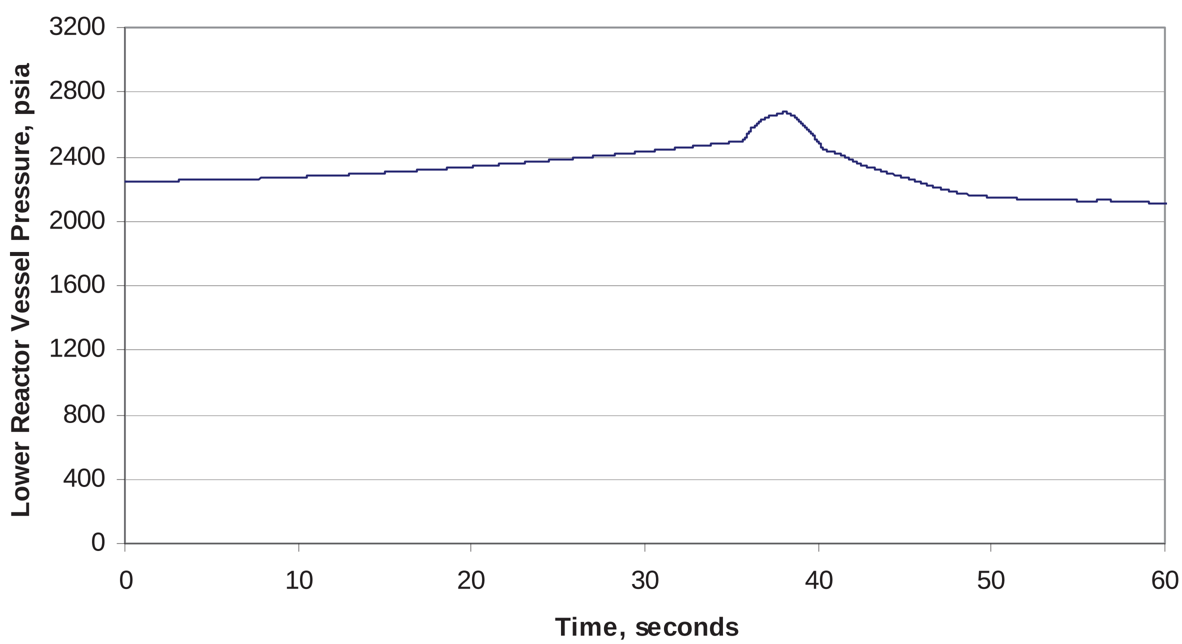


Figure 2.8.5.4.2-5: Rod Withdrawal at Power

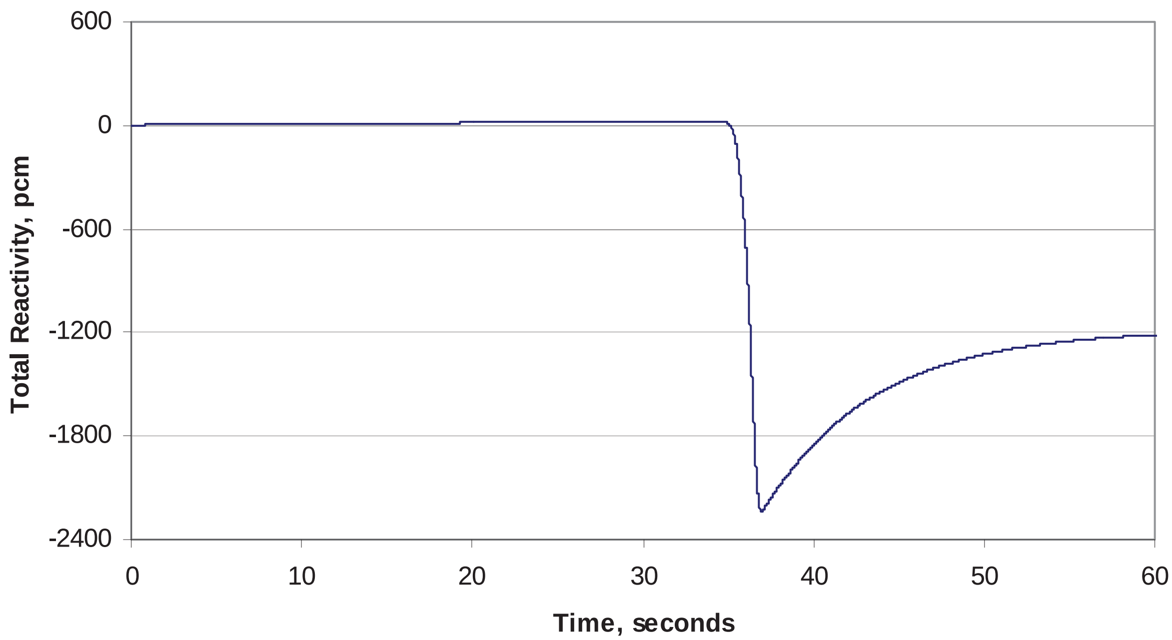
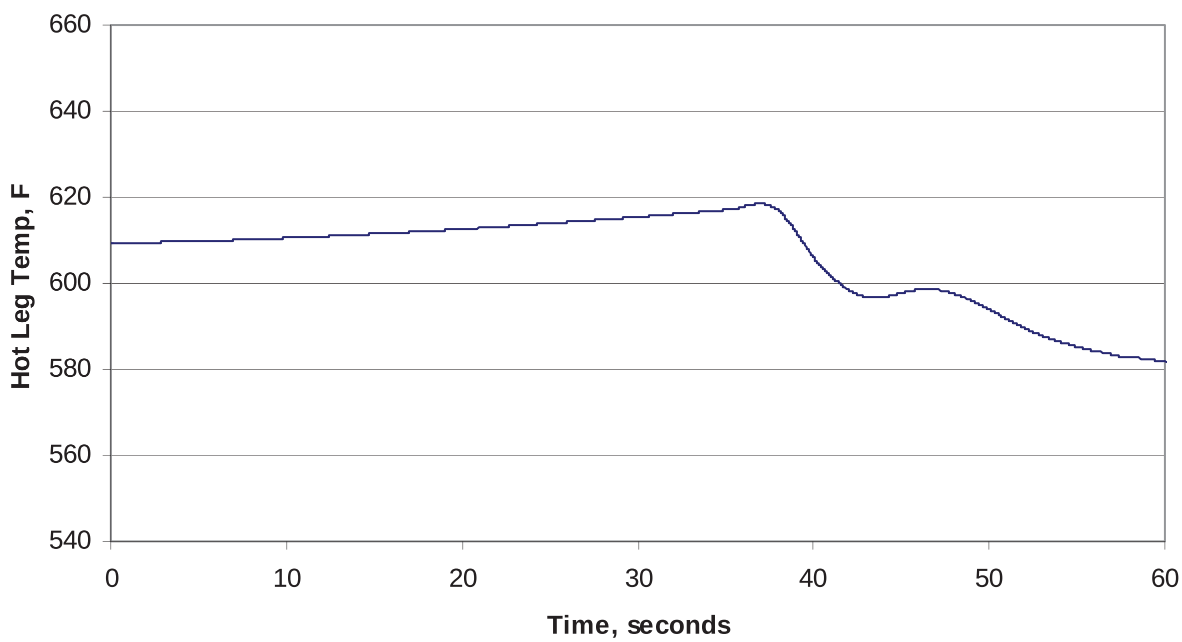


Figure 2.8.5.4.2-6: Rod Withdrawal at Power



2.8.5.4.3 Control Rod Mis-Operation

2.8.5.4.3.1 Regulatory Evaluation

The CR-3 review covered the types of Control Rod Mis-Operations that are assumed to occur, including those caused by a system malfunction or operator error. The review covered: 1) descriptions of rod position, flux, pressure, and temperature indication systems, and those actions initiated by these systems (e.g., turbine runback, rod withdrawal prohibit, rod block) which can mitigate the effects or prevent the occurrence of various mis-operations; 2) the sequence of events; 3) the analytical model used for analyses; 4) the important inputs to the calculations; and 5) the results of the analyses.

The NRC's acceptance criteria for Control Rod Mis-Operation are based on:

- GDC-10, insofar as it requires that the reactor coolant core is designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operations including the effects of anticipated operational occurrences (AOOs);
- GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to initiate automatically operation of systems and components important to safety under accident conditions; and
- GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

CR-3 Current Licensing Basis

As noted in CR-3 FSAR Section 1.4, the general design criteria used during the licensing of CR-3 predates the GDC provided in 10CFR50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in Section 1.4 of the FSAR were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the reactor coolant core is designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operations including the effects of AOOs, [GDC-10];
- FSAR Section 1.4.14, Core Protection Systems; and FSAR Section 1.4.15 Engineered Safety Features Protection Systems, insofar as these criteria require that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to initiate automatically operation of systems and components important to safety under accident conditions [GDC-20]; and
- FSAR Section1.4.31 Reactivity Control Systems Malfunction, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems. [GDC-25]

2.8.5.4.3.2 Technical Evaluation

Introduction

During normal plant operation, control rods maintain the desired core reactivity. Raising and lowering these rods into the core provides a way of controlling the reactivity and consequently, the power. Control rod assemblies are normally grouped into patterns which maintain a symmetric core power distribution. A mechanical or electrical failure can cause a control rod assembly to become misaligned from its group reference, causing an asymmetric reactivity distribution. A stuck out control rod can also result in a reduction in the total available control rod worth for shutdown of the reactor.

The three types of control rod misalignment that can occur include:

- A stuck-out control rod assembly.
- A stuck-in control rod assembly (encompassed by dropped rod as described below).
- A dropped control rod assembly (limiting in the category as described below).

Misalignment occurs on reactor trip if one rod fails to insert and remains stuck in the fully withdrawn position. This condition is the basis for the shutdown margin (SDM) evaluation to determine if sufficient negative reactivity addition is available to achieve hot shutdown condition when considering the maximum worth rod stuck to meet the Improved Technical Specification (ITS) required limit (ITS 3.1.1, Shutdown Margin). This shutdown margin requirement is met as a criterion of the reactor core design on a cycle-specific basis.

The second type of rod misalignment occurs during withdrawal of the control rods if one rod becomes stuck at some position as the other rods continue in motion. This condition will affect the power distribution in the core and could lead to excessive power peaking.

The third type of rod misalignment occurs when a control rod drops into the core. A dropped control rod is defined as the deviation of a control rod from the average group position by more than an indicated 5 inches (equivalent to a 9 inch absolute error). This definition then covers the action of a stuck-in control rod during withdrawal of the others and a dropped control rod. A stuck-in rod is less limiting due to the time required to raise the control rods. Raising a control rod completely out of the core from a fully inserted position requires approximately 6 minutes. If a rod becomes stuck, the operator is informed by several alarms and has time to take corrective action. The term, dropped rod, refers to a stuck-in or dropped control rod assembly for the remainder of this Section. The resulting transient causes a rapid reduction in power and temperature due to the negative reactivity addition to the core. The reduction in coolant and fuel temperature combined with a negative moderator temperature coefficient and a negative Doppler temperature coefficient provide positive reactivity addition to the core, all of which contribute to a return to power. The magnitude of the return to power, in consideration of the asymmetric power distribution, could lead to excessive localized power.

The Integrated Control System (ICS) will take protective action if a control rod deviates by more than 9 inches. The ICS normally withdraws control rods to compensate for a loss in power. However, this would not be suitable when near full power if a control rod drops. The withdrawal of other rods after a rod is dropped would result in increased axial power peaking. When a rod is dropped while operating above

60% of rated power, rod out motion is inhibited. The ICS actions are described in FSAR Section 7.2.3. No operator action is required to mitigate the event.

Description of Analyses and Evaluations

The dropped rod event was analyzed for the EPU using RELAP5/MOD2-B&W (Reference 1). The RELAP5/MOD2-B&W code has been approved by the NRC for use in non-loss of coolant accident (non-LOCA) transient analyses (Reference 2). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The Reactor Coolant System model provides for heat transfer from the core, transport of the coolant to the once-through steam generators (OTSGs), and heat transfer to the OTSGs. The secondary model includes a detailed depiction of the Main Steam System. The secondary model also includes the delivery of feedwater, both main and emergency, to the OTSGs. The analyses were performed in compliance with methodology that has been approved by the NRC for B&W plant non-LOCA transient analysis (Reference 2).

The key input parameters and initial conditions used in the analysis of the dropped rod event are as follows:

- The event is initiated at the nominal EPU power level plus heat balance uncertainty (3026.1 MWt).
- The reactor coolant average temperature is 582°F, consistent with the increase in T_{AVG} being implemented in conjunction with the EPU.
- The dropped rod analyses considered a range of Doppler temperature coefficients (-1.30 x 10^{-5} Δk/k/°F to -2.00 x 10^{-5} Δk/k/°F) representing variations within the fuel cycle. Likewise, a range of moderator temperature coefficients was considered, ranging from 0 Δk/k/°F to -5.0 x 10^{-4} Δk/k/°F.
- The dropped rod analyses considered a range of initial fuel temperatures representing variations within the fuel cycle.
- A range of dropped rod worth values was considered in order to define a conservative set of statepoints. The statepoints are then used in the evaluation of limiting condition of operation (LCO) axial offset limits that protect the core during operation, and specifically protect against the departure from nucleate boiling ratio (DNBR) consequences of the Dropped Rod Event. The generation of the LCO axial offset limits, which considers margins to power peaking limits, uses maximum dropped rod worths (Table 2.8.2-2). The historical assumption for maximum control worth (0.28% Δk/k) remains bounding in comparison to the maximum dropped rod worths predicted for the EPU fuel cycles (Table 2.8.2-2).

- The analysis models reactor trip to occur on either high reactor coolant system (RCS) pressure (2400 psia) or RCS low pressure (1893.95 psia). The setpoints modeled include allowances for instrument error and setpoint error. Also, appropriate delays for signal processing and control rod assembly release are modeled.
- Minimum thermal design RCS flow rates are assumed in accordance with approved methodology (Reference 2).

The FSAR criteria for reactor protection for the dropped control rod accident are:

- The minimum DNBR shall remain greater than the DNBR design limit. The dropped control rod transient was analyzed to generate conservative statepoints defining the normalized power, coolant temperature, coolant flow rate, and reactor coolant system pressure. The system analysis statepoints from the dropped rod transients are then provided as input into the cycle-specific evaluation of margins for DNBR.
- The Reactor Coolant System pressure shall not exceed 110% of the design pressure.

Results

The responses of neutron power, thermal power, moderator temperature, and system pressure to a dropped control rod worth of 0.28% $\Delta k/k$ are shown for beginning of life (BOL) and end of life (EOL) conditions in Figures 2.8.5.4.3-1 and 2.8.5.4.3-2, respectively. In both cases, the neutron power decreased rapidly due to the sudden insertion of negative reactivity. This caused a rapid decrease in the core moderator temperature and fuel temperature. The reactivity feedback associated with these temperature decreases compensated for the worth of the control rod, limiting the minimum neutron power reached and causing power to rise above the minimum value. In the BOL case, the minimum neutron power reached was 65% of the rated power. In the EOL case, the more negative moderator and Doppler temperature coefficients limited the minimum core power reached to 77% of the rated power.

In both the BOL and EOL cases, the dropped control rod results in a decrease in the system pressure. In the BOL case, the system pressure continuously decreases until the Reactor Protection System low Reactor Coolant System pressure trip is reached at 73.2 seconds, resulting in a reactor trip signal. Control rod insertion begins after a brief delay, at 73.8 seconds. The minimum pressure in the RCS is reached coincident with control rod insertion, and the transient is terminated shortly thereafter. In the EOL case, the system pressure reaches an equilibrium value that is less than the initial system pressure. Since the system pressure is always less than the initial pressure for both the BOL and EOL cases, the Reactor Coolant System pressure did not exceed the code pressure limit of 2750 psig.

The DNBR acceptance criterion is satisfied each reload by performing a cycle specific check, as discussed in Section 2.8.2, Nuclear Design. A set of LCO offset limits was generated that will provide DNBR protection, based on the conservative statepoints generated within the dropped rod system response described herein, and based on the predicted EPU fuel cycles.

2.8.5.4.3.3 Conclusion

CR-3 has reviewed the analyses of Control Rod Mis-Operation events and concludes that the analyses have adequately accounted for the changes in core design required for plant operation at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure the SAFDLs will not be exceeded during normal or anticipated operational transients. Based on this, CR-3 concludes that the plant will continue to meet the CR-3 current licensing basis requirements with respect to FSAR Section1.4.6, 1.4.14, 1.4.15, and 1.4.31 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Control Rod Mis-Operation events.

2.8.5.4.3.4 References

1. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary) and BAW-10164NP-A, Revision 6, June 2007 (Nonproprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

2. BAW-10193-P-A (Proprietary), RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors, Parece, M. V., January 2000.

Figure 2.8.5.4.3-1 CR-3 Dropped Control Rod from Rated Power at BOL Conditions (0.28% Δk/k Dropped Control Rod Worth)

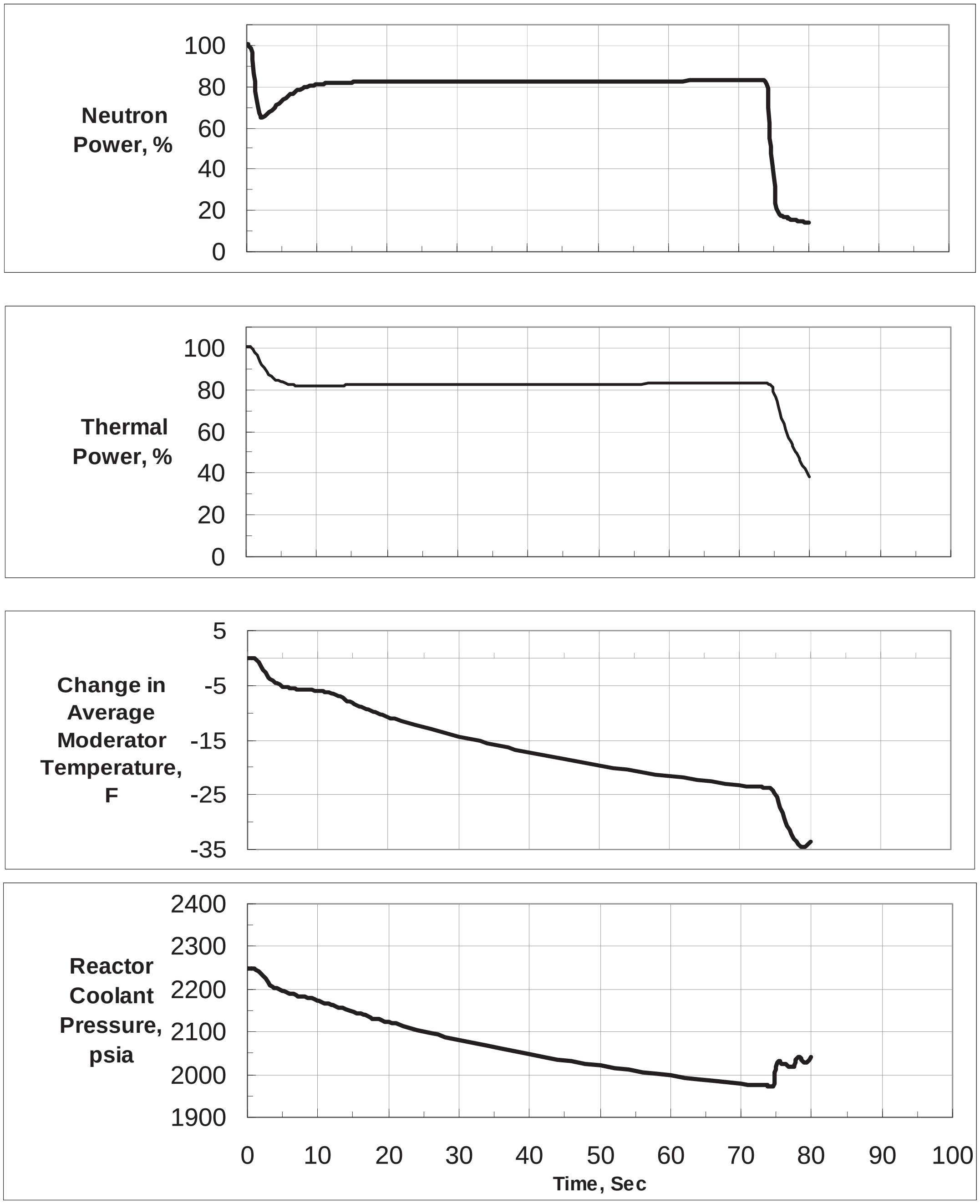
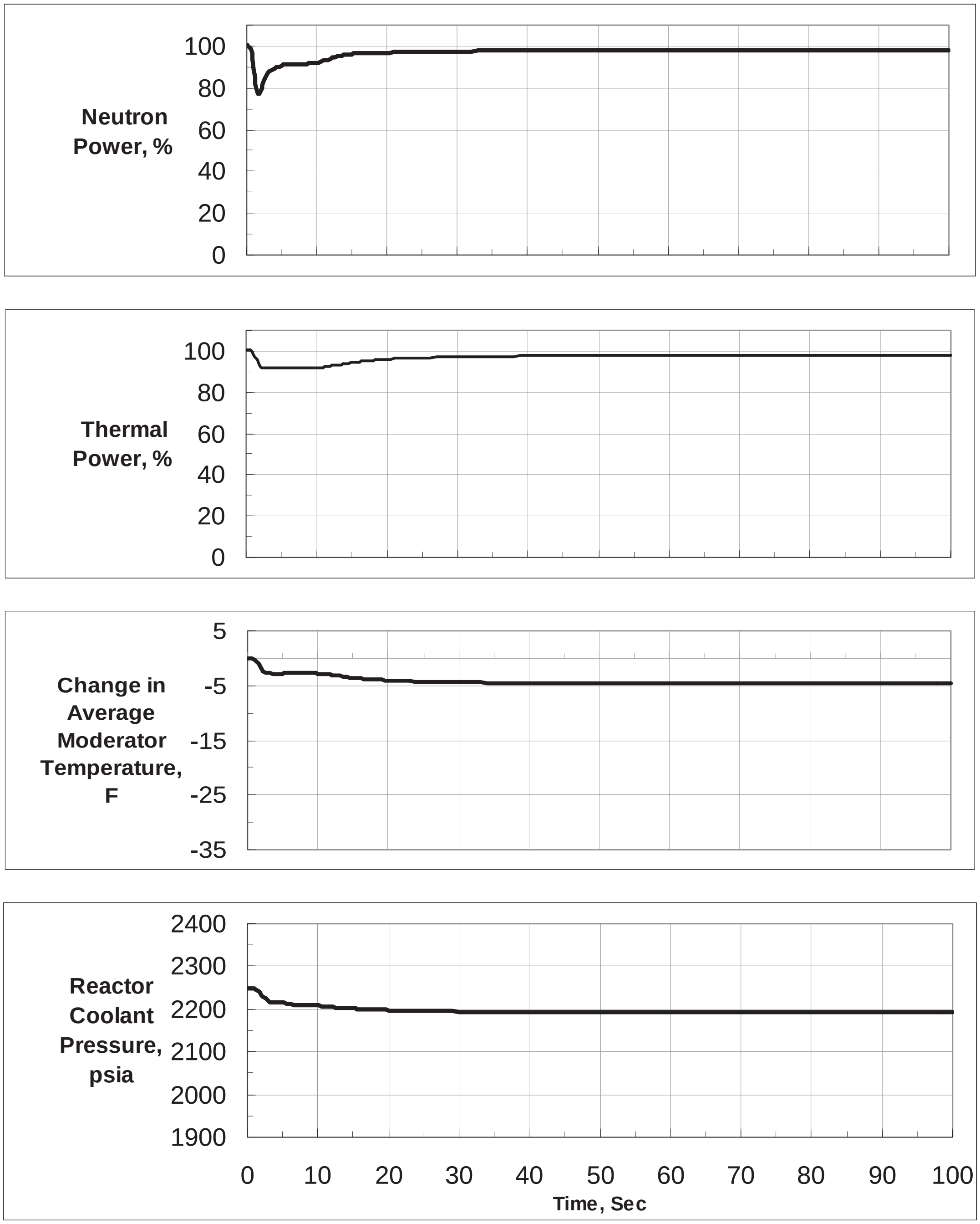


Figure 2.8.5.4.3-2 CR-3 Dropped Control Rod from Rated Power at EOL Conditions (0.28% Δk/k Dropped Control Rod Worth)



2.8.5.4.4 Startup of an Inactive Loop at an Incorrect Temperature

2.8.5.4.4.1 Regulatory Evaluation

A Startup of an Inactive Loop Transient may result in either an increased core flow or the introduction of cooler or deborated water into the core. This event causes an increase in core reactivity due to decreased moderator temperature or moderator boron concentration. The CR-3 review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses.

The NRC's acceptance criteria for Startup of an Inactive Loop Transient are based on:

- GDC-10, insofar as it requires that the Reactor Coolant System (RCS) be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs);
- GDC-20, insofar as it requires that the protection system be designed to automatically initiate the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences;
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the reactor coolant pressure boundary (RCPB) are not exceeded during AOOs;
- GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the RCS be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs [GDC-10];

- FSAR Section 1.4.14, Core Protection Systems; and CR-3 FSAR 1.4.15. Engineered Safety Features Protection Systems, insofar as these criteria require that the protection system be designed to automatically initiate the operation of appropriate systems to ensure that SAFDLs are not exceeded as a result of operational occurrences [GDC-20];

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during AOOs [GDC-15];

- FSAR Section 1.4.32, Maximum Reactivity Worth of Control Rods, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core [GDC-28]; and

- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded [GDC-26].

2.8.5.4.4.2 Technical Evaluation

Introduction

The CR-3 FSAR Section 14.1.2.5 refers to the Startup of an Inactive Loop at an Incorrect Temperature as the Cold Water Accident (CWA). The classic CWA is the start of an idle reactor coolant loop that has been isolated by primary system isolation valves. Since the CR-3 plant design does not include primary system isolation valves, the classic cold water accident cannot occur. However, when the reactor is operated with one idle pump and this pump is started, the increased flow rate causes the average core temperature to decrease. When the moderator temperature coefficient is negative, positive reactivity is introduced into the core and a power rise occurs.

Even though Operating License Condition 2.C.(3) does not allow the plant to be critical with less than three reactor coolant pumps operating, the analysis assumed that the plant was operating with only one reactor coolant pump in each loop. Furthermore, although plant operating procedures and the reactor coolant pump control circuitry prevent starting an idle pump if the power is above 30% of full power, the analysis assumes that the plant is initially operating at 50% of the rated EPU power. From this initial condition, the remaining two idle pumps were started. Startup of two idle pumps causes the system flow to rapidly increase to the system design flow.

The RELAP5/MOD2-B&W computer code (Reference 1) was used to evaluate the plant response to the CWA. The code was used in with the approved methodology described in Reference 2. The RELAP5/MOD2-B&W computer model includes the Reactor Coolant System including the reactor vessel and fuel, the steam generators, and the attached steam lines. The reactivity changes associated with the startup of two idle pumps is modeled and the expected changes in reactor coolant (RC) temperature and pressure are predicted.

Key input assumptions and boundary conditions for the CWA include:

- The event is initiated at 50% of the nominal EPU power level (1507.0 MWt).
- The reactor coolant average temperature is 582°F, consistent with the increase in T_{AVG} being implemented in conjunction with the EPU. The conditions in the idle loops are determined by the RELAP5/MOD2-B&W initialization.
- The reactor coolant pressure is set to the nominal value (2170 psia at the hot leg pressure tap).
- Minimum thermal design RCS flow rate of 374,880 gpm is assumed in accordance with approved methodology (Reference 2) and consistent with minimum thermal RCS flow requirements proposed in Improved Technical Specifications (ITS) 3.4.1 changes (see Attachment 2, Operating License and Technical Specification Changes (Markup)). The pre-event initial flow rates were verified to be less than 50% of the minimum design RCS flow. Once the event is initiated by starting the idle loop reactor coolant pumps, the flow increases to the minimum thermal design flow. The time over which the increase occurs is determined by RELAP5/MOD2-B&W.
- The Doppler temperature coefficient (-1.30×10^{-5} Δk/k/°F) is chosen to be the least negative within the range of Doppler coefficients expected for the EPU. This is conservative since fuel temperature increases throughout the event with increased thermal power response.
- The moderator temperature coefficient (MTC) is chosen to be -5.0×10^{-4} Δk/k/°F, and is based on end-of-cycle conditions, resulting in a conservative response to the reduction in moderator temperature.
- The initial pressurizer level (240 inches indicated) is increased relative to the nominal value to account for measurement uncertainty. This is conservative with respect to reactor coolant pressure response.
- The pressurizer code safety valves are modeled. Both valves are assumed available with a total relief capacity of 635,946 lbm/hr.
- The analysis models reactor trip to occur on either high RCS pressure (2400 psia) or high neutron flux (112 % of EPU nominal power level). The setpoints modeled include allowances for instrument error and setpoint error. Also, appropriate delays for signal processing and control rod assembly release are modeled.
- The tripped rod worth assumed for the analysis is based on a minimum shutdown margin of 1.0 %Δk/k. This is less than the minimum Modes 1 and 2 shutdown margin required for the EPU (1.3 %Δk/k as detailed in separate attachment associated with ITS changes).

The specific FSAR acceptance criteria applied by CR-3 for this event are as follows:

- The minimum departure from nucleate boiling ratio (DNBR) shall remain greater than the DNBR design limit.
- The RCS pressure shall not exceed 110% of the design pressure, or 2764.7 psia.

Results

The results of the CWA are illustrated in Figure 2.8.5.4.4-1. The maximum moderator temperature decrease is approximately 5°F. This temperature decrease, coupled with the end-of-life moderator temperature coefficient, results in a peak neutron power of 92% of 3014 MWt at 12.2 seconds. The thermal power lags behind the neutron power and peaks at 79% of 3014 MWt at 12.8 seconds.

Since the RCS flow increases throughout the event, the DNBR increases also. This ensures that the DNBR criterion continues to be met. Further, reactor thermal power remains well below 112% - the LCO offset limits for the plant are such that the plant is DNB-protected for thermal power levels up to 112% with four reactor coolant pumps operating. The increased core power due to moderator feedback causes a mismatch between the primary heat generation and the secondary heat removal. As a result, the primary pressure increases and peaks 17.8 seconds into the event at a value of 2337 psia. This is well below 110% of the design pressure (2764.7 psia), and therefore is acceptable.

Based on the analysis, it is concluded that the design of the plant prevents adverse conditions from occurring in the event that the idle pumps are started. Since the DNBR does not decrease below the DNB design limit and the primary pressure does not exceed 110% of the design pressure, the protection criteria are satisfied.

2.8.5.4.4.3 Conclusion

CR-3 has reviewed the analyses of the Startup of an Inactive Loop-Transient and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the analyses have demonstrated that the reactor protection and safety systems will continue to ensure that the specified acceptable fuel design limits and the reactor coolant pressure boundary pressure limits will not be exceeded as a result of this event. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, 1.4.14, 1.4.15, 1.4.27, and 1.4.32 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Startup of an Inactive Loop at an Incorrect Temperature event.

2.8.5.4.4.4 References

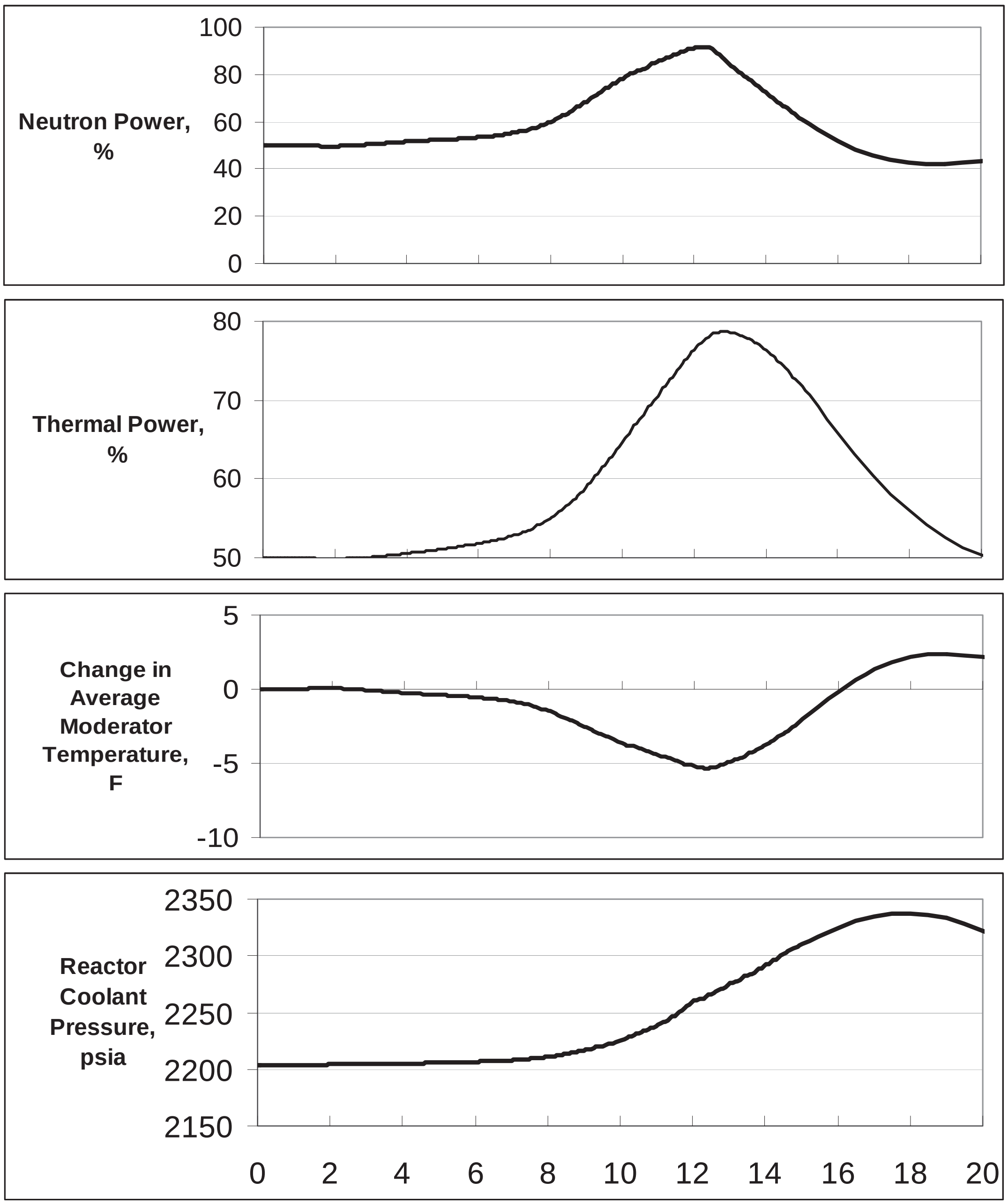
1. AREVA NP Topical Report BAW-10164PA, Rev. 6(Proprietary) and BAW-10164NP-A, Rev. 6 (Nonproprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

2. BAW-10193-PA, Rev. # (Proprietary), RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors, Parece, M. V..

Table 2.8.5.4.4-1: Cold Water Accident Sequence of Events

Event	Time (s)
Two Idle Reactor Coolant Pumps Started	0.0
Peak Neutron Power Reached	12.2
Minimum Core Average Moderator Temperature Reached	12.3
Peak Thermal Power Reached	12.8
Full Thermal Design Flow Reached	12.9
Peak Primary Pressure Reached	17.8
Analysis Terminated	30.0

Figure 2.8.5.4.4-1 CR-3 Two-Pump Startup From 50% Power



2.8.5.4.5 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

2.8.5.4.5.1 Regulatory Evaluation

Unborated water can be added to the Reactor Coolant System (RCS), via the Chemical and Volume Control System (CVCS). This may happen inadvertently because of operator error or CVCS malfunction, and cause an unwanted increase in reactivity and a decrease in shutdown margin. The operator should stop this unplanned dilution before the shutdown margin is eliminated. The CR-3 review covered (1) conditions at the time of the unplanned dilution, (2) causes, (3) initiating events, (4) the sequence of events, (5) the analytical model used for analyses, (6) the values of parameters used in the analytical model, and (7) results of the analyses.

The NRC's acceptance criteria for a Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant are based on:

- GDC-10, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).
- GDC-15, insofar as it requires that the Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including AOOs [GDC-10];

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs [GDC-15]; and

- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. [GDC-26].

2.8.5.4.5.2 Technical Evaluation

Introduction

The CVCS boron dilution function at CR-3 is performed by the Makeup and Purification (MU) System. CR-3 FSAR Section 14.1.2.4 includes consideration of a MU System malfunction that would result in a decrease in RCS boron concentration. This event is commonly referred to as the Moderator Dilution Accident (MDA). An MDA is defined as an inadvertent addition of unborated moderator into the RCS by either operator error or MU System malfunction which causes an unwanted increase in reactivity and a decrease in shutdown margin. This event is classified as Condition II, per the classifications in Section 2.8.5.0, Non-LOCA Analysis. The MDA is inherently terminated or requires operation of the normal protection systems to maintain the integrity of the fuel and/or the RCS.

For CR-3, the only credible flow path that exists for boron dilution is through the MU System from the demineralized water or reactor coolant bleed tanks (FSAR Section 14.1.2.4). Flow into the Makeup Tank is terminated as a result of reactor trip. Therefore, the dilution source after reactor trip is limited to the makeup tank volume. When the makeup pump is operating, a potential exists for dilution water to enter the Reactor Coolant System. Several interlocks and alarms, as described in FSAR Sections 7.2.2.3.3 and 9.1.2.6, and plant procedures are provided to prevent improper operation.

The MU System normally has one pump in operation, which supplies makeup to the RCS and the required seal flow to the Reactor Coolant Pumps (RCPs). Thus, the total makeup flow available is normally limited by pump capacity. When the makeup rate is greater than the letdown rate, the net water increase will cause the pressurizer level control to close the makeup valve. The nominal moderator dilution event considered is the pumping of water with zero boron concentration from the makeup tank to the RCS. It is possible, however, to have a slightly higher flow rate during transients when the system pressure is lower than the nominal value and the pressurizer level is below normal. Furthermore, with a combination of multiple valve failures or mis-operations, plus more than one makeup pump operating with reduced RCS pressure, the resultant inflow rate could be much higher than the nominal rate. This constitutes the maximum dilution accident.

For the EPU, the borated water storage tank (BWST) boron concentration is increased (Section 2.8.2, Nuclear Design). The moderator dilution accident conservatively assumes that the initial RCS boron concentration is equal to the BWST boron concentration. As a result, the impact of the moderator dilution is increased for the EPU. The analyses for the EPU consider the increased initial boron concentration in the determination of reactivity insertion rates for the dilution event.

Description of Analyses and Evaluations

The analysis used the methodology defined in BAW-10193 (Reference 1). The RELAP5/MOD2-B&W code (Reference 2) is used in conjunction with the methodology, which also incorporates conservative setpoints and capacities to arrive at a conservative result.

The MDA analysis at the EPU conditions considered dilution flow rates of 70 gpm, 140 gpm, 353.5 gpm, and 500 gpm. The 70 gpm flow rate matches the flow rate through a single letdown cooler. The maximum letdown flow rate is 140 gpm. The maximum realistic makeup and safety injection flow rate for CR-3 is 353.5 gpm. The 500 gpm dilution flow rate exceeds the maximum makeup flow rate that is possible at operating pressures, but is a historically analyzed value.

Two sensitivity studies were performed. The first study considered the MDA with and without transient pressurizer spray. Although it would seem that the spray would reduce the pressurizer pressure, it also served to delay the reactor trip and alter the peak pressure and peak thermal power. The second study evaluated the effect of initial pressurizer level on peak pressure and peak thermal power. All dilution rates were analyzed at the minimum and maximum initial pressurizer level.

The key input parameters and initial conditions used in the analysis of the MDA are as follows:

- The core power level assumed is the EPU targeted value, plus heat balance uncertainty (3026.1 MWt).
- The reactor coolant average temperature is assumed to be 582°F, consistent with the increase in T_{AVG} planned in conjunction with EPU.
- The RCS pressure is assumed to be 2170 psia in the RCS hot leg.
- A minimum RCS flow rate is assumed (374,880 gpm), consistent with approved methodology (Reference 1).
- The nominal pressurizer level at full power is 220 inches indicated. The MDA analysis considered variations in the initial pressurizer level ranging from 200 inches to 240 inches.
- A least-negative Doppler temperature coefficient (-1.30 x 10^{-5} Δk/k/°F) typical of beginning-of-cycle conditions is used since it yields the maximum rate of power increase. The MDA will result in increased fuel temperature, and so a least-negative Doppler coefficient minimizes the negative feedback as the fuel temperature increases.
- The most-positive moderator temperature coefficient (0.0 Δk/k/°F) permitted by CR-3 Improved Technical Specifications (ITS) is used since this yields a larger rate of power increase than a negative value for the moderator temperature coefficient (MTC) at hot full power (HFP).
- For the RCS response to the initial dilution event, the tripped rod worth is calculated to be that which results in the minimum Modes 1 and 2 shutdown margin of 1.0 %Δk/k at hot zero power (HZP). For the continued dilution post-reactor trip, the reactivity addition is assured to be less than 1.3 %Δk/k at HZP, which is the minimum shutdown margin for the EPU as required for the main steam line break (MSLB) core response (Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment) and as implemented in Improved

Technical Specification changes contained in Attachment 2, Operating License and Technical Specification Changes (Markup) to this document.

- A range of dilution flow rates is considered.

- The reactivity insertion rates corresponding to the postulated dilution flow rates are determined based on conservative assumptions for initial boron concentration and boron worth. The reactivity insertion rates are conservatively held constant throughout the event. In calculating reactivity insertion rates, a conservative ratio of boron concentration change to inverse boron worth is assumed.

- A reactor trip terminates unborated water addition to the makeup tank, therefore the available diluent volume post-trip cannot exceed the volume of the makeup tank, 600 ft^3. Although flow into the RCS would be terminated by high pressurizer level, the analysis assumes the full contents of the makeup tank continues to dilute the RCS post-reactor trip.

- Reactor trip is assumed to occur on either high RCS pressure (2400 psia) or nuclear overpower (112% power). These setpoints are consistent with pre-EPU values, which also include consideration of instrument uncertainties to arrive at the conservative Reactor Protective System (RPS) settings for the event.

- Post-reactor trip, the decay heat level is assumed to be 1.0 times the 1971 ANS standard with heavy actinides (Reference 1).

The MDA is analyzed to show that the core and Reactor Coolant System are not adversely affected by the event. The analysis demonstrates that the departure from nucleate boiling (DNB) limits are not violated and that the positive reactivity inserted into the core remains less than the minimum required shutdown margin (SDM). Also, the reactor coolant pressure is evaluated to ensure that the RCPB remains intact. The specific FSAR criteria for this event are:

- The reactor thermal power shall remain below 112 percent of 3014 MWt. This acceptance criterion is chosen since DNB and linear heat rate analyses assume a maximum power level of 112% of 3014 MWt. By maintaining thermal power below 112% of 3014 MWt the core is assured of avoiding DNB, and by maintaining total power below 112% of 3014 MWt centerline fuel melt limits will not be exceeded. Therefore cladding damage and fuel melt are avoided.

- The peak RCS pressure shall remain below 110 percent of the design pressure of the Reactor Coolant System. With a RCS design pressure of 2500 psig (2514.7 psia), the peak pressure shall remain below 2750 psig (2764.7 psia). This requirement stems from Section III of the American Society of Mechanical Engineers (ASME) Code that defines the safety limit for the Reactor Coolant System as 110% of the design pressure. Pressures up to but not exceeding the safety limit result in acceptable stresses in the RCPB.

- The positive reactivity addition due to the continued insertion of unborated water will be less than the available SDM at HZP with control rods inserted of 1.3% $\Delta k/k$. By limiting the reactivity inserted to less than the available SDM, criticality will be assured not to occur after reactor trip.

Results

The sequence of events for the MDA is listed in Table 2.8.5.4.5-1 and the calculated results are tabulated in Table 2.8.5.4.5-2 for the most limiting MDA transient analyzed. The results of the analyses demonstrate that a moderator dilution accident at the EPU conditions with dilution rates corresponding to 70 gpm, 140 gpm, 353.5 gpm, and 500 gpm meet the acceptance criteria for the CR-3 plant.

As the reactivity insertion rate (RIR) was increased (i.e., increased dilution flow), the peak normalized thermal power increased. The highest RIR produced the highest peak RCS pressure of 2698 psia, which was less than the RCS pressure limit of 2750 psig (110% design pressure). In all cases the reactor was tripped on the high RCS pressure trip.

Consistently for all dilution rates, the events with pressurizer spray actuated yielded the most limiting results. Using pressurizer spray, the dilution rates were analyzed using minimum (200 inches) and maximum (240 inches) initial pressurizer level. The peak primary pressures were predicted for the cases that considered maximum pressurizer level. The peak thermal power and RCS temperatures were calculated for the cases that modeled minimum pressurizer level. The peak primary pressure reached for each case was below the maximum allowable primary pressure of 2750 psig. The core thermal power for each case remained less than 112% of full power.

An additional requirement addressed by the MDA was the continued addition of the diluent post-trip. For each dilution rate, a calculation was performed to determine the amount of reactivity (shutdown margin) required to overcome the emptying of the contents of the entire makeup tank. The minimum required shutdown margin (1.3% $\Delta k/k$, required by changes to the Improved Technical Specifications as shown in Attachment 2) was shown to be adequate to preclude recriticality for this scenario, as shown in Table 2.8.5.4.5-2.

2.8.5.4.5.3 Conclusion

CR-3 has reviewed the analyses of the Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the reactor protection and safety systems will continue to ensure that the SAFDLs and the RCPB pressure limits will not be exceeded as a result of this event. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.6, 1.4.9, and 1.4.27 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant.

2.8.5.4.5.4 References

1. AREVA NP Topical Report BAW-10193-P-A, Revision 0, January 2000 (Proprietary), “RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors.”

2. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

Table 2.8.5.4.5-1: MDA Sequence of Events –

Dilution Rate	70 gpm (sec)	140 gpm (sec)	353.5 gpm (sec)	500 gpm (sec)
Dilution Initiated	0.0	0.0	0.0	0.0
High RCS Pressure Rx Trip	343.44	200.06	91.03	69.82
Control Rods Insert	344.06	200.68	91.64	70.43
Peak Thermal Power Reached	344.11	200.74	91.72	70.52
Peak Primary Pressure Reached	346.49	203.3	94.37	73.16

Table 2.8.5.4.5-2: MDA Results

Dilution Rate	70 gpm	140 gpm	353.5 gpm	500 gpm
Peak Thermal Power (%FP)	106.03	107.38	108.72	109.47
Peak Primary Pressure (psia)	2697.90	2677.31	2692.98	2698.00
Reactivity Insertion due to Continued Dilution (% $\Delta k/k$)	1.201	1.225	1.265	1.293

2.8.5.4.6 Spectrum of Rod Ejection Accidents

2.8.5.4.6.1 Regulatory Evaluation

Control rod ejection accidents cause a rapid positive reactivity insertion together with an adverse core power distribution, which could lead to localized fuel rod damage. CR-3 evaluated the consequences of a control rod ejection accident to determine the potential damage caused to the reactor coolant pressure boundary (RCPB) and to determine whether the fuel damage resulting from such an accident could impair cooling water flow. The CR-3 review covered initial conditions, rod patterns and worths, scram worth as a function of time, reactivity coefficients, the analytical model used for analyses, core parameters which affect the peak reactor pressure or the probability of fuel rod failure, and the results of the transient analyses.

The NRC's acceptance criteria for the Spectrum of Rod Ejection Events are based on:

- GDC-28, insofar as it requires that the reactor control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to impair significantly the capability to cool the core.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following is the applicable CR-3 specific criteria:

- FSAR Section 1.4.32, Maximum Reactivity Worth of Control Rods, insofar as it requires that the reactor control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to impair significantly the capability to cool the core. [GDC-28].

2.8.5.4.6.2 Technical Evaluation

Introduction

10 CFR 50.67 establishes radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone for accident conditions. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation as described in Reference 1, along with the fuel rod failure mechanisms. The radiological consequences for the rod ejection accident are addressed in Section 2.9.2, Radiological Consequences Analyses.

The rod ejection accident (REA) is a Condition IV event based on the definitions provided in Section 2.8.5.0, Non-LOCA Analysis Introduction. The dose consequences are closely tied to the predicted fuel failures for the REA. Using the historically-applied methods for the EPU conditions would challenge the dose consequences. A new methodology in Reference 1 was employed to evaluate the fuel consequences of such an event which changes both the methods employed and the criteria to be met.

The severity of the REA primarily depends upon the worth of the ejected rod, the local peaking, the Doppler feedback, the moderator feedback, the delayed neutron fraction, and the initial reactor power level. The ejected rod worth is inherently limited by the worth of the inserted control rods. Control rods are used to control load variations only and boron dilution is used to compensate for fuel depletion.

The consequences of a rod ejection accident depend largely on the total energy deposited during the pulse and the rate at which the thermal energy resulting from the nuclear excursion is released to the coolant. During a rod ejection accident, the neutron power rise is extremely rapid. The speed of the neutron power rise allows very little heat transfer out of the fuel to the coolant. The large power rise heats the fuel and the Doppler feedback causes a power reduction prior to a control rod insertion by reactor trip. As the energy in the fuel is transferred to the coolant the primary system pressure increases. The rapid increase in power can lead to fuel failures and Reactor Coolant System (RCS) pressure increases to challenge the RCPB. If the fuel rods remain intact while the excursion is being terminated by the negative Doppler coefficient and the reactor trip, then the energy release rate is limited by a relatively low surface-to-volume ratio for heat transfer. The energy stored in the fuel rods will be gradually released to the coolant, over a period of several seconds, at a rate that poses no threat to the integrity of the RCS. However, if the magnitude of the nuclear excursion is so great that the fuel rod cladding does not remain intact, then both fuel and cladding may be dispersed into the coolant to such an extent that the heat transfer rate increases significantly. This fuel dispersal effect will be referred to as coolability concern #1.

Power excursions caused by reactivity disturbances of the order of magnitude occurring in rod ejection accidents could lead to three potential modes of fuel rod failure. The first is associated with low worth rod ejections and leads to very little fuel fragmentation internal to the cladding. The localized cladding degradation due to fuel pin pressure increases is insufficient to rupture the cladding outright, but weakens it so departure from nucleate boiling (DNB) could lead to failure. The second fuel failure mode is linked to higher reactivity insertion rates which leads to significant fuel melt. The failure mode could include rupture of the cladding and dispersion of the fuel and cladding into the coolant. The third failure mode is the most serious. During this failure mode, the fuel transitions from solid to liquid. The result is the possibility of molten fuel failing the fuel pin and entering the coolant. These latter conditions that have fuel dispersal into the coolant can also add to the power surge to the primary system and could affect pressure vessel integrity. This fuel dispersal effect will be referred to as coolability concern #2.

In evaluating the effects of these modes of failure, two failure thresholds are considered. The first threshold is associated with a gradual and usually minor cladding failure leading to dose consequences. It can be defined by the minimum heat flux for DNB at the cladding surface. If a pin is in DNB and the internal fuel rod pressure is above system pressure, a balloon failure could occur, exposing fuel to the coolant and potentially restricting the flow. This ballooning effect will be referred to as coolability concern #3. The second failure threshold is used to describe the energy required to cause failure by either mechanism described by coolability concern #1 or #2. This threshold is commonly defined as the fuel enthalpy threshold for prompt fuel failure.

As a result of the postulated pressure housing failure associated with the accident, reactor coolant is lost from the system. The impact of the loss of inventory is addressed by the spectrum of break sizes considered in the Loss-of-Coolant Accident (LOCA) analysis, Section 2.8.5.6.3, Emergency Core Cooling

System and Loss-of-Coolant Accidents. The maximum size hole resulting from a rod ejection is 2.765 inches in diameter. The rate of energy input resulting from a rod ejection results in a much lower reactor building pressure than those obtained for any rupture sizes considered in the LOCA, since the limiting break sizes with respect to containment pressure response are double-ended ruptures (Section 2.6.1, Primary Containment Functional Design).

The over-pressurization analysis satisfies the peak RCS limits and the methods are unchanged with the EPU. The fuel performance analysis satisfies the requirements for NRC Standard Review Plan Section 4.2 Appendix B and involves a methodology change. To be clear on the association between the method and the results, the technical evaluation and results are presented together for each method that addresses over-pressurization and fuel performance.

Description of Analyses and Evaluations: REA Over-Pressurization

The REA over-pressurization analysis was performed in accordance with BAW-10193PA (Reference 2). The Reference 2 methodology utilizes the plant design bases to establish acceptance criteria and input boundary conditions. The approved methodology includes the manner for determining the responses of the primary system, the secondary system, and the core to postulated accidents. In addition, the methodology requires the use of conservative setpoints, valve and pump capacities, and reactivity coefficients to demonstrate adequate margin to the applicable limits.

The RELAP5/MOD2-B&W (R5/M2-B&W) computer code (Reference 3) was used for the analysis of the REA. This code has been approved by the NRC for use in non-LOCA safety analyses (Reference 2). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The RCS model provides for heat transfer from the core, transport of the coolant to the once through steam generators (OTSGs), and heat transfer to the OTSGs.

Key inputs to the over-pressurization analysis include:

- The core power level was assumed to be 3026.1 MWt for the hot full power (HFP) cases analyzed, which is equivalent to the nominal EPU power level plus heat balance uncertainty. The hot zero power (HZP) cases were initiated at a core power level of 3.0261 MWt.
- A minimum thermal design flow rate of 374,880 gpm is assumed for the initial reactor coolant flow rate.
- The initial HFP reactor coolant average temperature is assumed to be 582°F, consistent with the planned increase in HFP T_{AVG} in conjunction with the EPU. The HZP reactor coolant average temperature is assumed to be a nominal value of 532°F.
- The initial RCS pressure is 2169.7 psia, as measured at the hot leg tap.
- The initial pressurizer level is set to the nominal, plus an additional amount to account for level measurement uncertainty. The assumed value of 240 inches indicated is conservative with respect to the over-pressurization event.

- The pressurizer code safety valves (PSVs) are modeled with a total relief capacity of 635,946 lbm/hr of saturated steam at 2750 psig. Operation of the power-operated relief valve (PORV) was not credited.

- The ejected rod worth is assumed to be 0.65 %Δk/k for the HFP cases, and is assumed to be 1.0 %Δk/k for the HZP cases. These ejected rod worths conservatively bound the maximum ejected rod worths confirmed on a cycle-specific basis.

- The REA evaluated at HFP and HZP operating conditions for both Beginning-of-Cycle (BOC) and End-of-Cycle (EOC). BOC analyses assumed a least-negative Doppler temperature coefficient (DTC) of -1.30×10^{-5} Δk/k/°F for HFP and HZP cases. For EOC analyses, the least-negative DTC assumed is -1.45×10^{-5} Δk/k/°F for HFP and HZP cases.

- Similar to the DTC, least-negative moderator temperature coefficients (MTCs) are chosen for the time-in-life. The MTC was selected to be 0.0 Δk/k/°F at BOC for HFP conditions and 0.75×10^{-4} Δk/k/°F at BOC for HZP conditions. A bounding, least-negative MTC of -1.5×10^{-4} Δk/k/°F is assumed for the EOC analyses at HFP and HZP conditions.

- The effective delayed neutron fraction (β_{EF}) is assumed to be 0.0060 for BOC and 0.0045 at EOC, for both HFP and HZP analyses.

- The tripped rod worth is that amount required to meet a minimum shutdown margin requirement of 1.0 %Δk/k. This value is conservative with respect to the planned Modes 1 and 2 minimum shutdown margin of 1.3 %Δk/k planned in conjunction with the EPU.

- The analysis assumes that 97.3% of the energy generated is deposited within the fuel. The analysis conservatively assumes that the amount not deposited in the fuel directly heats the moderator.

- The decay heat post-reactor trip is based on 1.0 times ANS 1971, including heavy actinides.

- The Reactor Protection System will initiate a reactor trip using either the high neutron flux (112% of 3014 MWt) or high RCS pressure (2400 psia) trip functions. The analysis setpoints are conservatively derived, taking instrument uncertainties into account to ensure Technical Specification Allowable Values remain protected. An appropriate delay time is assumed prior to rod insertion, 0.610 seconds for the high flux trip and 0.420 seconds for the high reactor coolant pressure trip.

- In order to maximize the reactor coolant pressure response, it is assumed that no breach of RCS integrity occurs as a direct result of the rod ejection.

As described in the Introduction of this section, the methodology to address the overpressurization aspects of the rod ejection accident is unchanged, and thus is consistent with current methods. Separately, the fuel performance aspects of the rod ejection event are being addressed via a new methodology that is consistent with the most recent guidance issued by the NRC for reactivity insertion accidents. The over-pressurization analyses presented here are for the current methodology, for which the following acceptance criteria exist.

- In accordance with Reference 1, the peak fuel enthalpy limit of 280 cal/g threshold point to limit fuel fragmentation and minimize any dispersal of fuel to the coolant is no longer used and is replaced with the six criteria listed in Table 2.8.5.4.6-1.

- The peak RCS pressure shall be limited such that stresses will be prevented from exceeding ASME Service Level C limits. The NRC recognizes in RS-001 that a pressure limit of 3200 psig (3214.7 psia) as sufficient to preserve pressure-induced stress levels below the ASME Service Level C limit. Since the RELAP5/MOD2-B&W code will be used for this evaluation and the associated water properties are limited to the critical water pressure of 3200 psia, the practical RCS pressure acceptance criterion is 3200 psia. The analysis is bounding for RCS pressure response, and demonstrates that the over-pressure aspects are mitigated for the REA.

The current analysis reported in the FSAR demonstrates that further damage to the RCS due to the REA is avoided. This is addressed in the FSAR Section 14.2.2.4 by comparing the energy release associated with the rod ejection to the energy required to plastically deform the reactor vessel. The rod ejection event performed for the EPU addresses the criterion of no further damage to the RCS by adherence to the second acceptance criterion (related to RCS pressure) described above.

Description of Analyses and Evaluations: REA Fuel Performance Analysis

The methodology for the REA fuel performance analysis is contained in Reference 1. The computer codes used to analyze the REA event are COPERNIC (Reference 4), NEMO-K (Reference 5), LYNXT (Reference 6), and RELAP5/MOD2-B&W (References 3), respectively. COPERNIC is a fuel performance code that is used to obtain the gap conductance for both NEMO-K and LYNXT. The fuel property correlation equations from COPERNIC are used in NEMO-K and to develop inputs for LYNXT. NEMO-K is a 3-Dimensional kinetics code that is used to set initial boundary conditions for the ejected rod transient and to simulate the ejected rod transient. If there is not a high flux trip, the core power response from NEMO-K is input to RELAP5/MOD2-B&W. RELAP5/MOD2-B&W is a system Thermal-Hydraulic code to calculate the system response (system pressure, core inlet and outlet temperatures, and flow) of the ejected rod transient if the simulation continues past the initial power pulse without a high flux reactor trip. For this simulation, a low pressure is limiting for DNB ratio (DNBR) calculations so that the hole left by the ejected rod is considered. Two leak conditions are simulated as a full leak and a partial leak. The full leak area is defined as the inside diameter of the control rod flange (2.765 inch) as the break diameter and applied to the top of the upper head volume. An intermediate break size (partial leak) is defined as the area of the control rod flange minus the area of the control rod lead screw. The simulations continue until a trip in the RELAP5/MOD2-B&W model is reached. This simulation did not include any actions for the non-safety control systems that would tend to improve the results. The fuel rod powers from NEMO-K and system conditions from RELAP5/MOD2-B&W are supplied to LYNXT. LYNXT is an open channel T-H and fuel thermal code to calculate the fuel enthalpy, the temperature distributions, and the DNBR for the peak rod in the core during the transient simulation. The fuel failure threshold is calculated based on LYNXT calculations that determine the local power required to reach the DNBR Design Limit. The DNBR Design limit for the analysis is the fuel-related critical heat flux (CHF) correlation limit plus an allowance that is the difference between the Thermal Design Limit (TDL) and the Statistical Design Limit (SDL). The rods with powers in NEMO-K that are above the threshold power are counted as failed. If all the results are within the limits prescribed below, then the NRC criteria are met.

The key features of the REA performance analyses include:

- The RCS initial conditions (pressure, T_{AVG}) are consistent with the over-pressurization analyses.
- A range of ejected rod worths was considered.
- The analyses considered the impact of the rod ejection resulting in a leakage path for RCS coolant.
- In order to fully investigate the DNBR-related acceptance criteria, an additional power level of 20% of 3014 MWt was considered. It was assumed that T_{AVG} was equivalent to the HFP value for these cases.
- Uncertainties on reactivity parameters (ejected rod worth, MTC, DTC, β_{EFF}) are applied in accordance with Reference 3.
- A set of rod position limits was assumed for the analysis (Figure 2.8.5.4.6-1) that conservatively bounds cycle-to-cycle variation of ejected control rod worths and rod worth uncertainties.
- A range of reactivity parameters was considered for the REA performance calculations – including BOC and EOC conditions. The values are presented in Tables 2.8.5.4.6-4 and 2.8.5.4.6-5. The reactivity parameters may differ slightly from those used in the REA overpressure analyses, but nonetheless are bounding with respect to the EPU operation.
- The consideration of RCS depressurization and lower ejected rod worths results in scenarios where RPS high flux and high RC pressure trips are not activated. Other RPS functions are considered as described in Table 2.8.5.4.6-2. It should be noted that the Variable Low Pressure Trip (VLPT) is credited for the REA performance. This RPS trip function was not credited for accident mitigation prior to the EPU. However, the trip is fully functional and the trip setting is reviewed each refueling cycle to ensure that adequate core protection is maintained.

The radiological impact associated with the postulated REA at the EPU conditions was evaluated in accordance with Regulatory Guide 1.183, Appendix H guidance. Section 2.9.2, Radiological Consequences Analyses, describes the radiological consequences as well as pertinent assumptions and inputs used to evaluate the REA accident.

The methodology used to define the REA limits that need to be met to comply with fuel failure, coolability concerns and dose requirements is found in Reference 1. These limits are listed in Table 2.8.5.4.6-1. The first limit prevents fuel failure for the large pulses expected at HZP and ensures that coolability concerns #1 and #2 are precluded. This limit is based on the maximum expected oxide thickness for M5 clad employed at CR-3. The second limit ensures that after the pulse further insertion of energy does not fail the rod and also ensures that coolability concerns #1 and #2 are precluded. The third limit, no fuel melt, precludes coolability concern #2 for all initial power levels. The fourth limit is the maximum clad temperature below which clad ballooning failure is predicted not to occur and ensures that coolability concern #3 is precluded for all initial power levels. The fifth limit is the criterion for fuel failure and the sixth limit is the maximum pin failures that can fail and be within the dose limits defined by the analysis in Section 2.9.2, Radiological Consequences Analyses. In addition, for any rod failures above a certain enthalpy rise (Reference 1) the fission gas release is adjusted according to the formula defined by Reference 1.

Results: REA Over-Pressurization

The REA was analyzed at BOC and EOC for both HFP (based on the EPU power level) conditions and HZP conditions. The input assumptions, most notably the ejected rod worth values are chosen to provide conservative results for the overpressure analysis. It was also assumed that energy not deposited in the fuel contributed directly to heating the moderator. The results show that the peak RCS pressure remains within the acceptable limit (3200 psia) for all cases (Table 2.8.5.4.6-3). Therefore, no further loss of RCS integrity occurs as a result of REA events.

Results*:* REA Fuel Performance Analysis

The transient simulations for 0, 20, and 100 percent power are performed at BOC and EOC. The overall REA results for the plant transient analysis and fuel rod model are shown in Table 2.8.5.4.6-4 and Table 2.8.5.4.6-5 for BOC and EOC, respectively. All the results for maximum cal/g, maximum ∆cal/g, maximum fuel temperature, maximum clad temperature, and % failures meet the limits specified above.

The detailed results are shown for the transient conditions with the highest % failure which occurs at BOC for 20% power and the highest ∆cal/g which occurs at EOC for HZP. For BOC at 20% power, the sequence of events is shown in Table 2.8.5.4.6-6. The power versus time is shown in Figure 2.8.5.4.6-2 for the first several seconds where little system feedback occurs and in Figure 2.8.5.4.6-3 for the time after 5 seconds. The system temperature and pressure response is shown in Figure 2.8.5.4.6-4. The minimum DNBR, fuel and clad temperatures, and peak enthalpy rise versus time are shown in Figure 2.8.5.4.6-5, Figure 2.8.5.4.6-6, and Figure 2.8.5.4.6-7, respectively.

For EOC at HZP where the highest ∆cal/g occurs, the sequence of events is shown in Table 2.8.5.4.6-7. The power versus time is shown in Figure 2.8.5.4.6-8 for the first several seconds where little system feedback occurs and the plant trips on high flux. The minimum DNBR, fuel and clad temperatures, and peak enthalpy rise versus time are shown in Figure 2.8.5.4.6-9, Figure 2.8.5.4.6-10, and Figure 2.8.5.4.6-11, respectively.

2.8.5.4.6.3 Conclusion

CR-3 has reviewed the analyses of the Spectrum of Rod Ejection accidents and concludes that the analyses have adequately accounted for plant operation at the proposed power level and were performed using acceptable analytical models. CR-3 further concludes that the analyses have demonstrated that appropriate reactor protection and safety systems will prevent postulated reactivity accidents that could (1) result in damage to the RCPB greater than limited local yielding, or (2) cause sufficient damage that would significantly impair the capability to cool the core. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Section 1.4.32 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Spectrum of Rod Ejection events.

2.8.5.4.6.4 References

1. ANP-2788P, Rev.0 (Proprietary), “Crystal River 3 Rod Ejection Accident Methodology Report” (Submitted as an attachment to License Amendment Request #307, DPR-72, Approved January 28, 2010 as Amendment No. 237).

2. AREVA NP Topical Report BAW-10193-P-A, Revision 0, January 2000 (Proprietary), “RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors.”

3. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

4. BAW-10231PA, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” Framatome ANP, January 2004.

5. BAW-10221PA, Revision 0, “NEMO-K a Kinetics Solution in NEMO,” September 1998.

6. BAW-10156A, Revision 1, “LYNXT Core Transient Thermal-Hydraulic Program,” B&W Fuel Company, August 1993.

Table 2.8.5.4.6-1: REA Limits for CR-3 EPU

#	Limit Description	Limit
1	Maximum energy deposition during prompt power pulse for initial core powers $\leq$ 5%	$\leq$125 Δcal/g
2	Peak radial average fuel enthalpy for initial core powers $\leq$ 5%	$\leq$150 cal/g
3	Fuel Melt for all core power levels	= 0%
4	Maximum Cladding Temperature for all initial core power levels	Defined in Reference 1
5	Fuel Failure criterion for initial core powers > 5%	DNBR $\leq$ Design Limit
6	After power pulse, number of equivalent rods failed due to DNBR for the secondary side release [(a)], %	$\leq$ 4.3%

[a] Limit based on release fractions based on Regulatory Guide 1.183 values of 10% for noble gases and halogens.

Table 2.8.5.4.6-2 Trip Signal Parameters in Analysis

Trip Parameter	Analysis Limit	Sensor Scram Delay (seconds)
Excore High Flux, %RTP	112 (3/4 detectors[1])	0.42
Low RCS Pressure, psia	1893.95	0.61
High RCS Pressure, psia	2400.00	0.61
High Reactor Coolant Temperature, °F	620.00	5.67
Variable Low RCS Pressure, psia (T_{hot} is the RCS Hot Leg temperature)	11.59*T_{hot}-5049.46	5.67

1 Need 3 of 4 to trip in the model to conservatively account for 1 detector assumed failed and 2 of the remaining 3 detectors to sense a trip.

Table 2.8.5.4.6-3: REA - Summary of Pressure Results

Power Level	Cycle	Direct Mod. Heating	Peak RCS Pressure		Peak Thermal Power (Normalized to 3014 MWt)		Peak Neutron Power	
			Pressure psia	Time sec	%FP	Time sec	%FP	Time sec
HFP	BOC	without	2820.72	3.2738	145.77	0.7558	1166.39	0.1748
		with	2836.96	3.2060	149.74	0.7094	1207.3	0.1759
	EOC	without	2714.18	2.9223	214.24	0.2177	2522.79	0.1618
		with	2722.24	2.9462	202.34	0.2169	2362.65	0.1610
HZP	BOC	without	3035.26	3.2563	111.19	0.3976	13141.6	0.2714
		with	3130.57	2.9516	376.4	0.2724	15060.8	0.2730
	EOC	without	3035.93	2.1682	327.75	0.2706	22528.2	0.2229
		with	2977.09	2.2537	528.01	0.2228	20762.5	0.2226

Table 2.8.5.4.6-4 REA Performance Analysis Results for BOC

Parameter	Criterion	% Power Level		
		0	**20**	**100**
Rod Index Limit, % Withdrawn	-	0	125	265
Maximum Ejected Rod Worth, pcm	-	715	556	60
β_{eff}	-	0.0058	0.0058	0.0058
MTC, pcm/°F	-	2.5	0.0	-2.0
DTC, pcm/°F	-	-1.3	-1.24	-1.0
Initial F_Q	-	NA[a]	3.476	2.531
Maximum Transient F_Q	-	14.838	8.168	2.712
Initial $F_{\Delta H}$	-	NA[a]	2.272	1.710
Maximum Transient $F_{\Delta H}$	-	8.136	5.075	2.014
Max Neutron Power (Fraction of Power)	-	2.85	1.11	1.10
Maximum cal/g	$\leq$ 150	50.9	101.9[b]	98.8[b]
Maximum Δcal/g, prompt	$\leq$ 125	21.1	23.0[b]	6.3[b]
Maximum Fuel Temperature, °F	< fuel melt limit [d]	1670	3804	4231
Maximum Cladding Temperature, °F	< clad limit [d]	741	1353	1355
Minimum DNBR/Limit for rod failure	$\leq$ 1.000	2.150[b]	0.824	0.929
Time of High Flux Trip (initiation of safety bank insertion), seconds	-	0.795	None	None
Equivalent nominal rods failed for secondary side release [(c)], %	$\leq$ 4.3%	0.0	1.4	1.2

Notes:

[a] Not applicable since initial stored energy above the coolant temperature is zero.

[b] Criterion not applicable for these initial power levels but is shown for demonstrative purposes.

[c] Criteria based on release fractions based on Regulatory Guide 1.183 values of 10% for noble gases and halogens.

[d.] The fuel melt limit and the clad melt limit are described in detail in Reference 1.

Table 2.8.5.4.6-5 REA Performance Analysis Results for EOC

Parameter	Criterion	% Power Level		
		0	**20**	**100**
Rod Index Limit, % Withdrawn	-	0	125	265
Maximum Ejected Rod Worth, pcm	-	741	535	73
β_{eff}	-	0.0048	0.0048	0.0048
MTC, pcm/°F	-	-14.5	-25.0	-26.0
DTC, pcm/°F	-	-1.4	-1.36	-1.2
Initial F_Q	-	NA[a]	5.374	2.250
Maximum Transient F_Q	-	27.21	11.761	2.835
Initial $F_{\Delta H}$	-	NA[a]	2.272	1.711
Maximum Transient $F_{\Delta H}$	-	7.703	4.581	2.076
Max Neutron Power, (Fraction of Power)	-	6.71	1.88	1.14
Maximum cal/g	$\leq$ 150	54.1	77.8[b]	111.0[b]
Maximum Δcal/g, prompt	$\leq$ 125	34	17.4[b]	7.6[b]
Maximum Fuel Temperature, °F	< fuel melt limit [d]	1675	3354	4013
Maximum Cladding Temperature, °F	< clad limit [d]	1007	774	1436
Minimum DNBR/Limit for rod failure	$\leq$ 1.000	0.917[b]	1.263	0.939
Time of High Flux Trip (initiation of safety bank insertion)	-	0.705	None	None
Equivalent nominal rods failed for secondary side release [(c)] %	$\leq$ 4.3%	0.0	0.0	0.0

Notes:
[a] Not applicable since initial stored energy above the coolant temperature is zero.
[b] Criterion not applicable for these initial power levels but is shown for demonstrative purposes.
[c] Criteria based on release fractions based on Regulatory Guide 1.183 values of 10% for noble gases and halogens.
[d.] The fuel melt limit and the clad melt limit are described in detail in Reference 1.

Table 2.8.5.4.6-6 Event Timeline for BOC 20% Power

Event	Time (seconds)
Ejection begins	0.000
Rod N12 fully ejected	0.100
Peak Power reaches 111.3% power	0.137
Prompt enthalpy rise of 23.0 Δ cal/g	1.00
Power drops to 43.7% power	38.3
Minimum DNBR drops below limit for 2.5 GWD/MTU properties	8.4 – Full Leak N/A – Partial Leak
Minimum DNBR drops below limit for 50 GWD/MTU properties	11.5 – Full Leak N/A– Partial Leak
Event terminated on • Full Leak: High RCS Hot Leg Temperature Trip Full leak reaches 43.7% power, 0.824 Minimum DNBR/Design Limit • Partial Leak: High RCS Pressure Trip Partial leak reaches 45.7% power, 1.001 Minimum DNBR/Design Limit	39.3 – Full Leak 30.3 - Partial Leak

Table 2.8.5.4.6-7 Event Timeline for EOC HZP

Event	Time (seconds)
Ejection begins	0.000
Rod N12 fully ejected	0.100
High Flux Trip threshold reached	0.205
Peak Power reaches 671% power	0.218
Minimum DNBR drops below limit for 50 GWD/MTU properties	0.233
Minimum DNBR increase above the limit for 50 GWD/MTU properties	0.300
Prompt enthalpy rise of 33.8 Δ cal/g	0.350
Scram control rods begin to insert	0.705
Scram control rods are fully inserted	3.105

Figure 2.8.5.4.6-1 Rod Position Limits for REA Analysis

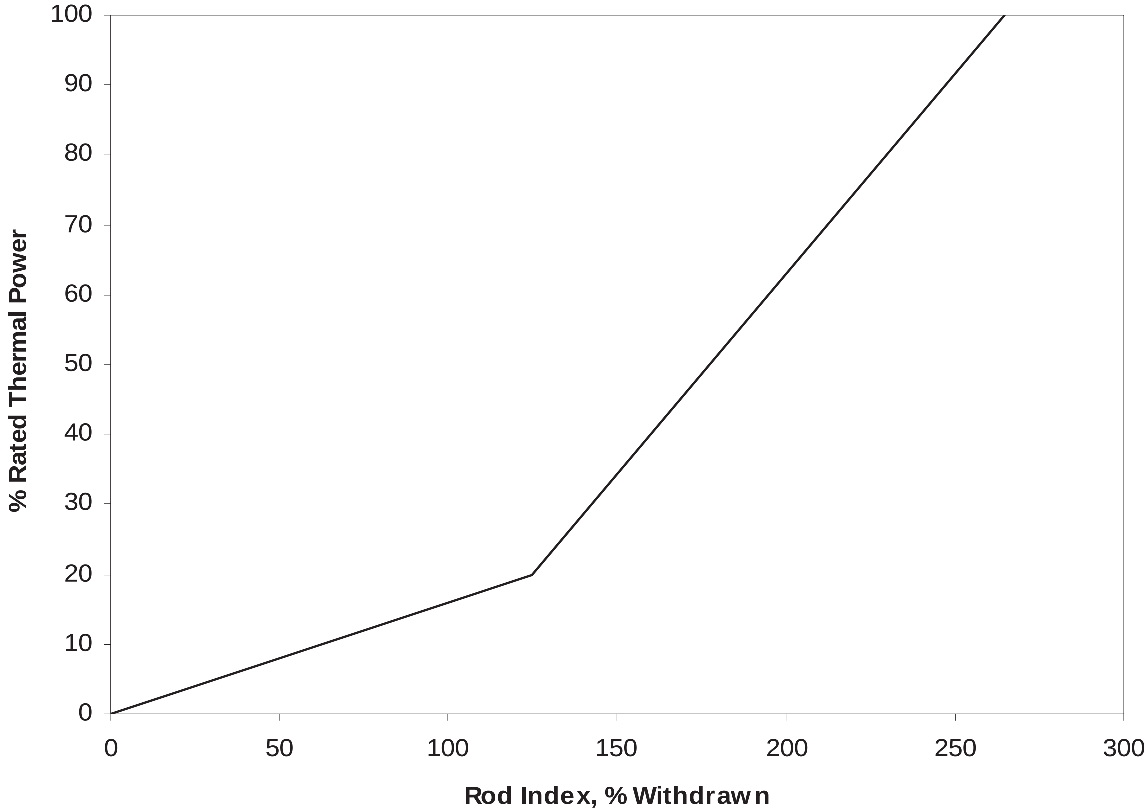


Figure 2.8.5.4.6-2 BOC 20% Power Transient

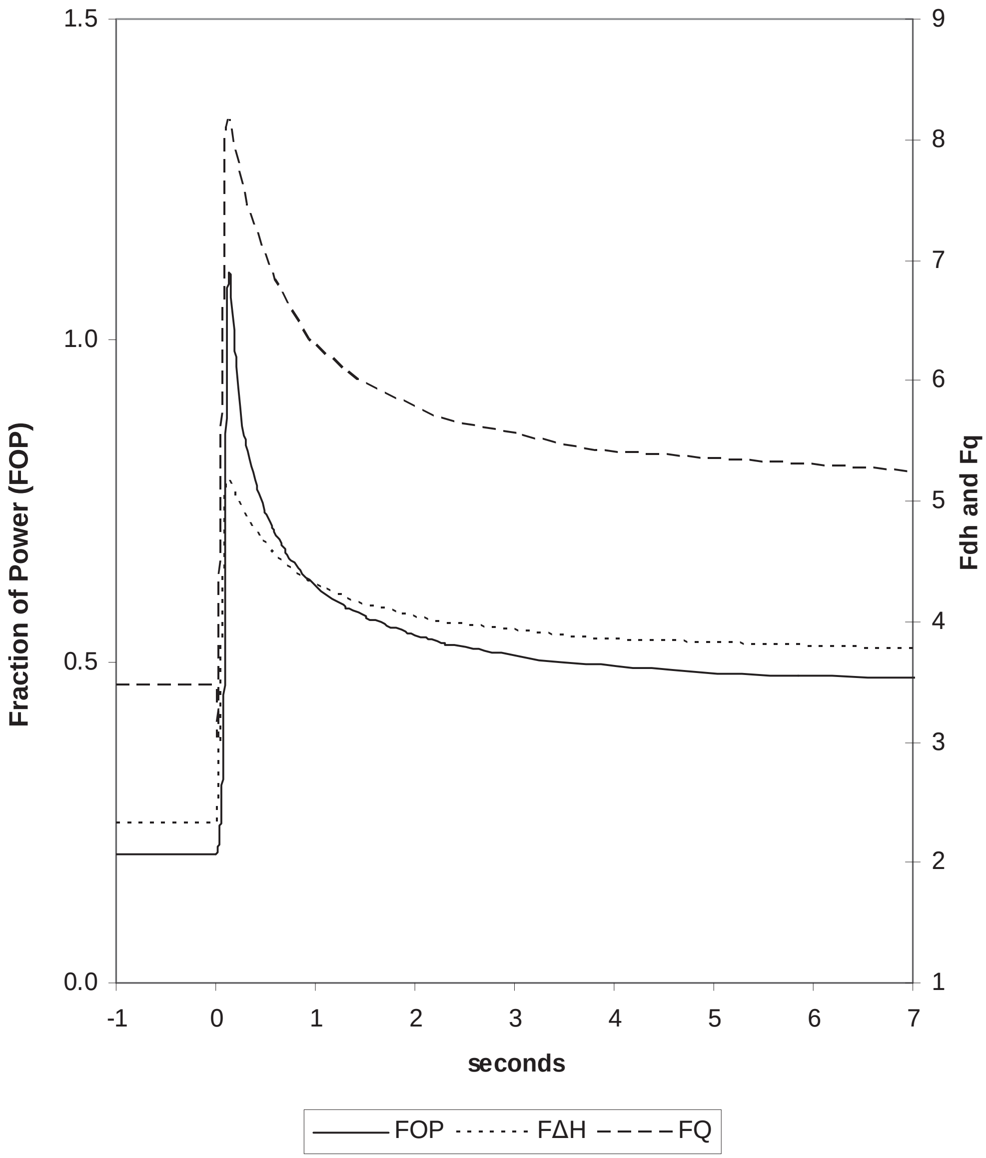


Figure 2.8.5.4.6-3 NEMO-K with RELAP5/MOD2 Conditions at BOC 20% Power

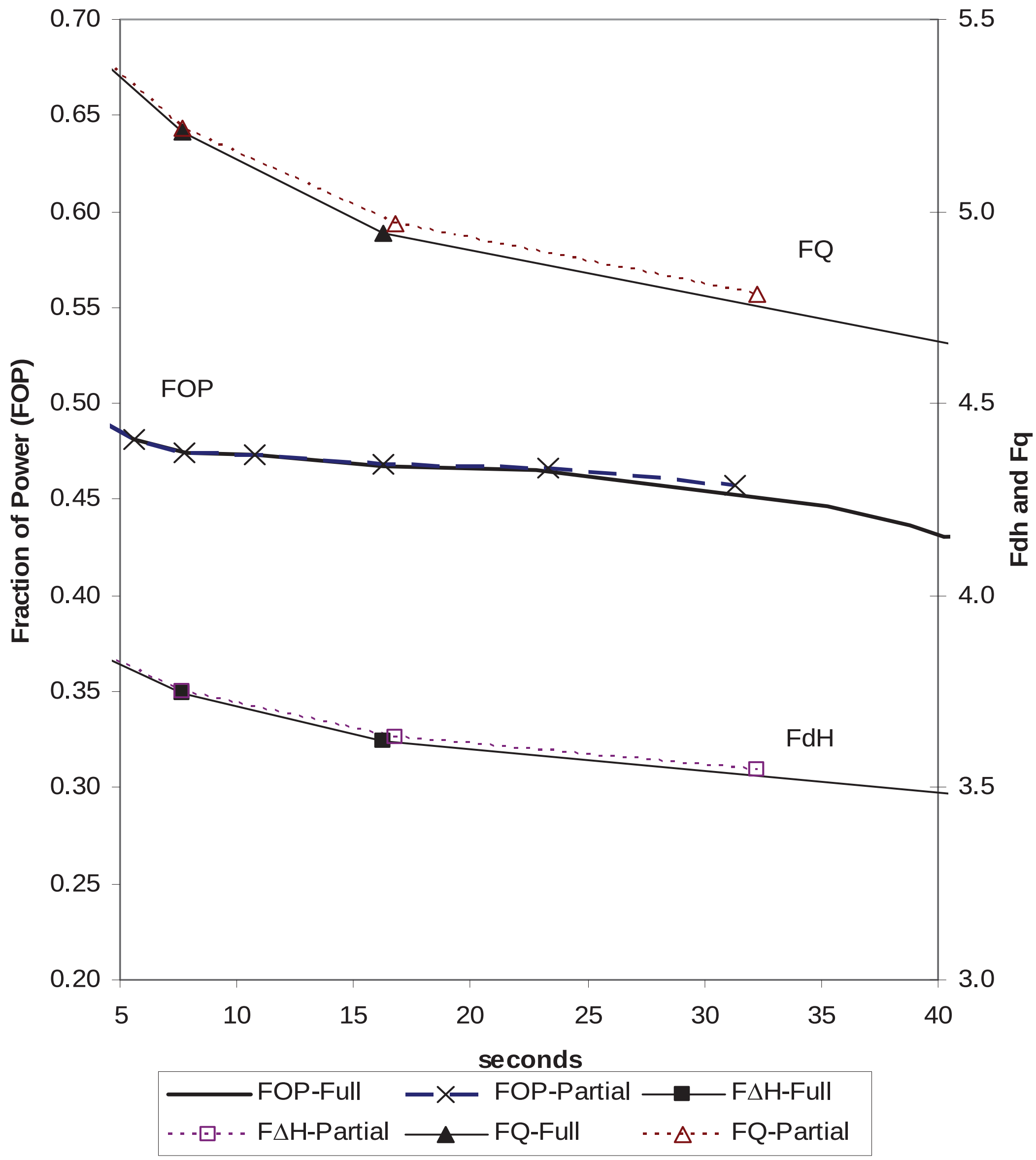


Figure 2.8.5.4.6-4 RELAP5/MOD2 Results for BOC 20% Power

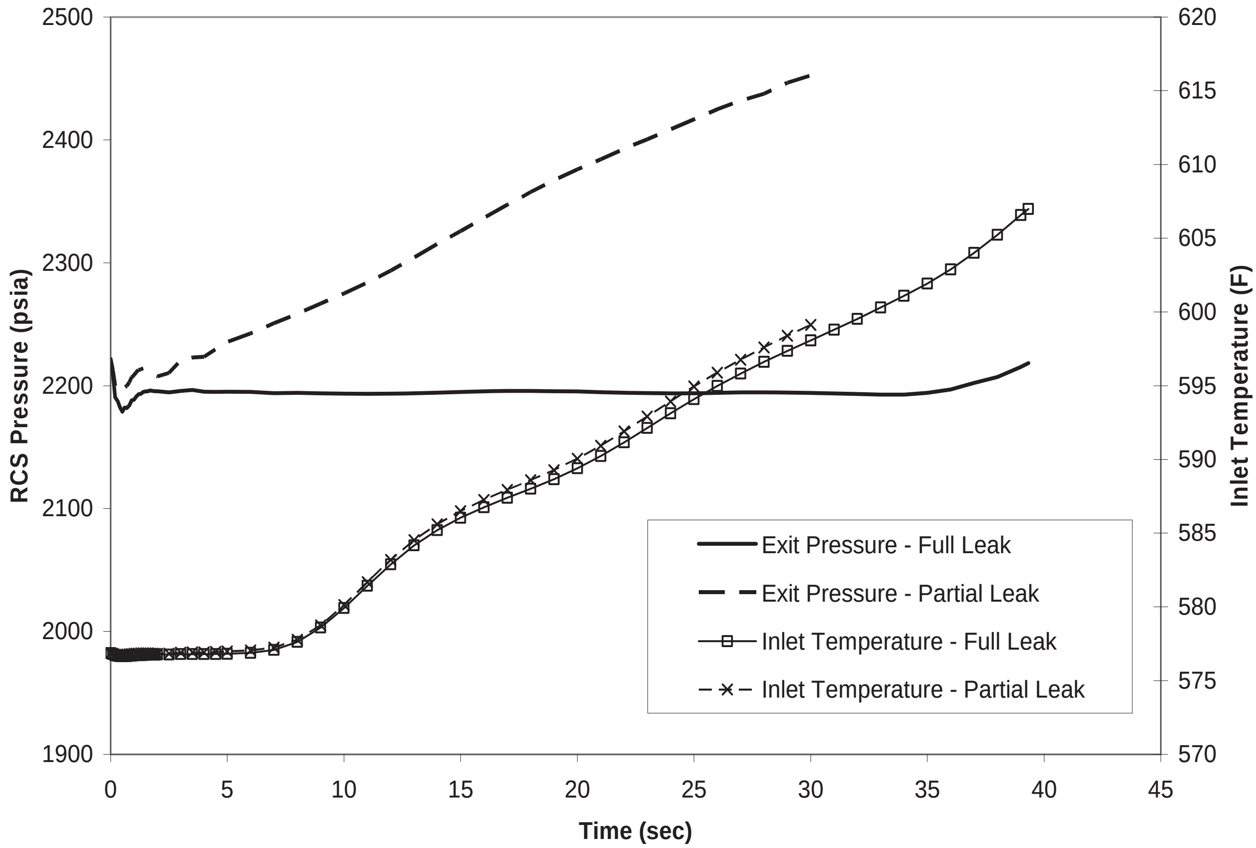


Figure 2.8.5.4.6-5 Minimum DNBR for BOC 20% Power

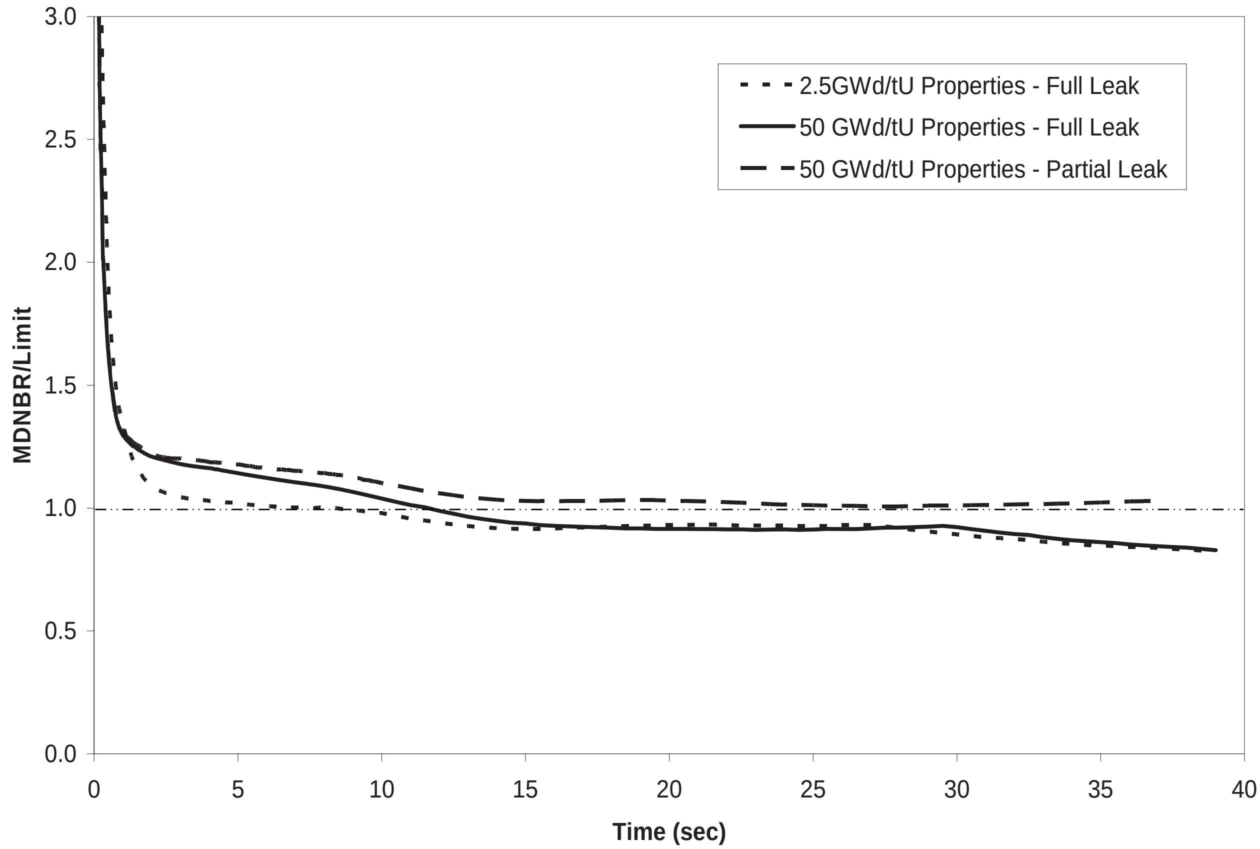


Figure 2.8.5.4.6-6 Fuel and Cladding Temperatures for BOC 20% Power

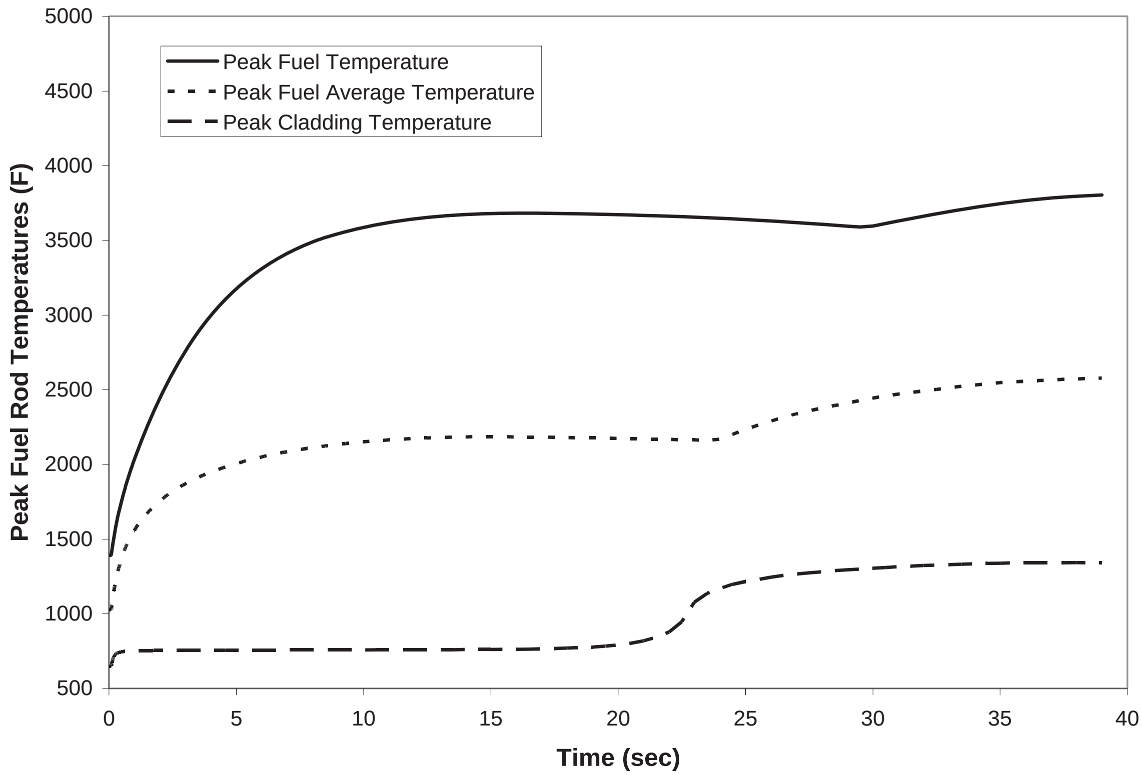


Figure 2.8.5.4.6-7 Peak Enthalpy Rise for BOC 20% Power

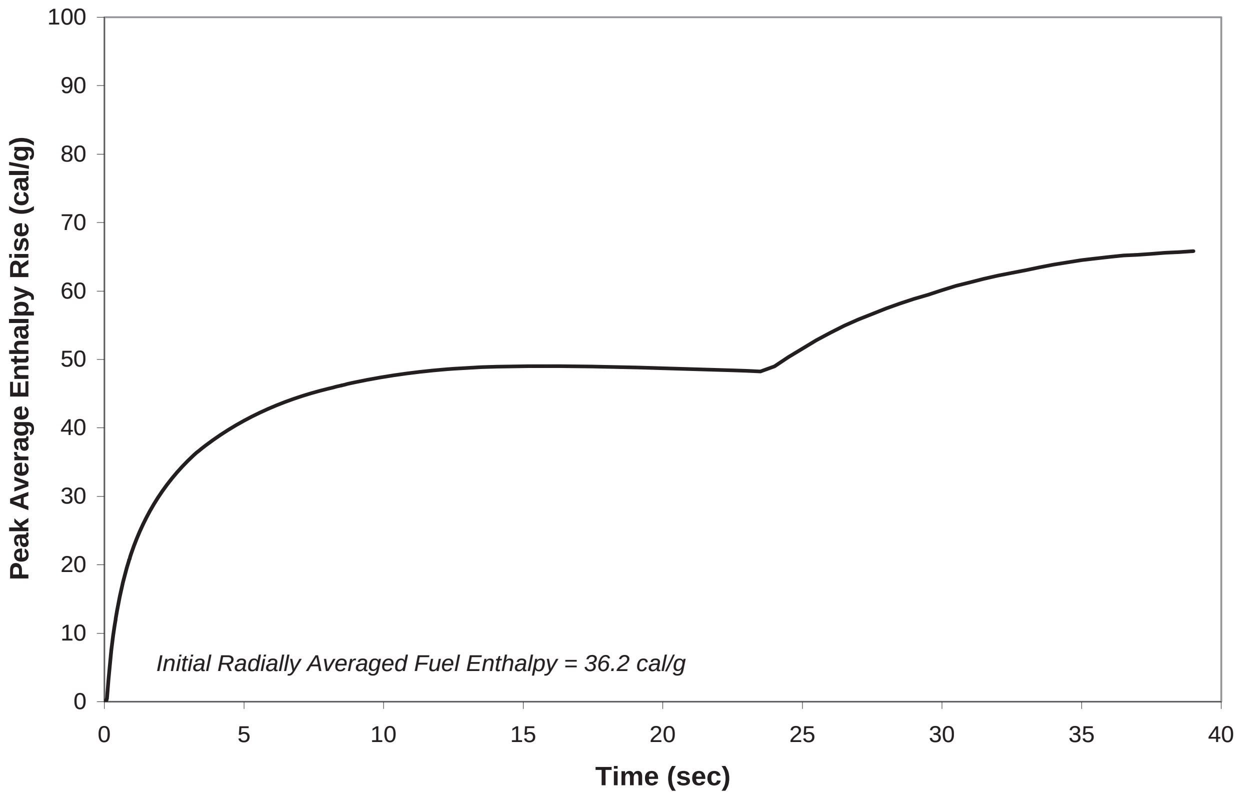


Figure 2.8.5.4.6-8 EOC HZP Transient

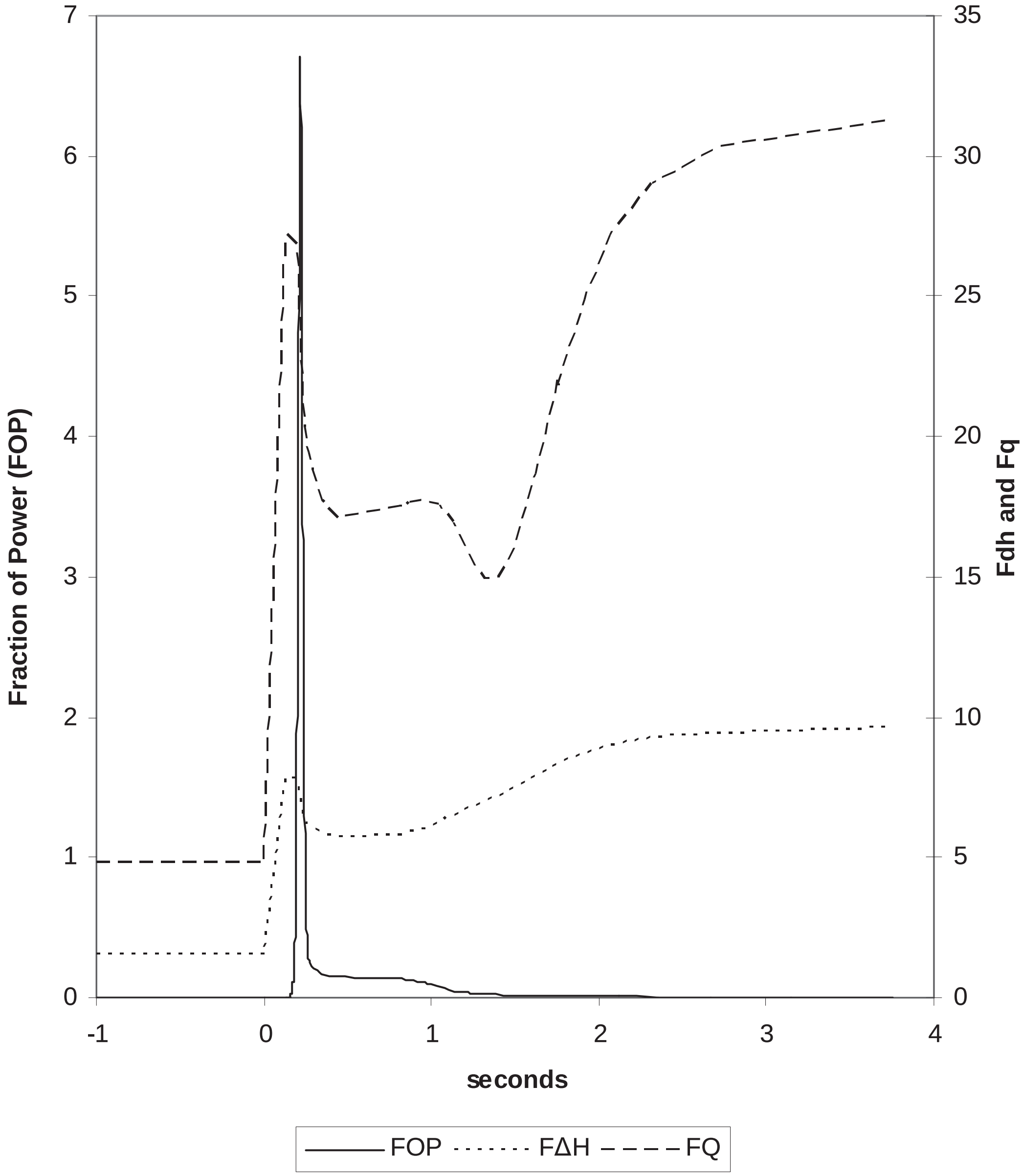


Figure 2.8.5.4.6-9 Minimum DNBR for EOC HZP

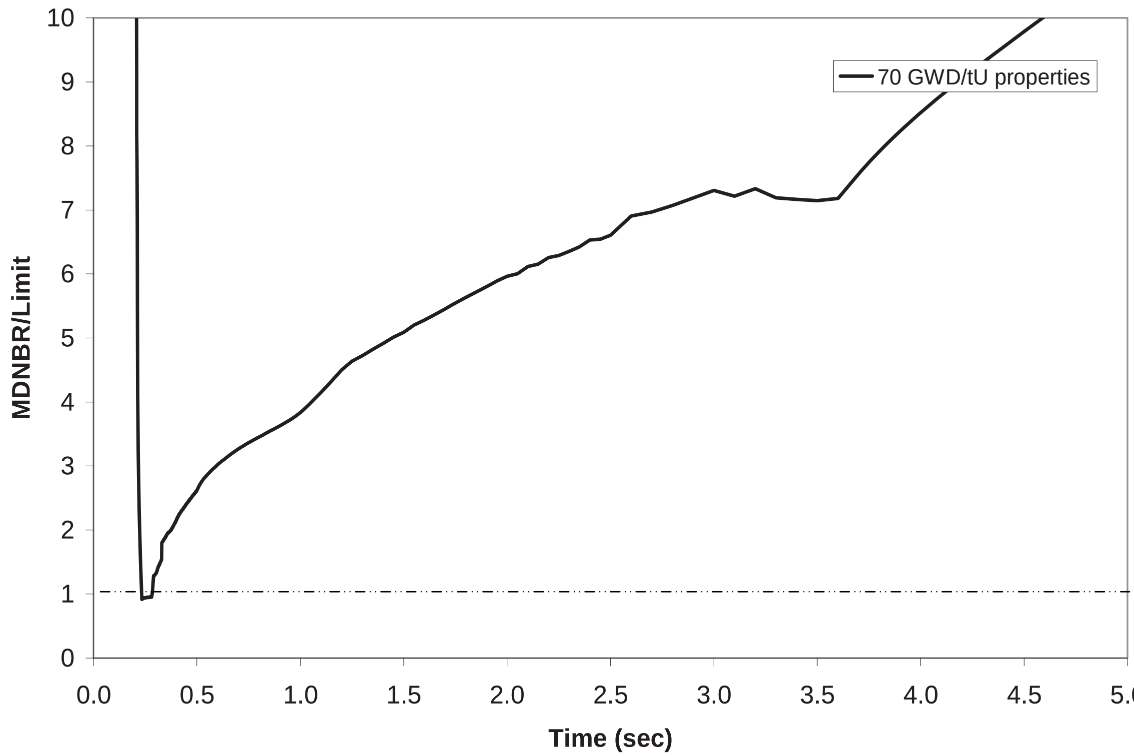


Figure 2.8.5.4.6-10 Fuel and Cladding Temperatures for EOC HZP

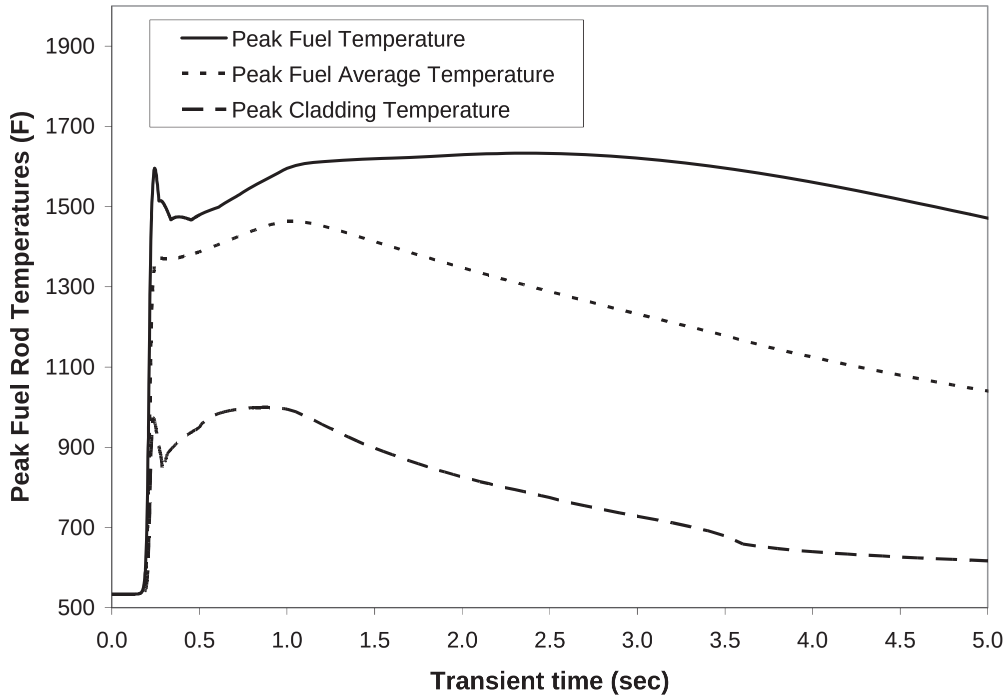
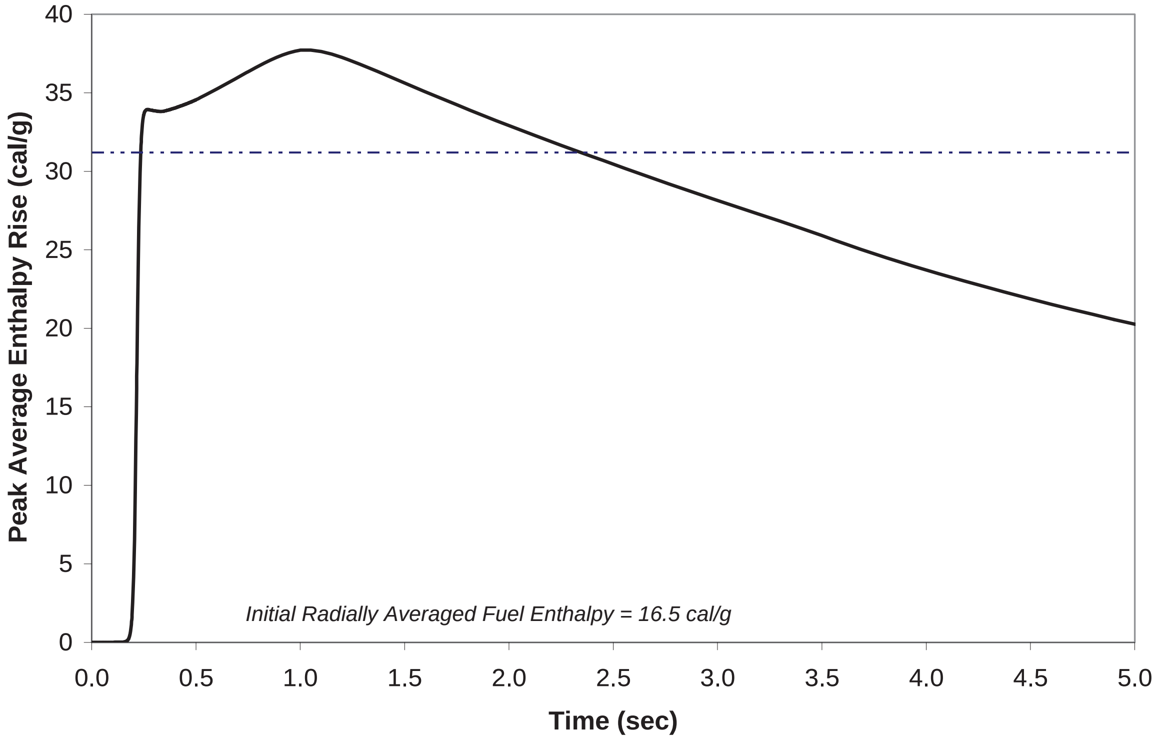


Figure 2.8.5.4.6-11 Peak Enthalpy Rise for EOC HZP



2.8.5.5 Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

2.8.5.5.1 Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the boron concentration and temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the Reactor Coolant System (RCS). Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The CR-3 review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses.

The NRC's acceptance criteria for Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory are based on:

- GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs);
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during AOOs; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

Inadvertent operation of the Emergency Core Cooling System (ECCS) and Chemical and Volume Control System (CR-3 Makeup System) malfunction events are not addressed in CR-3 FSAR Chapter 14 and are therefore, not analyzed for EPU consistent with CR-3 current licensing basis (CLB). As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design - insofar as it requires that the RCS be designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs [GDC-10];

- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary - insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during AOOs [GDC-15]; and
- FSAR Section 1.4.27, Redundancy of Reactivity Control - insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operation, including AOOs, SAFDLs are not exceeded. [GDC-26].

2.8.5.5.2 Technical Evaluation

Inadvertent operation of the ECCS and CR-3 Makeup System malfunction events are not analyzed for EPU consistent with CR-3 CLB.

2.8.5.5.3 Conclusion

Inadvertent operation of the ECCS and CR-3 Makeup System malfunction events are not analyzed for EPU consistent with CR-3 CLB.

2.8.5.5.4 References

None.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of Pressurizer Pressure Relief Valve

2.8.5.6.1.1 Regulatory Evaluation

The Inadvertent Opening of a Pressure Relief Valve results in a reactor coolant inventory decrease and a decrease in Reactor Coolant System (RCS) pressure. A reactor trip normally occurs due to low reactor coolant system pressure. The CR-3 review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses.

The NRC's acceptance criteria for Inadvertent Opening of a Pressure Relief Valve are based on:

- GDC-10, insofar as it requires that the RCS be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs);
- GDC-15, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operations, including AOOs; and
- GDC-26, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded.

CR-3 Current Licensing Basis

Inadvertent opening of the Pressurizer Pressure Relief Valve event is not addressed in CR-3 FSAR Chapter 14 and is therefore, not analyzed for EPU consistent with CR-3 current Licensing basis (CLB). As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3. The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.6, Reactor Core Design, insofar as it requires that the reactor coolant system (RCS) be designed with appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operations including anticipated operational occurrences (AOOs) [GDC-10];
- FSAR Section 1.4.9, Reactor Coolant Pressure Boundary, insofar as it requires that the RCS and its associated auxiliary systems be designed with sufficient margin to ensure that the design of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operations, including AOOs;[GDC-15]; and
- FSAR Section 1.4.27, Redundancy of Reactivity Control, insofar as it requires that a reactivity control system be provided, and be capable of reliably controlling the rate of reactivity

changes to ensure that under conditions of normal operations, including AOOs, SAFDLs are not exceeded [GDC-26].

2.8.5.6.1.2 Technical Evaluation

Inadvertent opening of the Pressurizer Pressure Relief Valve event is not analyzed for EPU consistent with CR-3 CLB.

2.8.5.6.1.3 Conclusion

Inadvertent opening of the Pressurizer Pressure Relief Valve event is not analyzed for EPU consistent with CR-3 CLB.

2.8.5.6.1.4 References

None.

2.8.5.6.2 Steam Generator Tube Rupture

2.8.5.6.2.1 Regulatory Evaluation

A steam generator tube rupture (SGTR) event causes a direct release of radioactive material contained in the primary coolant to the environment through the ruptured SG tube and main steam safety or atmospheric relief valves. Reactor protection and Engineered Safety Features (ESFs) are actuated to mitigate the accident and restrict the offsite dose to within the guidelines of 10 CFR Part 50.67. The CR-3 review covered (1) postulated initial core and plant conditions, (2) method of thermal and hydraulic analyses, (3) the sequence of events (assuming offsite power available), (4) assumed reactions of reactor system components, (5) functional and operation characteristics of the reactor protection system, (6) operator actions consistent with the plant's emergency operating procedures (EOPs), and (7) the results of the accident analyses.

The NRC's acceptance criteria are focused on the thermal and hydraulic analysis of the SGTR in order to:

- Determine whether 10 CFR Part 50.67 is satisfied with respect to radiological consequences, which are discussed in Section 2.9.2; and
- Confirm that the faulted SG does not experience an overfill. Preventing SG overfill is necessary in order to prevent failure of the steam lines.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The applicable CR-3 specific criterion is FSAR 1.4.70, Control Of Releases Of Radioactivity To The Environment, insofar as it sets standards for radiological consequences of postulated accidents, [10 CFR Part 50.67]. The radiological dose analyses for this event are addressed in Section 2.9.2, Radiological Consequences Analyses, which demonstrates that the radiological consequences do not exceed the allowable limits prescribed by 10CFR50.67 and Regulatory Guide 1.183.

FSAR Section 14.2.2 states it is possible that isolation of the affected SG might be required if any Tube Rupture Alternate Control Criteria (TRACC) are reached. The use of TRACC criteria during the mitigation of a SGTR is addressed in a letter from the NRC to B&W Owners Group Operator Support Committee, "Completion of Review of the Babcock & Wilcox Emergency Operating Procedures Guidelines (TAC No. M54946)," dated November 5, 1999. It is anticipated that, for the design basis tube rupture event, the borated water storage tank (BWST) low level SG isolation limit would be reached prior to establishing Decay Heat Removal System (DHRS) cut-in conditions. Thus, isolation of the affected SG could be required at some time after the hot leg temperature and pressure are reduced below the saturation point corresponding to the atmospheric dump valve (ADV) or main steam safety valve (MSSV) lift pressure. This SG isolation is consistent with the bases in the current B&W Unit EOP Technical Bases Document 74-1152414-10. If the affected SG is isolated, the leak would fill the affected SG to the main Steam isolation valves. The piping has been evaluated for the associated loads. As such, prevention of SG overfill has been determined not applicable to CR-3.

Additionally, FSAR Section 14.2.2.2 provides the criterion that the SGTR event must not result in additional tube failures and further degradation of the integrity of the reactor coolant pressure boundary (RCPB) caused by the effects of temperature gradients (i.e., thermally induced tube loadings). FSAR 14.2.2.2 also provides the current licensing basis assumption crediting the turbine bypass valves (TBVs) and main condenser for SGTR mitigation.

2.8.5.6.2.2 Technical Evaluation

<u>Introduction</u>

The CR-3 SGTR analysis assumes that a double-ended rupture of one SG tube allows activity to be discharged directly into the secondary system. In the event of a SGTR, activity contained in the reactor coolant is released to the secondary system. Some of the radioactive noble gases and iodine are released to the atmosphere through the MSSVs and ADVs until steam flow rate decreases to within the capacity of the turbine bypass valves (TBVs) at which point radioactive release is routed through the condenser air removal system.

The CR-3 SGTR licensing basis analysis (pre-EPU) is a calculation of primary-to-secondary leakage based on a constant leak rate of 435 gpm (60.38 lbm/sec). The constant leak rate is maintained throughout a plant cooldown that assumes the steaming of both SGs until the plant is cooled down to the DHRS cut-in temperature of 280°F (conservatively chosen as 8 hours) at which point the event is considered terminated. The leak flow rate is conservative because it does not credit the decrease in the leakage rate with RCS depressurization, the secondary side pressurization following reactor trip and turbine trip, nor the hydraulic losses through the SG tube. Offsite power is assumed to be available in the current licensing basis for the plant. The radiological dose evaluations continue, at EPU conditions, to conservatively assume a constant leak rate of 435 gpm for offsite and control room dose and conservatively assume steaming both SGs for 24 hours.

In support of the EPU, the mass release analysis was analyzed with a system thermal-hydraulic code. The SGTR analyses used the methodology defined in BAW-10193 (Reference 1). The RELAP5/MOD2-B&W code (Reference 2) was used in conjunction with the methodology, which also incorporated conservative setpoints and capacities to arrive at a conservative result. The SGTR thermal-hydraulic analysis and subsequent dose analysis assume offsite power is available and does not consider a limiting single failure consistent with CR-3 current licensing basis.

Description of Analyses and Evaluations

The SGTR sequence of events assumes the double-ended rupture of a SG tube occurs at full power and resulting calculated SG tube break flow maximizes at approximately 286 gpm (38.60 lbm/sec). The break flow in this thermal-hydraulic analysis is derived from the RELAP5/MOD2-B&W code and is conservative with respect to the flow rate used in the radiological dose evaluations. Reactor Coolant System (RCS) pressure decreases until a reactor trip occurs on RCS low pressure. The turbine trips as a result of the reactor trip and the turbine stop valves (TSVs) close rapidly. Closure of the TSVs causes the main steam line pressure to increase and open the MSSVs. The MSSVs were modeled to pop fully open upon reaching the setpoint (no valve accumulation) in order to maximize the release through the MSSVs into the atmosphere. The Main Feedwater (MFW) System runs back and automatically maintains SG liquid levels at the low level limit setpoint. The TBVs are modeled to respond in order to control secondary pressure post-trip, and as a consequence, the MSSVs reseat and steam release from the affected steam generator is directed to the condenser via the TBVs. The automatic opening of the ADVs post-trip is not explicitly modeled in the SGTR thermal-hydraulic analyses, however all post-trip steam release to the atmosphere is captured via the MSSVs and is included in the radiological consequence evaluation. Post-reactor trip, the RCS pressure continues to decrease until the Engineered Safeguards Actuation System (ESAS) RCS low pressure setpoint is reached, at which point high pressure injection (HPI) is actuated. Following ESAS actuation, HPI flow is throttled as necessary to control pressurizer level and subcooling margin (SCM). Additionally, it is assumed that RCS SCM is actively controlled using pressurizer spray as needed. Following the reactor trip, RCS cooldown is established with a target cooldown rate of 100°F/hr. Once the RCS is below 500°F, the thermal-hydraulic analysis is terminated. Although the SGTR thermal-hydraulic analysis was not extended until the RCS reached the DHRS cut-in temperature, RCS cooldown would continue by steaming both SGs using the TBVs.

The key input parameters and initial conditions used in the SGTR analysis are as follows:

- The initial core power level was set to 3026.1 MWt (100.4% of 3014 MWt), consistent with the increased power level planned in conjunction with the EPU. A conservative reactor coolant pump (RCP) heat was also included.
- The analysis modeled the reactor to be at hot full power conditions with a nominal average temperature of 582°F, consistent with the planned increase in T_{AVG} associated with the EPU. The initial hot leg pressure is assumed to be the nominal value of 2170 psia. Nominal values are in accordance with the methodology described in Reference 1.
- The initial reactor coolant volumetric flow was equal to the minimum DNB flow rate (374,880 gpm) in accordance with the methodology described in Reference 1.
- The initial pressurizer level was modeled as nominal (220 inches) in accordance with the methodology described in Reference 1.
- Pressurizer heaters were modeled since heat addition to the pressurizer helped maintain RCS pressure and delay a reactor trip (1638 kW).
- Reactor trip was modeled to occur on a nominal low RCS pressure setpoint minus uncertainty (1893.95 psia), since delayed reactor trip is conservative for the event.

- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- The event is not sensitive to the choice of reactivity parameters. However, a bounding Doppler temperature coefficient and moderator coefficient were chosen to minimize the decrease in power prior to reactor trip and to provide a conservative rate of power decrease post-trip.
- Maximum safety injection (HPI flow from 2 pumps) was modeled to initiate at the ESAS low RCS pressure trip plus uncertainty of 1714 psia. Earlier initiation and maximum safety injection flow maximizes the primary-to-secondary leakage through the ruptured tube. Boron reactivity contribution was not credited.
- The MSSVs were modeled to lift at the nominal setpoints minus 3% lift tolerance and no accumulation to maximize releases to the atmosphere. A nominal blowdown value of 5% was used.
- Makeup and letdown were modeled since they help maintain the RCS pressure and postpone the reactor trip. The flow from one makeup pump was modeled on the loop with the pressurizer. The letdown flow was modeled at a constant rate of 75 gpm on the loop opposite the pressurizer. Makeup and letdown were isolated after the ESAS trip.
- TBVs were modeled to open following the reactor trip.
- The following manual operator action times were assumed and based on current CR-3 procedures:
    - Upon receipt of the ESAS signal (approximately 34 minutes after event initiation), the HPI flow is assumed to be throttled to control RCS pressure, pressurizer level, and SCM. Based on the criteria defined, the first occurrence of HPI throttling did not occur until approximately 18 minutes after the ESAS signal was received (or approximately 52 minutes after event initiation);
    - At 40 minutes following the reactor trip, it was assumed that pressurizer spray was available to control RCS SCM.
    - At 45 minutes following the reactor trip, RCS cooldown is assumed by steaming both SGs to the condenser via the TBVs.

The acceptance criteria for the SGTR event are that the public radiological doses must not exceed the allowable limits prescribed by 10 CFR 50.67 and Regulatory Guide 1.183, and that the event must not result in additional tube failures and further degradation of the integrity of the RCPB caused by the effects of temperature gradients (i.e., thermally induced tube loadings). Acceptable maximum tube / shell temperature difference during heat-up conditions is +90°F (tube is hotter); and during cool-down conditions is -140°F (tube is colder).

Results

The sequence of events for the SGTR is provided in Table 2.8.5.6.2-1. Even though It is anticipated that the BWST level would reach the emergency operating procedure TRACC limit prior to establishing DHRS cut-in conditions requiring isolation of the affected SG, the specific dose evaluation and acceptance criteria for the SGTR presented in Section 2.9.2 conservatively assume radioactive release from the affected SG to the condenser via the TBVs will continue for 24 hours.

The thermal-hydraulic analysis was terminated when RCS temperature reached 500°F, which was within 2 hours for the analyzed event. Although the SGTR thermal-hydraulic analysis was not extended until the RCS reached the DHRS cut-in temperature, RCS cooldown would continue by steaming both SGs using the TBVs.

Per CR-3 procedures, RCS pressure will be reduced using any available means, including pressurizer spray, High pressure auxiliary spray, opening of the PORV or pressurizer high point vent valves. Though RCS depressurization was not modeled in the SGTR dose analysis (conservatively maximizing the break flow), both SGs are assumed to continue steaming until DHRS cut-in conditions are reached.

An additional CR-3 acceptance criterion for the SGTR analysis stipulates that additional tube failures and loss of RCPB integrity resulting from the temperature gradients (thermally induced tube loading) shall not occur. The criterion preventing additional tube failures is evaluated by confirming that the SG tube-to-shell temperature difference remains bounded at EPU conditions by normal heatup and cooldown limits. For the SGTR, the maximum tube-to-shell difference occurs for the tensile condition (shell hotter than tubes). The SGTR maximum tensile tube-to-shell difference observed in the analysis (30°F) was less than the normal cooldown limit for the SG. In the event that SG isolation leads to overfill, the idle-loop tube-to-shell temperature difference would not exceed normal heatup/cooldown limits. The tube temperature would trend down with RCS temperature. With the SG otherwise isolated, secondary pressure follows the RC pressure and secondary temperature is at saturation. Shell temperature would lag due to the time to cool the thick metal shell, but would also trend down since secondary pressure / temperature are decreasing (as saturation temperature decreases). As a result, tube integrity is assured throughout the SGTR event.

2.8.5.6.2.3 Conclusion

CR-3 has reviewed the analysis of the SGTR accident and concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical methods and approved computer codes. CR-3 further concludes that the assumptions used in this analysis are conservative and that the event does not lead to additional tube failures and loss of reactor coolant boundary integrity. Therefore, CR-3 finds the proposed EPU acceptable with respect to SGTR event.

2.8.5.6.2.4 References

1. BAW-10193PA-00 (Proprietary), “RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors”.

2. BAW-10164PA-06 (Proprietary), “RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis”.

Table 2.8.5.6.2-1: SGTR Sequence of Events

Event	Time (sec)	Time (min)
Event Initiated	0.0	0.0
Reactor Trip on low RCS pressure	2019.34	33.66
Control Rods Insert and Turbine Trip	2019.96	33.67
MSSVs first open affected/unaffected loop	2021.14/2021.16	33.69 / 33.69
ESAS Low RCS Pressure Signal	2032.60	33.88
HPI Flow Starts	2032.60	33.88
MSSVs last close affected/unaffected loop	2109.6*/2100.36	35.2 / 35.0
RCS cooldown begins using TBVs	4719.35	78.66
RCS temperature < 500°F. Thermal-hydraulic analysis terminated.	6746.10	112.44

- An affected-loop MSSV opens briefly due to a model perturbation after initiation of the controlled cooldown via the TBVs. This occurs at 4733.52 seconds after transient initiation.

2.8.5.6.3 Emergency Core Cooling System and Loss-of-Coolant Accidents

2.8.5.6.3.1 Regulatory Evaluation

A Loss-of-Coolant Accident (LOCA) is a postulated accident that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary (RCPB) at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and Emergency Core Cooling Systems (ECCS) are provided to mitigate these accidents. The CR-3 review covered:

- the licensee's determination of break locations and break sizes;
- postulated initial conditions;
- the sequence of events;
- the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients;
- calculations of peak cladding temperature (PCT), peak local oxidation of the cladding, whole core hydrogen generation, changes in core geometry, and long-term cooling;
- functional and operational characteristics of the reactor protection and ECCS systems; and
- operator actions.

The NRC's acceptance criteria for Emergency Core Cooling System and Loss-of-Coolant Accidents are based on:

- GDC-4, insofar as it requires that the structures, systems and components important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer;
- GDC-27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;
- GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented;
- 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; and

10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 pre-date the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the CR-3-specific criteria related to LOCA analyses:

- FSAR Section 1.4.23, Protection against Multiple Disability for Protection Systems, insofar as it requires that the structures, systems and components important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer; [GDC-4]

- FSAR Section 1.4.28, Reactivity Hot Shutdown Capability; FSAR Section 1.4.29, Reactivity Shutdown Capability; FSAR Section 1.4.30, Reactivity Holddown Capability, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained. [GDC-27]

- FSAR Section 1.4.37, Engineered Safety Features Basis for Design, FSAR Section 1.4.41, Engineered Safety Features Performance Capability; FSAR Section 1.4.42, Engineered Safety Features Components Capability; FSAR Section 1.4.44, Emergency Core Cooling Systems, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented. [GDC-35]

Current licensing applications for CR-3 are performed based on the AREVA NP LOCA Evaluation Model (BWNT LOCA EM) topical report, BAW-10192-A (Reference 1) insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance [10 CFR § 50.46], and insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA. [10 CFR Part 50, Appendix K]

2.8.5.6.3.2 Technical Evaluation – ECCS and LOCA Core Cooling

Introduction

As described in FSAR Section 14.2.2.5, a LOCA occurs as the result of postulated rupture of the primary coolant piping. The LOCA analyses are classically divided into two groups: Large Break (LBLOCA) and Small Break (SBLOCA). FSAR Section 14.2.2.5.6 describes the LBLOCA characterizations and analyses at the pre-EPU core power level of up to 2619 MWt including heat balance uncertainty. FSAR Section 14.2.2.5.7 describes the SBLOCA characterizations and analyses at the pre-EPU core power. The 10 CFR 50.46 acceptance criteria are met, with the limiting PCTs as 1994°F and 1535°F for the LBLOCA and SBLOCA, respectively as reported in Reference 7. The FSAR sections for both groups distinguish the event progression and the key phenomena for potentially limiting break sizes in each respective class.

FSAR Section 14.2.5.2 indicates that LOCA analyses are used for two purposes. The primary purpose is to qualify the ECCS performance as demonstrated by compliance with the 10 CFR 50.46 acceptance criteria at the maximum allowed peaking limits. The secondary purpose is to use the maximum allowed peaking limits contained in the LOCA analyses to determine the plant operating limits, which are a portion of the limits used to define operating windows for rod insertion and axial imbalance.

To accommodate the power increase, several modifications are being implemented. Those pertinent to this section include an automatic Reactor Coolant Pump (RCP) trip within one minute of losing subcooling margin via the new Inadequate Core Cooling Mitigation System (ICCMS). Modifications to the Low Pressure Injection (LPI) system for flowpath cross-tie and Hot Leg Injection (HLI), and to the Atmospheric Dump Valves (ADVs) with the Fast Cooldown System (FCS) for rapid secondary side pressure reduction and post-blowdown control are also being implemented. The ICCMS actuates the FCS to trigger a secondary side blowdown that reduces primary side pressure, thus maximizing ECCS. The functional descriptions of these modifications are provided later in the Mitigating Systems section, and detailed design descriptions are provided in Appendix E, Major Plant Modifications. The LBLOCA and SBLOCA EPU power analyses with these modifications are discussed below and compared to the pre-EPU analyses results.

<u>LBLOCA EPU Results</u>

For the CR-3 EPU, eleven new LBLOCA LHR analyses were completed with five at Beginning of Life (BOL), five at Middle of Life (MOL), and one at End of Life (EOL). These analyses included the power level increase, increase in the nominal RCS average temperature, SGs with five percent tube plugging, and various other key input parameter changes listed in Table 2.8.5.6.3-1. The new ADV modifications to initiate FCS operation were not credited in the LBLOCA analyses. The FCS actuation time (10 minutes) is well after the reflooding phase is complete and the core is quenched. The LBLOCA analyses also included an updated minimum containment pressure analysis described in Section 2.6.6, Pressure Analysis for ECCS Performance Capability.

Overall, the EPU LBLOCA analyses yielded a maximum PCT of 1947°F. The maximum cladding local oxidation was less than 3 percent, while the core-wide whole core hydrogen generation was less than 0.2 percent. Each of these three maxima has ample margin to the PCT limit of 2200°F, local oxidation limit of 17 percent, and whole core hydrogen generation limit of 1 percent.

The LBLOCA PCT for the EPU is similar to, but less than that of the current (or pre-EPU) analyses. This is an outcome resulting from several factors. One is the application of more restrictive limits for the EPU condition. Figure 2.8.5.6.3-1 shows that the relative maximum peaking of ~2.6 is lower than the ~2.9 for the pre-EPU power. The EPU power level also increases the rod average power and power history envelope, which cause the MOL burnup to be limited to 40 GWd/mtU for the EPU, in contrast to the pre-EPU MOL burnup of 45 GWd/mtU. The PCT change also reflects a beneficial effect of several plant input changes including the reduction in the unit-average Steam Generator Tube Plug (SGTP) from 20% for the pre-EPU analyses to 5% for the EPU analyses.

<u>SBLOCA EPU Results</u>

FSAR Sections 14.2.2.5.7.2 – 4 provide a description of the transient evolution for the Cold Leg Pump Discharge (CLPD), HPI line, and CFT line breaks at the pre-EPU power level. These breaks are generally characterized in categories as described in FSAR Section 14.2.2.5.7. These breaks produce the most limiting results with a Loss of Offsite Power (LOOP) imposed at the time of reactor trip and failure of an Emergency Diesel Generator (EDG) to start. For the analyzed power of 2619 MWt including the heat balance uncertainty, the CLPD spectrum of break sizes analyzed with the 9.5-ft axial power shape predicted core uncovering for break sizes of 0.07 to 0.5 ft^2 with the calculated or estimated PCTs versus break size as shown in Figure 2.8.5.6.3-2. The smaller break sizes, that depressurize the RCS slowly such that the Core Flood Tank (CFT) and LPI may not actuate for a considerable time after break opening, rely on primary-to-secondary heat transfer and on HPI flow to keep the core covered at the pre-EPU power level. With the more conservative 11-ft axial peak at 2619 MWt, the highest PCT for the pre-EPU analyses is 1535°F for the CLPD break.

A power uprate increases the core decay heat power and increases the core boiloff rates, slowing the RCS depressurization rate and slightly reducing the ECCS flow. Without any plant modifications, the EPU power increase would lead to more core uncovering and higher PCTs for the breaks that the mixture level drops below the top of the heated core region. In addition, smaller SBLOCA break sizes may result in core uncovering with fuel pin heatup at the EPU power level.

The EPU power increase requires additional ECCS flow to remove the core decay heat. The primary means of increasing the ECCS flow for the EPU is accomplished by the secondary side ADV replacement and development of the FCS described later in the Mitigating Systems section. For SBLOCAs, the cooldown induced by the FCS and EFW heat removal depressurizes the RCS to approximately the secondary side ADV modulation pressure, increasing HPI flow and initiating some CFT liquid discharge. This increased HPI and CFT flow compensates for the increased core power. The secondary side depressurization can cause the existing Feed Only Good Generator (FOGG) circuitry, which protects the plant from overcooling by terminating EFW to a potentially faulted SG, to occasionally and only briefly interrupt EFW to whichever SG is below the FOGG pressure setpoint of 600 psig, or is at a pressure 125 psid lower than the other SG when SG pressures are below 600 psig. The FOGG signal resets, restoring EFW, when the SG pressure is above 600 psig, or the difference decreases below 125 psid at SG pressures less than 600 psig. The net effect of the interrupted EFW was found to be negligible on the transient progression and did not adversely affect compliance with 10 CFR 50.46 acceptance criteria.

In addition to the FCS, the LPI System was modified to include an LPI cross-tie flowpath for mitigating the CFT line breaks, as described in the Mitigating Systems section. The ADV/FCS, LPI, and HPI changes were considered in the analyses along with other key inputs described in Table 2.8.5.6.3-1 through Table 2.8.5.6.3-4, and the key operator actions discussed in Table 2.8.5.6.3-5.

Additionally, a full power SBLOCA spectrum was analyzed with the flow from two HPI pumps, but without credit for the FCS. This study is germane to the development of the ICCMS, which determines whether adequate HPI is available before initiating the FCS as mentioned in the Mitigating Systems section, and as described in detail in Appendix E. When two HPI pumps are in service, the uncertainty- adjusted total indicated HPI flow should exceed the one-pump flow by at least 30 percent and this additional ECCS obviates the need for actuating FCS. These analyses show that the additional flow is sufficient to keep the limiting PCT less than the EPU FCS one-pump results described in the SBLOCA sections below.

Finally, a new consideration for operating at the EPU power level is the new Improved Technical Specification (ITS) 3.7.20, along with ITS 3.3.19 and 3.3.20, that set the Limiting Conditions of Operation for the new safety-related ADVs and FCS. In the event that either the ADVs or FCS is inoperable, the ITS requires that power be reduced to 2609 MWt, the pre-EPU power level that corresponds to an analyzed power level of 2619 MWt. At this power level, at EPU RCS conditions (e.g., higher RCS average temperature, etc.), analyses show that the power reduction is adequate to offset the loss of FCS functionality.

Cold Leg Pump Discharge (CLPD) Pipe SBLOCAs

For the CR-3 EPU, a spectrum of small CLPD LOCAs was analyzed to demonstrate compliance to 10 CFR 50.46 in accordance with BAW-10192-A. Break sizes considered range from the smallest break that exceeds the makeup system capacity up to the largest break size that does not cause the fuel pin cladding to undergo departure from nucleate boiling (DNB) during the initial blowdown. By analyzing and evaluating this break range, the adequacy and interdependencies of the EFW SG heat removal, HPI pumped injection rate and flow split between injection lines, CFT fill pressure and liquid inventory delivery rate, LPI pumped injection rate and flow split between injection lines, and requisite actions to mitigate the event can be assessed.

The smallest break size that exceeds the makeup flow only has a very small net outflow of a few gpm. For these scenarios, the break flow plus RCS outflow from letdown and RCS leakage only slightly exceed the net inflow of the normal makeup and Reactor Coolant Pump (RCP) seal injection inleakage. Since the RCS normally holds over 80000 gallons of liquid, a very small leak of a few gpm will take hours to days to deplete the RCS to the point that the core could have insufficient liquid to keep it continuously covered with a two-phase mixture capable of removing the core decay heat. For these scenarios, the break cannot remove the core generated energy so the SG provides nearly all the core heat removal via EFW and steaming through the Main Steam Safety Valves. If the EFW is inadequate, the RCS may repressurize to the PORV and pressurizer safety valve lift pressure. For these smaller sized breaks, once the HPI is initiated, the break flow is functionally equivalent to the HPI delivery rate and there is little RCS liquid lost out of the break. The subsequent SG depressurization with the FCS may not be very effective for the smallest break sizes because the loop flows could be interrupted and the SG tube levels may not be low enough to achieve boiler condenser cooling.

As the CLPD RCS break size gets larger, the net outflow increases and the break energy relief also increases. There is less reliance on primary-to-secondary heat transfer. With higher RCS inventory loss rates the HPI flow capacity and flow split can be challenged for transients that remain at higher RCS pressures. However, the FCS depressurizes the secondary and induces further primary-to-secondary heat transfer that in turn depressurizes the RCS to increase the HPI flow and obtain some CFT flow. For these break sizes, EFW, HPI, and CFT work together to mitigate the consequences of the LOCA. Use of the FCS at the EPU power level will keep some of the smaller break sizes from uncovering, but there will be some with partial core uncovering and PCT ascensions for these intermediate SBLOCA break sizes.

The larger SBLOCA break sizes can depressurize the RCS below the CFT fill pressure (~600 psig) and below the pressure that the ADVs are modulated (≤350 psig). These break sizes remove all the core decay heat via the break so primary-to-secondary heat transfer has little to no effect on the event. The HPI, CFT, and longer-term LPI flows manage the RCS inventory loss and refill the system to limit the duration and magnitude of the core uncovering period.

The results of the EPU SBLOCA spectrum are shown in Figure 2.8.5.6.3-2 along with the pre-EPU results. The EPU analyses were performed with a 1.7 axial core power peak at 11-ft, credit for the FCS, and key changes in Table 2.8.5.6.3-1 from the pre-EPU analyses. For conservatism, the throttled HPI flows (Table 2.8.5.6.3-2A) were used for this break location.

Use of the FCS limited both the range of breaks that uncovered the core and the predicted PCTs that occur later than 10 minutes into the event. The EPU power level and higher axial peak to bound the middle- and end-of-cycle peaking are responsible for the increases in PCT for the break sizes greater than 0.2 ft^2. The observed dip in PCT for the 0.15 ft^2 EPU case was a direct result of credit for the FCS cooldown just prior to the time of PCT.

The EPU SBLOCA CLPD analyses show a maximum PCT of 1426°F at a break size of 0.13 ft^2, with a maximum cladding local oxidation less than 1 percent, and a core-wide whole core hydrogen generation rate of less than 0.03 percent. The PCT for the EPU is less than that for the pre-EPU maximum PCT, demonstrating the effectiveness of the FCS in augmenting the available HPI flow over a wide range of break sizes. The PCT, maximum local oxidation, and the whole core hydrogen generation show ample margin to the PCT limit of 2200°F, local oxidation limit of 17 percent, and whole core hydrogen limit of 1 percent.

As mentioned in the EPU SBLOCA Results section, an additional consideration is evaluating the SBLOCA in the event that FCS cannot be actuated, requiring a reduction in analyzed power to 2619 MWt. At this lower power level at EPU operating conditions, analyses show PCTs that do not exceed the full power CLPD PCT. Therefore, all of the CLPD break analyses for the EPU demonstrate compliance with 10 CFR 50.46.

<u>HPI Line Break</u>

The HPI Line Break scenario postulates the severance of one of the four HPI lines at its cold leg connection. The scenario presents a unique challenge due to the asymmetric pressure difference between the four HPI lines and the higher HPI flow fraction lost through the break. The HPI line break location was modeled in the HPI line providing normal makeup just upstream of the safe-end of the HPI nozzle. For the full-area HPI line pipe break, a separate set of ECCS hydraulic input is developed to simulate the reduction of HPI flow for this scenario. Both throttled and unthrottled HPI flows, shown in Table 2.8.5.6.3-2B, were investigated to address whether unthrottled HPI through the broken HPI line could result in more limiting consequences than those where the HPI lines remained throttled. For the HPI throttled scenario, credit for the FCS depressurized the RCS and kept the core continuously covered with a two-phase mixture level. Without any core uncovering, the PCT remained at the initial steady-state cladding temperature. For the unthrottled scenario, no core uncovering was predicted. Again, the FCS was able to maximize ECCS in this scenario.

For the part-power scenario wherein the FCS is not credited, the unthrottled HPI flow was used in order to minimize the available ECCS at the pressure range of this small break scenario. Without FCS, somewhat higher PCTs were observed, which were expected due to the elevated RCS pressure during the period of interest. However, the results of the analysis remain bounded by those of the CLPD breaks, and stay well within the acceptance criterion for PCT. Therefore, the HPI line break analysis for the EPU demonstrates compliance with 10 CFR 50.46.

<u>Core Flood Line Break</u>

The Core Flood Line Break (CFLB) scenario postulates a break in one of the two Core Flood lines leading to the reactor vessel nozzle. The effective RCS break area is limited to 0.44 ft^2 by the cross-sectional area of the nozzle insert described in FSAR 6.1.3.1.3. Each Core Flood line is also the flow path of each

LPI train into the reactor vessel. The limiting single failure for the off-site power available scenario is the loss of an Engineered Safeguards (ES) bus that supplies power to one ECCS train.

If a LOOP occurs, the limiting single failure is one EDG (or associated ES bus). With the failed EDG, one HPI and one LPI train are not available for coolant injection. Prior to the installation of the LPI cross-tie (described below in the Mitigating Systems section), the remaining LPI train, powered by the operating EDG, must also be assumed to be unavailable for coolant injection because it may be aligned to the broken CFT line. As a consequence, only one train of HPI and the inventory of one CFT was assured to be available prior to the LPI modifications for the EPU (see Mitigating Systems below). The modifications to the LPI system assure the availability of some LPI flow for the larger core flood breaks by providing a passive LPI crossflow path. With these modifications, the analyses of a spectrum of CFLBs show that the core remains continuously covered with a two-phase mixture and the PCT is set by the steady-state cladding temperature.

The CFLB has a large enough break size that the RCS and secondary are rapidly decoupled, diminishing the value of the FCS for these larger SBLOCAs. Additionally, no significant core uncovery was predicted for the pre-EPU CFLB at 2619 MWt. As such, no part-power SBLOCA analyses were performed for the scenario where FCS is not credited at EPU conditions, and the CFLB analyses for the EPU continue to demonstrate compliance with 10 CFR 50.46.

<u>Mitigating Systems</u>

The plant systems pertinent to LOCA analyses performed for the EPU are described in several sections of the FSAR. The Reactor Protection System (RPS) described in FSAR Section 7.1.2 ensures that the reactor is tripped when RCS pressure decreases below the low pressure trip setpoint. The capability of the reactivity control system to meet GDC-27 is discussed in FSAR Sections 3.2.4.3 and 7.2.2. For conservatism, a minimum control rod worth is used for SBLOCA analyses. For LBLOCA analyses, control rod insertion is not credited during the blowdown phase. From the start of core recovery through long-term core cooling, the combination of inserted control rods, injected boron, and appropriate operator actions ensure adequate subcritical margins.

For the ECCS described in FSAR Section 6.1, the sub-systems and key components are the HPI, LPI, borated water storage tank (BWST), and CFTs, reactor building spray and reactor building sump. The sump plays a significant role for assuring long-term core cooling, because as the BWST inventory depletes, actions will be taken to swap LPI suction from the BWST to the sump, recirculating the liquid inventory. Additionally, HPI flow suction may also be taken from the discharge of an LPI pump, a mode called piggyback, during this sump recirculation phase.

As part of the EPU effort, modifications to the LPI piping described in Appendix E are being implemented to cross-tie the two LPI trains to assure flow to the core during the postulated core flood line break described previously. An always-open cross-tie line between the two LPI flow paths (A&B) inside the Reactor Building will ensure an available flow path in the event either of the LPI flow paths is not available due to a limiting single failure of a pump, valve, or train of power. In conjunction with the installation of this line, a set of always-open throttle valves will be installed upstream of where each LPI train joins with its corresponding Core Flood line. These throttle valves will be set such that a portion of LPI will be provided to the reactor vessel through the intact LPI line during a CFLB when the RCS is within 69 psi of the containment pressure.

Also included in the LPI cross-tie modification is a new connection from the new cross-tie line to the decay heat drop line (DHDL) to facilitate post-LOCA hot leg injection (HLI) for boric acid precipitation control. Conditions may result in boron becoming concentrated in the reactor core following a CLPD break due to boiling in the reactor vessel. Under certain temperature and concentration conditions, boron could precipitate out of solution, potentially blocking flow and reducing core heat transfer. The two boron precipitation control methods currently licensed for CR-3 (FSAR Section 14.2.2.5.13) are:

- Recirculation (Dump-To-Sump – DTS) of RCS fluid from the hot leg to the Reactor Building (RB) sump using the decay heat drop line. This method reduces the core boron concentration by inducing RCS flow from the top of the core into the sump.

- Hot leg injection via the Auxiliary Pressurizer Spray (APS). This method injects a portion of LPI flow into the pressurizer through the auxiliary spray line. The pressurizer fills with APS liquid and then flow drains back into the hot leg and into the vessel. If the APS flow exceeds the core boiloff rate by a small fraction, a reverse flow is induced in the core and the core boron concentration is lowered.

Each of these methods has operational limitations. When DTS is utilized, RCS pressure must be below a minimum limit to prevent damage to the RB sump internal structures near the decay heat line intake. In addition, the backflow of saturated hot leg liquid flashes as it enters the sump and this steam could be entrained into the adjacent decay heat line intake that is providing flow to the operating LPI pump. When APS is utilized, its effectiveness is limited by the available flow and decay heat at which a reverse core flow can be achieved. APS will only be effective at higher RCS pressures when the elapsed post-trip time is on the order of 3 to 6 days. Additionally, the current methods are not single failure-proof. A failure of the Engineered Safeguards (ES) bus ES MCC-3AB can render both methods unavailable. For this specific vulnerability, a single failure exemption was requested and obtained (Reference 8).

The new HLI connection provides a single failure-proof flow path for boric acid precipitation control, thus assuring compliance with the long-term core cooling acceptance criterion by providing a flow path to inject a portion of the LPI into the DHDL. The HLI line is normally closed, but it will be opened by the operators during the transition to the sump recirculation, which will be prior to the occurrence of significant boric acid concentration. This approach to initiating boric acid precipitation control is based on direct action, instead of on elapsed time, or inferred or monitored symptoms. The flow path is single failure-proof such that HLI will be available for alignment with ongoing LPI injection, ensuring a direct flow path to the core region. The HLI line is hydraulically designed to provide sufficient flow to prevent boric acid precipitation for the entire spectrum of LOCA break sizes. Actuation of the HLI method will occur near the time of switching from the BWST to the reactor building sump, which will be well before precipitation is expected to occur. The line delivers a flow of at least 400 gpm at atmospheric pressure, and will manage the boron concentration for the LBLOCA scenario, which has the highest required HLI flow rate. For LBLOCAs, the HLI flow exceeds the core boiloff shortly after its initiation. The excess HLI flow that is not boiled off by the core decay heat dilutes the core boric acid concentration via reverse core flow prior to the core reaching concentrations that could precipitate. For SBLOCAs, the RCS pressure could be above the LPI shutoff head or in the range where the HLI flow does not match core decay heat. However, at these elevated RCS pressures, which correspond to elevated saturation temperatures, the solubility limit is above the maximum concentration that the core could achieve. The HLI flow will increase as the pressure decreases (either naturally or through operator-initiated cooldown) such that the flows match core decay heat and provide a boric acid dilution flow prior to reaching the solubility limit. Additionally, for those small breaks with RCS pressures remaining elevated above the LPI shutoff head, additional procedural guidance will be provided as described in the Long Term Core Cooling section, in order to meet the fifth criterion of 10 CFR 50.46.

For further details on the boron precipitation evaluation and mitigation strategy see EPU Technical Report Appendix D, "Core Boric Acid Dilution Control for CR-3 at EPU Conditions.".

As part of the short term actions, the Reactor Coolant Pumps are automatically tripped by the new ICCMS within one minute of losing subcooling margin and reactor trip. ECCS actuation is governed by the Engineered Safeguards Actuation System (ESAS) described in FSAR Section 7.1.3. The ESAS monitors variables to detect the loss of RCS boundary condition: upon detection of “out of limit” conditions, it initiates active components of the ECCS. For the EPU, no physical modifications are required for the ESAS.

While not part of the ECCS, the EFW System (FSAR Section 10.5) and SGs (FSAR Section 4.2.2.2) play a significant role in mitigating the smallest SBLOCAs. The primary-to-secondary heat transfer promoted by EFW and SG primary condensate augments core cooling for the smaller breaks that depressurize slowly. EFW is initiated and controlled by the Emergency Feedwater Initiation and Control (EFIC) System described in FSAR Section 7.2.4. The secondary side also includes a modification to the Atmospheric Dump Valves (ADVs) in support of the EPU. Their capacity and their safety-related classification are changed such that they can be credited in SBLOCA mitigation.

Along with its modified controls, the ADVs comprise the Fast Cooldown System (FCS), which will ensure a safety-related, blowdown of secondary side pressure in both steam generators. The FCS will be actuated by the ICCMS within 10 minutes of a reactor trip with a LSCM and sustained inadequate HPI margin. The actuation will automatically reduce secondary side pressure to, and control at, a value of ≤ 350 psig, As mentioned in the SBLOCA Results section, the secondary side depressurization reduces primary side pressure, thus improving pumped HPI injection and delivering some CFT liquid to the core. The depressurization can also cause the existing FOGG circuitry to occasionally interrupt EFW. The FOGG signal actuates and resets, with the net effect being that the occasionally interrupted EFW was found to have negligible effects on the transient progression, and did not adversely affect compliance with 10 CFR 50.46 acceptance criteria.

ICCMS determines whether a fast cooldown is appropriate based on adequate HPI flow. Assuming the limiting single failure scenario of losing one emergency core cooling train, ICCMS will always actuate the FCS in the full power analyses. ICCMS will also automatically raise the SG level to the Loss of Subcooling Margin (LOSCM) setpoint within 10 minutes after the reactor trip and loss of subcooling.

The upgrade to safety-related controls and components permit explicit credit for secondary side cooldown as either an automatic or manual action in the SBLOCA analyses (see SBLOCA Action 6 in Table 2.8.5.6.3-5). Section 2.5.5.3, Steam Dump System, and Appendix E, Major Plant Modifications, discuss the modified ADVs in more detail.

Finally, the existing High Pressure Injection (HPI) throttle valves will be fully opened to allow higher flowrates to the RCS. Tables 2.8.6.5.3-2A through 2C compare the flows for the various SBLOCA scenarios before and after the throttle valve resetting.

Description of Analysis Methods and Compliance with the SER Limitations and Restrictions

The LOCA analyses are described in FSAR Section 14.2.2.5. The primary objective of the LOCA analyses is to qualify the ECCS, EFW flow, and operator actions needed to demonstrate compliance with 10 CFR 50.46. A secondary result of the LOCA analyses is to determine the maximum linear heat rate (LHR) limits (in kW/ft) that establish the limits of normal core operation for the fuel reload process. These limits are evaluated on a cycle-specific basis to assure they remain applicable. The fuel assembly design used in the LOCA analyses is the Mark-B-HTP fuel design. Details on the fuel design are provided in Section 2.8.2, Nuclear Design.

The EPU LOCA analyses are performed in accordance with the NRC–approved, RELAP5-based Evaluation Model (EM) described in BAW-10192-A (Reference 1) as amended by any NRC-approved code topical revisions, 10 CFR 50.46 changes, and methodology changes from the NRC-approved topical reports (e.g., M5™ topical report, BAW-10227-A, Reference 2). The methods for analyzing LBLOCAs for the B&W-designed 177 fuel assembly lowered-loop plant are contained in Volume I of BAW-10192-A, while the small break SBLOCA methods are described in Volume II of BAW-10192-A. No new methods are being implemented for the EPU LOCA analyses although a higher elevation-skewed axial power shape was included to bound the middle and end of cycle shapes that could occur.

The NRC Safety Evaluation Report (SER) on BAW-10192-A contains eleven restrictions related to the use of the EM. The restrictions are provided on pages LA-160 and LA-161 of Reference 1. Compliance with these eleven restrictions was confirmed in the EPU LOCA analyses. There are three EM changes made under 10 CFR 50.46 that are not approved within the topical reports. One applies to both LBLOCA and SBLOCA analyses and pertains to the uncertainty-adjusted core flood tank (CFT) parameters. It is satisfied by evaluating a combination of minimum and maximum CFT initial liquid volumes and pressures and using the combination that results in the highest peak cladding temperature (PCT). Two others apply to reactor coolant pump two-phase degradation modeling for LBLOCA and SBLOCA. Both changes were resolved by using a pump degradation curve that provided the limiting PCT results. For the SBLOCA change in particular, a key part of the resolution was the need to immediately trip the RCPs for non-LOOP scenarios. For CR-3, a one minute time limit has been established for this action following LSCM (References 5 and 6). These resolutions have been incorporated in the LOCA analyses.

The EPU LOCA analyses consider plant modifications in support of the EPU such as LPI cross-tie, HLI, and FCS described above. Accordingly, these modifications prompted changes to analysis inputs. However, some key inputs (e.g., HPI flows for SBLOCA, LPI flows for LBLOCA) were unchanged from the current analyses of record. Listed in the Tables 2.8.5.6.3-1 through 2.8.5.6.3-4 are the key input parameters that differ from those used in the pre-EPU LOCA analyses. Operator and new automatic actions credited are also tabulated in Table 2.8.5.6.3-5.

Demonstration of Compliance with 10 CFR 50.46

Summary of the EPU Peak Clad Temperature, Local Oxidation, and Whole Core Hydrogen Generation

The EPU LBLOCA analyses specifically determine compliance with the first three 10 CFR 50.46 acceptance criteria. For LBLOCA analyses, the limiting results are summarized in Table 2.8.5.6.3-6, LBLOCA Analyses Results.

The EPU SBLOCA analyses specifically determine compliance with the first three 10 CFR 50.46 acceptance criteria. For SBLOCA analyses, the limiting results are summarized in Table 2.8.5.6.3-7, SBLOCA Analyses Results.

Maintaining Core Coolable Geometry

Additional analyses and evaluations are performed for the CR-3 EPU to show compliance with the last two criteria of 10 CFR 50.46, maintaining a coolable core geometry and establishing long-term cooling (LTC) for the core. The evaluations are performed to determine the effects on initial fuel assembly flow area considering fuel rod bowing (based on pin peaking limits), mechanical deformation from LOCA plus seismic (safe shutdown earthquake) dynamic loads, and the swelling and rupture-induced alterations of the fuel pins and assembly flow area from the thermal effects during a LOCA. These contributions are evaluated to ensure that gross flow blockage will not occur (i.e., reduction in fuel assembly flow area by more than 90 percent) and that the changes in the geometry will not impair or prevent the insertion of the control rods.

The effects of fuel rod bowing on assembly flow area and control rod guide tubes are considered in the fuel assembly and fuel rod designs, which minimize the potential for rod bow. CR-3 performs control rod drop tests based on Improved Technical Specification 3.1.4 to confirm that the control rods will fully insert into the fuel assemblies. The effects of rod bowing are also considered on pin peaking limits using the method described in Reference 3. The minor adjustments of fuel pin pitch due to rod bowing do not alter the fuel assembly flow area substantially, and the average subchannel flow area is preserved until the LOCA transient is initiated.

When the LOCA is initiated, the mechanical loads on the reactor vessel from the break opening results in short-term or dynamic loads that could cause permanent distortion of the core support structures, reactor vessel internals, and the fuel assemblies, thereby altering the core flow area or flow resistance. The maximum assembly loading occurs before the fuel pin experiences any significant heat up. Therefore, the mechanical effects are evaluated separately from the LOCA 10 CFR 50.46 analyses. Stress analyses of these dynamic blowdown effects, in combination with the seismic loads from an earthquake, are used to evaluate the mechanical loads on these components and confirm no fuel assembly deformation occurs. The leak-before-break (LBB) methodology in BAW-2292 (Reference 4) was used for determining these mechanical loads.

Long-Term Core Cooling

The fifth acceptance criterion of 10 CFR 50.46 states that the calculated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period of time required by long-lived radioactivity remaining in the core.

Successful initial operation of the ECCS is shown by demonstrating that the fuel pins remain in compliance with the first three criteria of 50.46 and the core is refilled and quenched, with the cladding temperature returned to near the saturation temperature. Thereafter, LTC is achieved by preserving continuous flow from the pumped injection systems until normal decay heat removal is established. These systems are redundant and a variety of configurations are able to provide a continuous flow of cooling water to the core fuel assemblies so long as the coolant channels in the core remain open. Moreover, operator actions directed by Emergency Operating Procedures (EOPs) assure LTC by actions such as swapping ECCS suction from the BWST to the reactor building sump, thus assuring the continued availability of pumped injection. The EOPs also direct actions to perform further cooldowns as needed until normal decay heat removal is achieved. These LTC actions are not significantly altered for the EPU.

In addition to the above, two other areas are also evaluated to show compliance with the fifth criterion: Generic Safety Issue (GSI) -191 and post-LOCA boron precipitation. GSI -191 concerns the ability of the ECCS to cool the core when debris from the containment could be potentially entrained into the ECCS flow. To address this concern, evaluations include the characterizing the types and quantity of the debris generated by the LOCA, its transport and potential obstruction of the sump screens, and the downstream effects of debris that passes through the sump screens to the ECCS pumps, RCS, and finally the core. CR-3 has modified its sump to incorporate debris-intercepting structures. The sump is now more effectively partitioned in order to filter the recirculation flow such that debris are trapped or otherwise hindered from entering the ECCS flowpath. As the result of fibrous insulation reduction efforts, CR-3 has a relatively small quantity of fibrous material that could affect reactor building sump and ECCS performance. As noted in correspondence dated October 2, 2009 (Reference 9), the NRC has acknowledged CR-3's compliance with GSI-191, with one open item related to in-vessel downstream effects, Progress Energy is currently participating in the PWR Owners Group effort to resolve the GSI-191 concerns associated with downstream effects (i.e., potential fuel assembly blockage) using testing and additional evaluations. The ultimate resolution for GSI-191 is applicable to both the current and the EPU power levels. The addition of the HLI line for post-LOCA boron precipitation control also provides some benefit with regard to the GSI-191 mitigation, and will be discussed below.

The other area concerns the potential for boric acid precipitation in the core region that could prevent coolant flow from reaching certain portions of the core. As stated in the Mitigating Systems section, Progress Energy is implementing modifications to the ECCS piping arrangement that includes a new cross-tie between the LPI trains and a new single failure proof connection from the new cross-tie to the DHDL to facilitate hot leg injection, or HLI. This flow path can be initiated and sufficient flow provided to dilute the boron for the entire spectrum of LOCA break sizes that could concentrate the boron in the core prior to when the solubility limit is reached. In general, the LBLOCA is the scenario of concern because the time frame wherein boric acid concentration approaches the solubility limit tends to be shorter, approximately one to two hours. The smaller break sizes of the SBLOCA, wherein vessel inventories are initially higher and RCS pressures are elevated longer, tend to provide more mixing volume, thus mitigating the concentrating mechanisms. Additionally, if RCS pressure remains elevated above the LPI shutoff head, operators will be provided instructions to perform a controlled cooldown/depressurization that allows HLI flow to improve dilution in the core region for the break sizes and locations where boron precipitation can occur, and instructions to ensure that other sources of boric acid are secured**.** In either case, the HLI line will be opened by operators during the transition to the sump recirculation, which will be prior to the occurrence of significant boric acid concentration.

The EPU post-LOCA core boric acid concentration analyses use the 10 CFR 50 Appendix K fission product decay heat with a multiplier of 1.2, with consideration for heavy actinides. The analyses include the effects of a spectrum of postulated cold leg break sizes for both large and small break LOCAs. For certain break sizes, the operation of Reactor Vessel Internal Vent Valves, which are unique to the B&W design, provides a flowpath from the upper plenum to the vessel downcomer that mitigates the concentration mechanism in the core region. The core mixing volume, which is limited to the liquid mass in the core region, upper plenum, and outlet annulus region below the mixture level, considers adjustments to account for the EPU power level and other volume reductions that address recent operating experience on post-LOCA boric acid precipitation from other plants' licensing submittals. The core mixing volume liquid mass is determined based on pure water properties. The steam in the core mixing volume is excluded from the available mixing volume. The analyses do not credit any lower plenum liquid volume.

Other penalties include a 4 weight percent H_3BO_3 uncertainty subtracted from the solubility limit. No containment over-pressure is credited for LBLOCA, but the RCS pressure and saturation temperature is considered in the solubility limit for SBLOCAs. While the analysis do not explicitly include the related concerns stemming from GSI-191, opening the HLI valve(s) during the time of sump switchover and the magnitude of the HLI flows do provide margin to address certain GSI-191 effects on the core mixing volume and uniformity of the core boric acid concentrations.

2.8.5.6.3.3 Technical Evaluation – LOCA Forces

Introduction

LOCA forces are generated during high-energy line breaks (HELBs) as a result of asymmetric cavity pressure, jet impingement, thrust internal forces from changes in area and direction, and internal pressure acting on components. The analysis of the LOCA hydraulic forces generates the hydraulic forcing functions that act on RCS components as a result of postulated LOCA.

Description of Analyses and Evaluations

LOCA forces on RCS components are primarily a function of RCS temperature and pressure. To conservatively calculate LOCA hydraulic forces for CR-3 under the EPU conditions, T_{HOT} was evaluated at 604°F and T_{COLD} was evaluated at 550°F. This set of minimum temperatures was used in the modeling of the RCS since it results in higher RCS densities and, thus, overall higher loads during a postulated LOCA. These temperatures are a lower bound of those provided in Table 1.1-1. A nominal RCS pressure of 2250 psia was used in the calculations since, as stated in Section 1, RCS pressure is unchanging for the EPU.

Due to application of LBB (see Section 2.1.6, Leak-Before-Break, for additional discussion), the bounding set of worst case HELB locations considered in the evaluation of LOCA forces are: the decay heat line at the hot leg nozzle, core flood line at the reactor vessel nozzle and surge line at the hot leg nozzle (see Section 2.2.1, Pipe Rupture Locations and Associated Dynamic Effects, for discussion of pipe rupture locations and associated dynamic effects).

Based on the operating conditions and break locations mentioned above, forcing functions representing the depressurization waves (blowdown) traveling through the loop during the LOCA were calculated at various locations throughout the RCS. Forces on the RCS components due to jet impingement, thrust and asymmetric cavity pressure were also calculated and summed with the blowdown forcing functions to create an overall set of LOCA forcing functions acting on the RCS piping and components.

These LOCA forcing functions combined with deadweight, seismic and thermal loads, were used in the structural evaluations to determine the resultant mechanical loads on the RCS loop piping, components and supports. The RCS loop forces are provided for use in the structural analyses described in Sections 2.2.2.1 (NSSS Piping, Components and Supports), 2.2.2.3 (Reactor Vessel and Supports) and 2.2.2.6 (Reactor Coolant Pumps and Supports).

For the reactor vessel, vertical LOCA forces on the core were calculated for the EPU conditions for the same break locations as mentioned above. These forces were used in the structural evaluation of the fuel assemblies in Section 2.8.1, Fuel System Design.

Results

In summary of the technical evaluation above for LOCA forces, all relevant LOCA hydraulic forces analyses were performed at the EPU power level of 3014 MWt, using models specific to the CR-3 NSSS design. The results of these analyses were then used as input to the calculations that demonstrated continued component qualification discussed in Section 2.8.1 for fuel and Sections 2.2.2.1, 2.2.2.3, and 2.2.2.6 for the RCS loop piping, components and supports.

In summary of the technical evaluation above for ECCS and LOCA, the LBLOCA and SBLOCA RELAP5-based system analyses, performed in accordance with the approved Evaluation Model, demonstrated compliance with the first three acceptance criteria (peak clad temperature, maximum local oxidation, maximum hydrogen generation) of 10 CFR 50.46. The core geometry was also evaluated to be amenable to cooling, which satisfies the fourth criterion of 10 CFR 50.46. For the fifth acceptance criterion of establishing and maintaining long-term core cooling, appropriate EOP actions and system alignments assure the availability of pumped injection and of achieving normal decay heat removal conditions. Boric acid precipitation control is achieved using the HLI line, which will provide a single failure-proof dilution flow path. GSI-191 issues are being resolved by Progress Energy in response to Generic Letter 2004-02. Based on the continued compliance with 10 CFR 50.46 criteria, CR-3 FSAR design criteria described previously will continue to be met at the EPU conditions.

2.5.5.6.3.4 Conclusion

CR-3 has performed analyses of the LOCA events and the ECCS. CR-3 concludes that the analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. CR-3 further concludes that the licensee has demonstrated that the reactor protection system and the ECCS will continue to ensure that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, CR-3 concludes that the plant will continue to meet the requirements of FSAR Sections 1.4.23, 1.4.28, 1.4.29, 1.4.30, 1.4.37, 1.4.41, 1.4.42, and 1.4.44, as well as 10CFR 50.46 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Emergency Core Cooling System and Loss-of-Coolant Accidents.

2.8.5.6.3.5 References

1. BAW-10192-A, Revision 0, “BWNT LOCA – Loss of Coolant Accident Evaluation Model for Once Through Steam Generator Plants,” June 1998.

2. BAW-10227NP-A, Rev. 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” June 2003.

3. BAW-10179NP-A, Rev. 7, “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses,” January 2008.

4. BAW-2292, Rev. 0, “Mk-B FA Spacer Grid Deformation,” February 1997.

5. H. Berkow (USNRC) Letter to J. Mallay (Framatome ANP), Evaluation of Framatome ANP Preliminary Safety Concern (PSC 2-00) Relating to Core Flood Line Break and Operator Action Time (TAC No. MA9973). April 10, 2003.

6. R. A. Gramm (USNRC) Letter to J. Holm (Framatome ANP), Request for Amendment of Safety Evaluation for “Report of Preliminary Safety Concern (PSC) 2-00 Related to Core Flood Line Break with 2-Minute Operator Action Time.” January 10, 2005.

7. S. J. Cahill (Progress Energy) Letter to U.S. Nuclear Regulatory Commission, Crystal River Unit 3 – 10CFR 50.46 Notification of Change in Peak Cladding Temperature for Small Break Loss of Coolant Accident Analyses, September 8, 2010.

8. L. A. Wiens (USNRC) to J. P. Cowan (Florida Power Corporation), Issuance of Exemption from the Requirements of 10 CFR Part 50, Appendix K, Section I.D.1 – Crystal River Unit 3 (TAC No. M99892). October 29, 1998.

9. Farideh E. Saba (NRC) Letter to Jon A. Franke (Progress Energy), Crystal River Unit 3 Nuclear Generating Plant -Partial Close Out And Request For Additional Information Related To Generic Letter 2004-02 (TAC NO. MC4678), October 2, 2009.

Table 2.8.5.6.3-1 Key Input Parameter Differences

Parameter	Pre-EPU Value	EPU Value	Remarks
Rated Core Power, MWt	2619 including heat balance uncertainty	3026 including heat balance uncertainty	Power increase
RCS Average Temperature, °F	579	582	Supports power increase
SG Tube Plugging, %	20% average for LBLOCAs (OTSGs). 5% average for SBLOCAs (OTSGs), and 10% of EFW-wetted tube region is assumed plugged for SBLOCA	5% average for all LOCAs, and 10% of EFW-wetted tube region is assumed plugged (for SBLOCAs)	Reduction due to replacement of steam generators prior to the EPU
ADV Capacity	ADV Not Modeled in both LOCAs	589,000 lbm/hr at 540°F for SBLOCAs	Fast Cooldown System modification to improve ECCS performance for SBLOCAs
ADV Control	ADV Not Modeled in both LOCAs	On actuation, both ADVs open to blowdown and establish automatic secondary side pressure modulation at ≤350 psig for SBLOCAs	Fast Cooldown System modification to improve ECCS performance for SBLOCAs
EFW Flow Rate, gpm	200/SG for SBLOCAs	300/SG maximum; or EFIC-determined flowrate, whichever is less for SBLOCAs	Result of crediting Fast Cooldown included consideration for EFIC
SBLOCA Power Shape	1.7 axial at 9.536 ft	1.7 axial at 11 ft	Skewed the axial power shape to cover the MOC to EOC axial power profiles at the EPU
HPI Flows	Table 2.8.5.6.3-2A, for CLPD break; Table 2.8.5.6.3-2B, for HPI line double-ended break; Table 2.8.5.6.3-2C, for CF line break.	Table 2.8.5.6.3-2A, Throttled for CLPD break; Table 2.8.5.6.3-2B, Throttled or Unthrottled for HPI line double-ended break; Table 2.8.5.6.3-2C, Throttled for CF line break.	As described above, HPI in some EPU cases reflected continued use of throttled flows (full power), while others reflected unthrottled flows (part power).
LPI Flows	Pre-EPU did not have LPI crosstie	EPU with LPI crosstie	Flows listed in Table 2.8.5.6.3-3 for CLPD breaks and in Table 2.8.5.6.3-4 for the CFT line breaks

Table 2.8.5.6.3-2A SBLOCA Available HPI Flow Rates – CLPD Line Break

NOTE: The Cold Leg (CL) Flows reflect the distribution of HPI into the RCS and are credited for cooling

Throttled (Pre-EPU)			Unthrottled (EPU)		
RCS Pressure (psia)	**Broken CL Flow (gpm)**	**Intact CL Flow (gpm)**	**RCS Pressure (psia)**	**Broken CL Flow (gpm)**	**Intact CL Flow (gpm)**
15	135.7	332.4	15	127.3	374.9
615	121.9	298.5	915	108	318
915	114.1	279.5	1815	82.2	241.9
1215	105.6	258.7	2115	70.5	207.5
1515	96.1	235.6	2615	39.6	116.6
1815	85.4	209.2			
2115	72.8	178.1			
2415	56.7	139.0			

Table 2.8.5.6.3-2B SBLOCA Available HPI Flow Rates - HPI Line Break

NOTE: No flow from the broken HPI line is credited for core cooling. Also, above 615 psia, the throttled HPI valves assured a more balanced distribution between the broken and intact HPI lines. Unthrottling the valves improved the CLPD response, but allowed more flow lost through the broken HPI line. However, the FCS ensures that this break location remained bounded by the CLPD breaks.

Throttled (Pre-EPU)			Unthrottled (EPU)		
RCS Pressure (psia)	**Broken Flow (gpm)**	**Intact HPI Flow (gpm)**	**RCS Pressure (psia)**	**Broken Flow (gpm)**	**Intact HPI Flow (gpm)**
15	0	332.4	15	0	374.9
615	0	281.2	315	0	331.6
915	0	253.1	915	0	243.2
1215	0	222.7	1515	0	144.2
1515	0	189.2	1815	0	92.6
1815	0	141.0	2115	0	37.2
2115	0	90.2	2215	0	14.2
2415	0	25.7			

Table 2.8.5.6-2C SBLOCA Available HPI Flow Rates - CF Line Break

Throttled (Pre-EPU)		**Unthrottled (EPU)**	
RCS Pressure (psia)	**Total Flow to RCS (gpm)**	**RCS Pressure (psia)**	**Total Flow to RCS (gpm)**
15	468.1	15	502.2
615	420.4	915	426
915	393.6	1815	324.1
1215	364.3	2115	278
1515	331.7	2615	156.2
1815	294.6		
2115	250.9		
2415	195.7		

Table 2.8.5.6.3-3 SBLOCA CLPD Line Break LPI Flows

Pre-EPU		**EPU (Modified LPI System)**	
RCS Pressure (psia)	LPI Flow Rate (gpm)	RCS Pressure (psia)	LPI Flow Rate (gpm)
14.7	2685	14.7	2886
50	2685	84	2886
75	2685	100	2687
100	2615	125	2286
124	2110	150	1715
140	1685	173	625
150	1360	175	200
165	610	> 175	0
170	0	----	----

Table 2.8.5.6.3-4 SBLOCA CFT Line Break LPI Flows

Pre-EPU		**EPU (Modified LPI System)**[Note]	
Pressure (psia)	LPI Flow Rate Intact Line (gpm)	RCS to Containment Pressure Difference (psia)	Intact Line LPI Flow Rate (gpm)
14.7	0		
50	0		
75	0	0.0	1435
100	0	30.0	930
124	0	60.0	417
140	0	69.0	236
150	0	69.1	0
165	0		
170	0		

Note: These flow rates are valid for containment pressures below 43 psia.

Table 2.8.5.6.3-5 Operator and New Automatic Actions

Action	Description	EPU Comparison
LBLOCA		
1	A continuous ECCS source is maintained, such as through transferring ECCS suction from the BWST to the sump for long-term cooling.	Same as pre-EPU
2	Appropriate boric acid concentration control is maintained to prevent precipitation or recriticality and to ensure long-term cooling.	The Hot Leg Injection Line provides a dilution flow path when opened. The redundant, single failure-proof flow path is sized to provide at least a minimum of 400 gpm at atmospheric pressure (see Mitigation System section).
3	For LBLOCA analyses that do not postulate LOOP (not explicitly analyzed), automatic or manual operator action to trip the RCPs within one minute of a Loss of Subcooling Margin indication is assumed.	The addition of automatic RCP trip on a reactor trip and loss of subcooling margin allows either automatic or manual action. The action time used is applicable to either mode of actuation.
4	Terminate any fluid additions to the Makeup and Purification System from the Boric Acid Storage Tanks, the RC Bleed Tanks, and the Demineralized Water System.	New action for EPU to minimize potential for excessive boron addition or dilution.
SBLOCA		
1	Initiate raising the secondary SG level to a minimum actual level of 73 percent of operating range (LSCM setpoint) by 10 minutes after the LSCM. The level setpoint at the plant must consider appropriate instrument error and uncertainty to ensure that 73 percent of operating range is protected. This level is approximately two feet above the RCP spillover elevation and ensures that a condensing surface is available before core uncovering occurs.	The addition of ICCMS allows either automatic or manual action for performing this action. The action time used is applicable to either mode of actuation.
2	For SBLOCAs that do not postulate LOOP (not explicitly analyzed), action to trip the RCPs within one minute after Loss of Subcooling Margin indication is assumed.	The addition of automatic RCP trip on a reactor trip and loss of subcooling margin allows either automatic or manual action. The action time used is applicable to either mode of actuation.

Table 2.8.5.6.3-5 Operator and New Automatic Actions

Action	Description	EPU Comparison
3	For SBLOCA analyses that do not predict automatic ES actuation during the initial depressurization phase, immediate operator action (with a conservative delay) to manually initiate ES after the LSCM will be assumed.	Same as pre-EPU
4	A continuous ECCS source is maintained, such as through transferring ECCS suction from the BWST to the sump for long-term cooling.	Same as pre-EPU
5	Appropriate boric acid concentration control is maintained to prevent precipitation or recriticality and to ensure long-term cooling.	The Hot Leg Injection Line provides a dilution flow path when opened. The redundant, single failure-proof flow path is sized to provide adequate flow for the entire SBLOCA LTC pressure range (see Mitigation System section).
6	Automatic or manual operator action to open an ADV is credited at 10 minutes after LSCM. This fast cooldown with the ADVs will continue to depressurize secondary side pressure down to 350 psig, with the ADVs modulating at ≤350 psig thereafter.	New action for EPU. Once actuated, the safety-related ADVs provide blowdown and automatic modulation of secondary pressure. The action time used is applicable to either automatic (ICCMS) or manual mode of actuation.
7	Terminate any fluid additions to the Makeup and Purification System from the Boric Acid Storage Tanks, the RC Bleed Tanks, and the Demineralized Water System.	New action for EPU to minimize potential for excessive boron addition or dilution.
8	For certain smaller LOCAs, where RCS pressure remains above the shutoff head of the LPI pumps following transition to the RB sump, operators will limit RCS depressurization and cooldown that could cause boron precipitation until adequate HPI flow has flushed the core region.	New action for EPU to avoid intentionally operating the plant in a region that could cause either boron precipitation or dilution problems.

Table 2.8.5.6.3-6 LBLOCA Analyses Results

10 CFR 50.46 Criteria	Pre-EPU Results	EPU Results
PCT (<2200°F)	1994°F	1947°F
Maximum Local Oxidation (<17%)	< 5%	< 3%
Whole Core Hydrogen Generation (<1%)	< 0.2%	< 0.2%

Table 2.8.5.6.3-7 SBLOCA Analyses Results

10 CFR 50.46 Criteria	Pre-EPU Results	EPU Results
PCT (<2200°F)	1535°F	1426°F
Maximum Local Oxidation (<17%)	<0.3 %	<1 %
Whole Core Hydrogen Generation (<1%)	< 0.01%	< 0.03%

Figure 2.8.5.6.3-1. Comparison of the LBLOCA Mark-BHTP UO_2 LOCA LHR Limits for BOL.

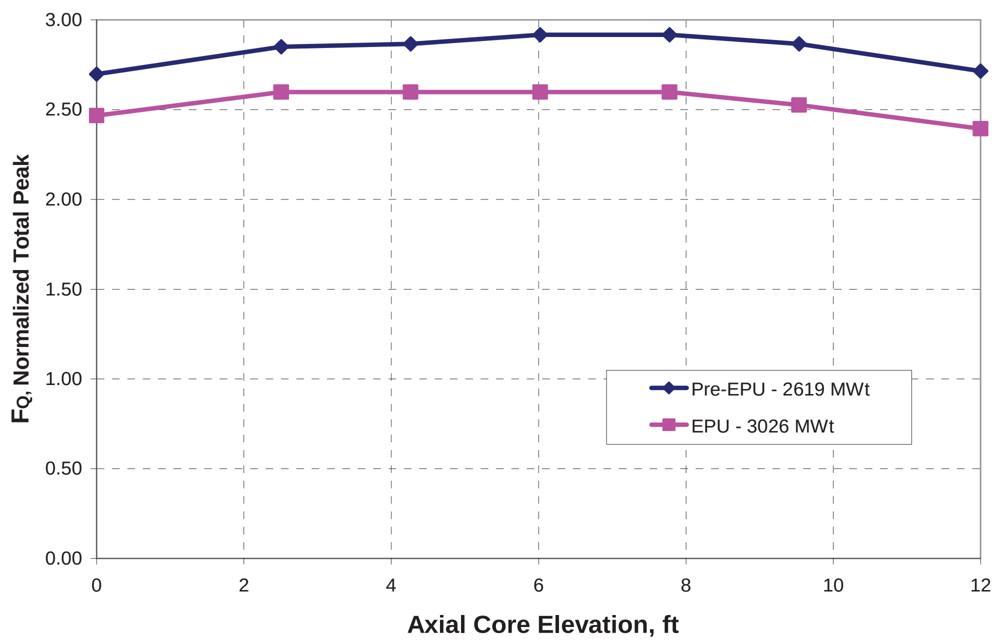
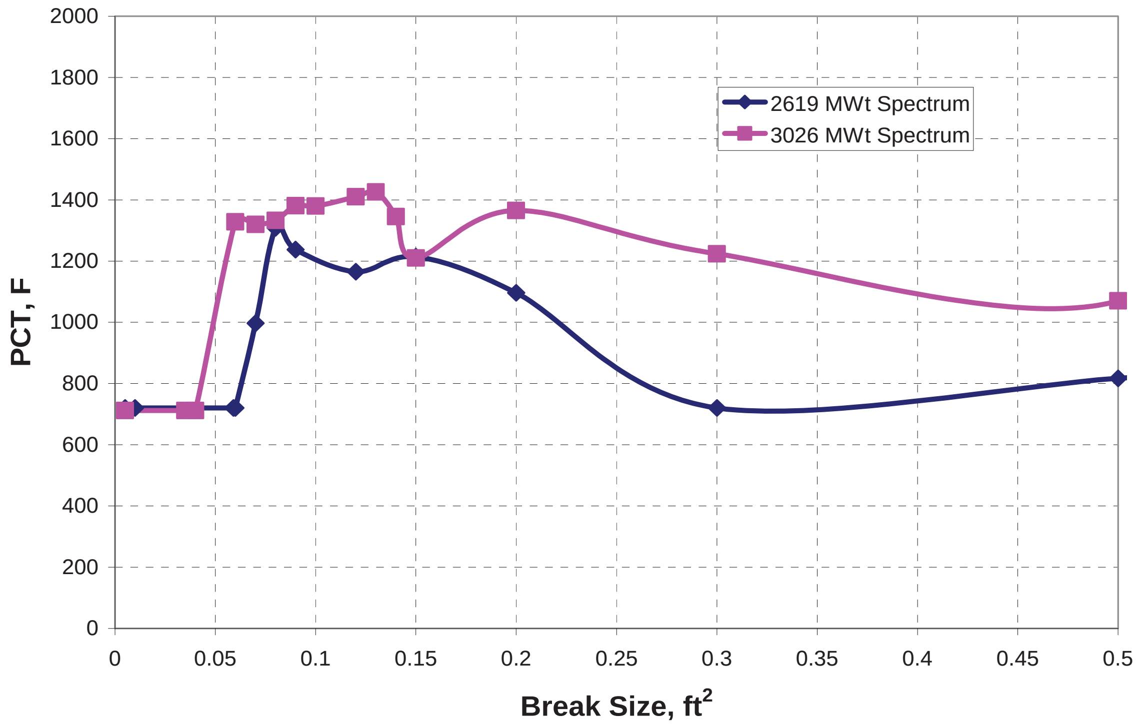


Figure 2.8.5.6.3-2. Comparison of the Full Power CLPD SBLOCA PCTs Versus Break Size.

NOTE: The reported maximum PCT for 2619 MWt is 1535 F, reflecting an applied 225 F penalty to account for an 11-ft axial peak. The 2619 MWt spectrum below reflects a 9.5-ft axial peak which does not include the 225 F penalty. However, the 3026 MWt spectrum reflects the 11-ft axial peak.



2.8.5.7 Anticipated Transients Without Scram

2.8.5.7.1 Regulatory Evaluation

Anticipated Transients Without Scram (ATWS) are defined as an anticipated operational occurrence followed by the failure of the reactor portion of the protection system as specified in GDC-20. The regulation at 10 CFR 50.62 requires that:

- Each pressurized water reactor (PWR) must have equipment that is diverse from the reactor trip system to automatically initiate the Auxiliary (or Emergency) Feedwater System and initiate a turbine trip under conditions indicative of an ATWS. This equipment must perform its function in a reliable manner and be independent of the existing reactor trip system, and,

- Each PWR manufactured by Combustion Engineering (CE) or Babcock and Wilcox (B&W) must have a Diverse Scram System. This scram system must be designed to perform its function in a reliable manner and be independent of the existing reactor trip system.

The CR-3 review was conducted to ensure that (1) the above requirements are met, and (2) the setpoint for the ATWS Mitigating System Actuation Circuitry (AMSAC) and Diverse Scram System (DSS) remain valid for the proposed EPU. In addition, the CR-3 review verified that the consequences of an ATWS are acceptable. The acceptance criterion is that the peak primary system pressure should not exceed the ASME Boiler and Pressure Vessel (B&PV) Code, Service Level C Limit, which corresponds to a peak pressure limit of 3200 psig. The peak ATWS pressure is primarily a function of the moderator temperature coefficient (MTC) and the primary system relief capacity. CR-3 reviewed (1) the limiting event determination, (2) the sequence of events, (3) the analytical model and its applicability, (4) the values of the parameters used in the analytical model, and (5) the results of the analyses. CR-3 reviewed the applicability of generic vendor analyses to CR-3 and the operating conditions for the proposed EPU.

CR-3 Current Licensing Basis

The analysis of the ATWS event is not described in the CR-3 FSAR; however the design of the DSS and AMSAC systems is discussed in FSAR Section 7.5.

2.8.5.7.2 Technical Evaluation

Introduction

FSAR Section 7.5 describes the ATWS equipment which was installed at CR-3 including (1) a DSS independent of the Reactor Protection System (RPS), and (2) an AMSAC independent and diverse from the RPS which initiates Emergency Feedwater (EFW) and trips the turbine. The ATWS System is not required to be safety-related; however, all interfaces with a safety system are in compliance with the requirements of that particular safety system.

ATWS events were analyzed for the B&W designed plants which modeled a “typical” plant generally assuming a composite of plant characteristics (Reference 1). It was determined that upon evaluation of all applicable transients, the most limiting event for peak system pressure is the Loss of Feedwater (LOFW) event. As a result of the feedwater reduction, the heat removal capability of the once through steam generators (OTSGs) is diminished. Due to the EFW actuation delay, the steam generators boil dry before EFW reaches the OTSGs. Consequently the Reactor Coolant System (RCS) temperature and pressure increase rapidly. The pressurizer spray is actuated in an attempt to reduce the pressure increase. The actions of the pressurizer spray are insufficient to prevent the pressurizer pressure from increasing. A failure of the RPS to trip the reactor is postulated for the LOFW ATWS event. Thus, following the loss of heat sink and actuation of pressurizer spray, the RCS pressure and temperature continue to increase until the DSS high RCS pressure setpoint is reached and the control rod groups tied to DSS are tripped.

The increase in RCS pressure is mitigated by the pressurizer power operated relief valve (PORV) and the pressurizer code safety valves (PCSVs). The PORV and PCSVs continue to cycle until EFW fluid reaches the steam generators and the heat removal capability of the EFW fluid is adequate to remove core decay heat, RCS latent heat, and reactor coolant pump (RCP) heat. At this time the reactor coolant (RC) temperature will begin to decrease. Once an adequate heat sink is restored and RCS pressure is decreasing, the event is effectively mitigated.

Description of Analyses and Evaluations

For operation at the EPU conditions, the LOFW ATWS event analysis was performed using the NRC-approved RELAP5/MOD2-B&W thermal-hydraulic computer code (Reference 2), in conjunction with plant-specific inputs. The RELAP5/MOD2-B&W computer code has been approved by the NRC for use in non-Loss of Coolant Accident (LOCA) safety analyses. The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The RCS model provides for heat transfer from the core, transport of the coolant to the OTSGs, and heat transfer to the OTSGs. The RELAP5/MOD2-B&W secondary model includes a detailed depiction of the Main Steam System, including steam relief to the atmosphere through the main steam safety valves (MSSVs) and simulation of the turbine stop valves (TSVs). The secondary model also includes the delivery of feedwater, both main and emergency, to the OTSGs (as appropriate).

The key input parameters and initial conditions used in the LOFW ATWS analysis are as follows:

- The initial power level is assumed to be the nominal EPU power level (3014 MWt).
- The initial hot leg pressure is assumed to be the nominal value of 2170 psia.
- The average RCS temperature is assume to be 582°F, consistent with the planned increase in T_{AVG} planned in conjunction with the EPU.
- The initial pressurizer level is assumed to be the nominal, hot full power value of 220 inches indicated.
- Pressurizer spray is modeled with nominal setpoints and spray capacity.
- The PORV is modeled with nominal open/close setpoints and relief capacity.
- The PCSVs are modeled with a conservative opening setpoint and nominal relief capacity.
- The initiating event is a loss of main feedwater, which is simulated with a linear reduction in main feedwater flow over 3.2 seconds.

- The design of AMSAC is such that emergency feedwater is initiated once the main feedwater flow rate in both loops decreases to less than 17% of nominal feedwater flow. However, the LOFW ATWS analysis conservatively assumes that EFW is initiated on the low steam generator level Emergency Feedwater Initiation and Control (EFIC) trip, resulting in a longer time to EFW actuation. In addition, a conservative delay time is assumed for initiation of EFW. The assumed EFW flow rate is less than the minimum available for the EPU, and the delay time is conservatively longer than the maximum delay expected for the EPU. Due to the late delivery of EFW, this parameter has no impact on peak RCS conditions.
- Steam generator tube plugging levels of up to 5% are considered, which is slightly conservative with respect to the RCS pressure response.
- A hot full power MTC was chosen that is expected to bound 95% of the plant life. MTC as a function of burnup was reviewed for historical operating cycles as well as the expected EPU cycles to arrive at a value that will conservatively bound 95% of the plant life. The value was further adjusted to account for maneuvering uncertainties. The MTC was assumed to be -0.0062 %Δk/k/°F.
- RPS is assumed to fail, and the DSS high RC pressure setpoint is modeled to be 2464.7 psia. A conservative delay time is applied prior to control rod insertion.
- A conservative determination was made for the scram worth available. The DSS is associated only with a limited number of control rod groups. The total rod worth available was assumed to be 1.5 %Δk/k.
- Offsite power is assumed available, since the heat addition from the reactor coolant pumps increases the severity of the overheating event.
- No single failures are assumed consistent with the ATWS design basis.
- No operator actions are credited due to the short duration of the event.

The acceptance criteria for the LOFW ATWS event was that peak RCS pressure remains less than the ASME B&PV Code Service Level C limit stress criteria, which corresponds to a pressure of 3200 psig (3214.7 psia).

Results

The results of the LOFW ATWS evaluation for the EPU demonstrated that the resulting peak RCS pressure was lower than the ASME B&PV Code, Service Level C limit stress criteria, which correspond to a maximum pressure of 3200 psig (3214.7 psia). The peak pressure observed for the analysis was 2687.49 psia (Figure 2.8.5.7-1). Therefore, the analytical basis for compliance with 10 CFR 50.62 continues to be met for operation of CR-3 at the EPU power level. The setpoints for the AMSAC and DSS remain valid for the proposed EPU.

2.8.5.7.3 Conclusion

CR-3 has reviewed the analysis of the ATWS and concludes that it has adequately accounted for the effects of the proposed EPU on ATWS. CR-3 also concluded that the analyses were performed using an acceptable analytical approach for the event. CR-3 further concludes that the analyses have demonstrated that the DSS and AMSAC systems will continue to meet the requirements of 10 CFR 50.62 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to Anticipated Transients Without Scram events.

2.8.5.7.4 References

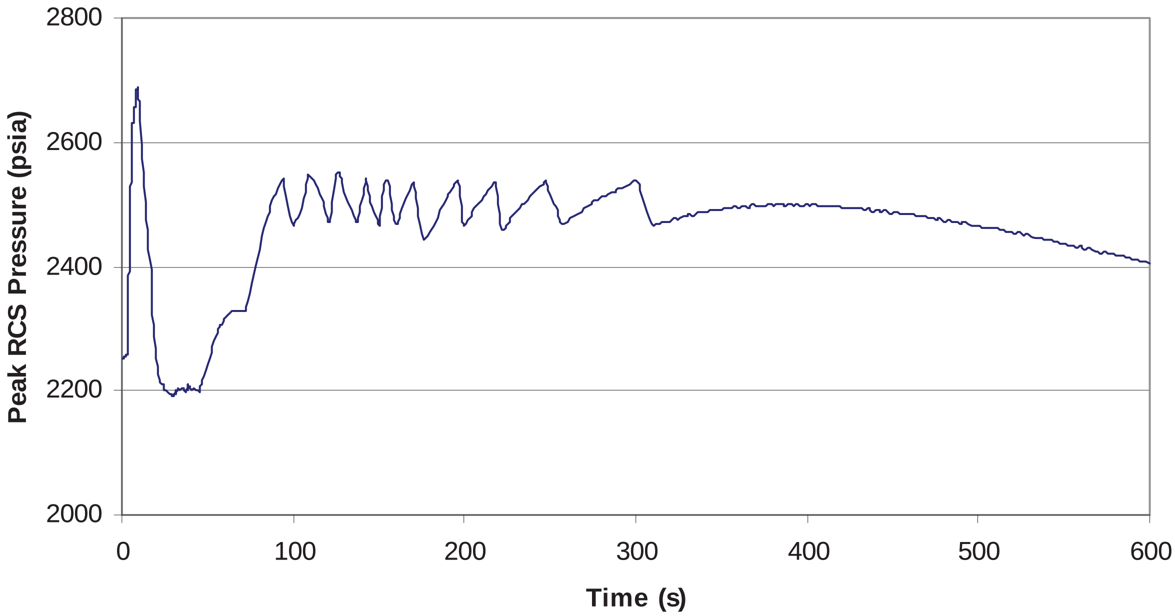
1. BAW-1610, "Analysis of B&W NSS Response to ATWS Events," January 1980.

2. AREVA NP Topical Report BAW-10164P-A, Revision 6, June 2007 (Proprietary), "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

Table 2.8.5.7-1: Sequence of Events for the HFP LOFW ATWS Event

Event	Time (s)
Event Initiated	0.0
AMSAC Signal Initiates Closure of Turbine Stop Valves	2.7
MFW Flow Decreases to Zero	3.2
PZR Spray Flow Initiated	3.5
Initial MSSV Lift	4.3
Initial PZR PORV Lift	5.5
Reactor Tripped by DSS	7.4
Peak RCS Pressure	9.0
Indicated PZR Level Off-Scale High	132.0
EFW Flow Reaches SGs	154.21
Final PORV Closure	310.0
Transient analysis ends	600.0

Figure 2.8.5.7-1: Loss of Feedwater ATWS



2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

2.8.6.1.1 Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The CR-3 review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the New Fuel Storage facilities.

The NRC's acceptance criteria for New Fuel Storage are based on:

- GDC-62, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.66, Prevention of Fuel Storage Criticality, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. [GDC-62]

2.8.6.1.2 Technical Evaluation

New fuel is stored in the New Fuel Storage Pit located in the Auxiliary Building or is placed directly into the Spent Fuel Pool. The New Fuel Storage Pit is intended for the receipt and storage of new fuel under dry conditions. The impact of the EPU on storage of new fuel in the Spent Fuel Pool is presented in Section 2.8.6.2, Spent Fuel Storage. As a result, this technical evaluation is focused only on storage of new fuel in the New Fuel Storage Pit.

Storage of new fuel in the New Fuel Storage Pit is not impacted by the EPU, since no new fuel design is required or introduced. Mark-B-HTP fuel will continue to be used, and is therefore bounded by the existing criticality evaluation of the New Fuel Storage Pit. Furthermore, EPU activities are not adding any new components, modifying existing components, or introducing any new functions for existing components relative to the New Fuel Storage Pit.

2.8.6.1.3 Conclusion

CR-3 has reviewed the existing criticality evaluation of the New Fuel Storage Pit and concludes that the analysis remain bounding. CR-3 further concludes that the New Fuel Storage facilities will continue to meet the requirements of FSAR Section 1.4.66 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the storage of new fuel in the New Fuel Storage Pit.

2.8.6.1.4 References

None

2.8.6.2 Spent Fuel Storage

2.8.6.2.1 Regulatory Evaluation

Nuclear power plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The CR-3 review covered the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy).

The NRC's acceptance criteria for Spent Fuel Storage are based on:

- GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; and
- GDC-62, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.66, Prevention of Fuel Storage Criticality, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. [GDC-62]

Additionally, FSAR Section 9.6.1.2.2, Spent Fuel Storage, provides criteria for spent fuel storage, insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. [GDC-4].

2.8.6.2.2 Technical Evaluation

<u>Introduction</u>

Both new fuel and spent fuel are stored underwater in a Spent Fuel Pool (designated as Pool A and Pool B; both of which feature high density storage racks). The design of Spent Fuel Pool A utilizes boron carbide (B_4C) poison and a centerline spacing of 10.5 inches between assemblies to prevent criticality. The design of Spent Fuel Pool B utilizes a neutron absorbing material (Boral®) and a center-to-center spacing of 9.11 inches. Boral® is a metallic composite of a hot-rolled (sintered) aluminum matrix containing B_4C sandwiched between and bonded to type 1100 alloy aluminum. After EPU implementation, Mark-B-HTP fuel will continue to be used at a maximum enrichment of 5.0 wt% U-235.

At pre-EPU conditions, no credit is taken for soluble boron for spent fuel storage. The licensing basis is being revised for EPU conditions such that spent fuel storage will rely on crediting sufficient boron concentrations. Improved Technical Specifications (ITS) 3.7.14 and 4.3.1.1 are being revised to reflect the changes to licensing basis.

Description of Analyses and Evaluations

After EPU implementation, Mark-B-HTP fuel will continue to be used at a maximum enrichment of 5.0 wt% U-235. The impact of EPU operation on the core operating parameters and the axial burnup profile used in the current Spent Fuel Storage Facility criticality analysis is shown in Table 2.8.6.2-1 and Figure 2.8.6.2-1, respectively. These parameter changes were evaluated for impact on the current criticality analysis of the Spent Fuel Storage Facility with no change in computer codes or application methodologies.

Results

Studies performed in support of the evaluation discussed above demonstrate that continued use of a uniform axial burnup profile remains conservative for EPU conditions.

The results of the evaluation of the impact on the current criticality analysis noted above are summarized in Table 2.8.6.2-2 for the normal condition and in Table 2.8.6.2-3 for the accident condition (defined as a misloaded assembly of 5.0 wt% U-235 enrichment). These results show that if current administrative controls governed by CR-3 Improved Technical Specifications (ITS) 3.7.15 (which include appropriate configurations based on enrichment and burn-up) are coupled with boron credit, 203 ppm for normal conditions and 571 ppm for accident conditions, the Spent Fuel Storage Facility design meets the criticality design basis. Figures 2.8.6.2-2 and 2.8.6.2-3 are included as clarifying references to demonstrate how the ITS limits are practically implemented.

To provide added margin to the boron credit required in the evaluation, ITS 3.7.14 will continue to require ≥ 1925 ppm boron in the spent fuel pools. Additionally, the APPLICABILITY of LCO 3.7.14 is being revised to require maintaining this limit when fuel assemblies are stored in the spent fuel pool. The pools are normally maintained at a boron concentration of ≥ 2000 ppm.

2.8.6.2.3 Conclusion

CR-3 has reviewed the analyses related to the effects of the proposed EPU on the Spent Fuel Storage capability . The licensing basis is being revised for EPU conditions such that spent fuel storage will rely on crediting sufficient boron concentrations. CR-3 concludes that the analyses and supporting amendments to the ITS adequately accounts for the effects of the proposed EPU on the spent fuel rack and criticality analyses. Additional margin, conservative restrictions are also being proposed to impose the ITS 3.7.14 spent fuel pool boron concentration when fuel assemblies are stored in the spent fuel pool. Based on this, CR-3 concludes that the Spent Fuel Storage will meet the requirements of FSAR Section 1.4.66 and 9.6.1.2.2 under the proposed EPU conditions. Therefore, CR-3 finds the proposed EPU acceptable with respect to Spent Fuel Storage.

2.8.6.2.4 References

None

Table 2.8.6.2-1
Analysis Input Comparison

Parameter	Current	EPU
Soluble Boron Concentration (cycle average), ppm	1000	1180
Reactor Specific Power, MW/MTU	30.0	35.0
Core Average Fuel Temperature, ºF	1238	1349
Core Average Moderator Temperature at the Top of the Active Region, ºF	604	611

Table 2.8.6.2-2
Summary of EPU Criticality Safety Analysis for Normal Conditions

Spent Fuel Pool and Arrangement	Required Boron (resultant K_{eff} (1))	Unborated K_{eff} (2)
Pool A – Uniform Loading(3)	141 ppm (0.945)	0.9628
Pool A – Checkerboard Loading(3)	0 ppm (0.9449)	0.9449
Pool B – Uniform Loading(3)	203 ppm (0.945)	0.9694
Pool B – Peripheral Storage(3)	193 ppm (0.945)	0.9634
Pool B - New Fuel Surrounded by empty storage cells(3)	68 ppm (0.945)	0.9528

(1) Regulatory Limit (K_{eff} < 0.95)
(2) Regulatory Limit (K_{eff} < 1.0)
(3) Loading Pattern Illustrated in Figures 2.8.6.2-4 and -5

Table 2.8.6.2-3
Summary of EPU Criticality Safety Analysis for Accident Conditions (Misloaded Assembly)

Spent Fuel Pool and Arrangement	Required Boron (resultant K_{eff} [1])
Pool A - Uniform Loading[2]	198 ppm (0.945)
Pool A - Checkerboard Loading[2]	159 ppm (0.945)
Pool B - Uniform Loading[2]	556 ppm (0.945)
Pool B - Peripheral Storage[2]	571 ppm (0.945)
Pool B - New Fuel Surrounded by empty storage cells[2]	77 ppm (0.945)

[1] Regulatory Limit (K_{eff} < 0.95)
[2] Loading Pattern Illustrated in Figures 2.8.6.2-2 and 2.8.6.2-3

Figure 2.8.6.2-1
Comparison of Pre-EPU and EPU Axial Burnup Distributions

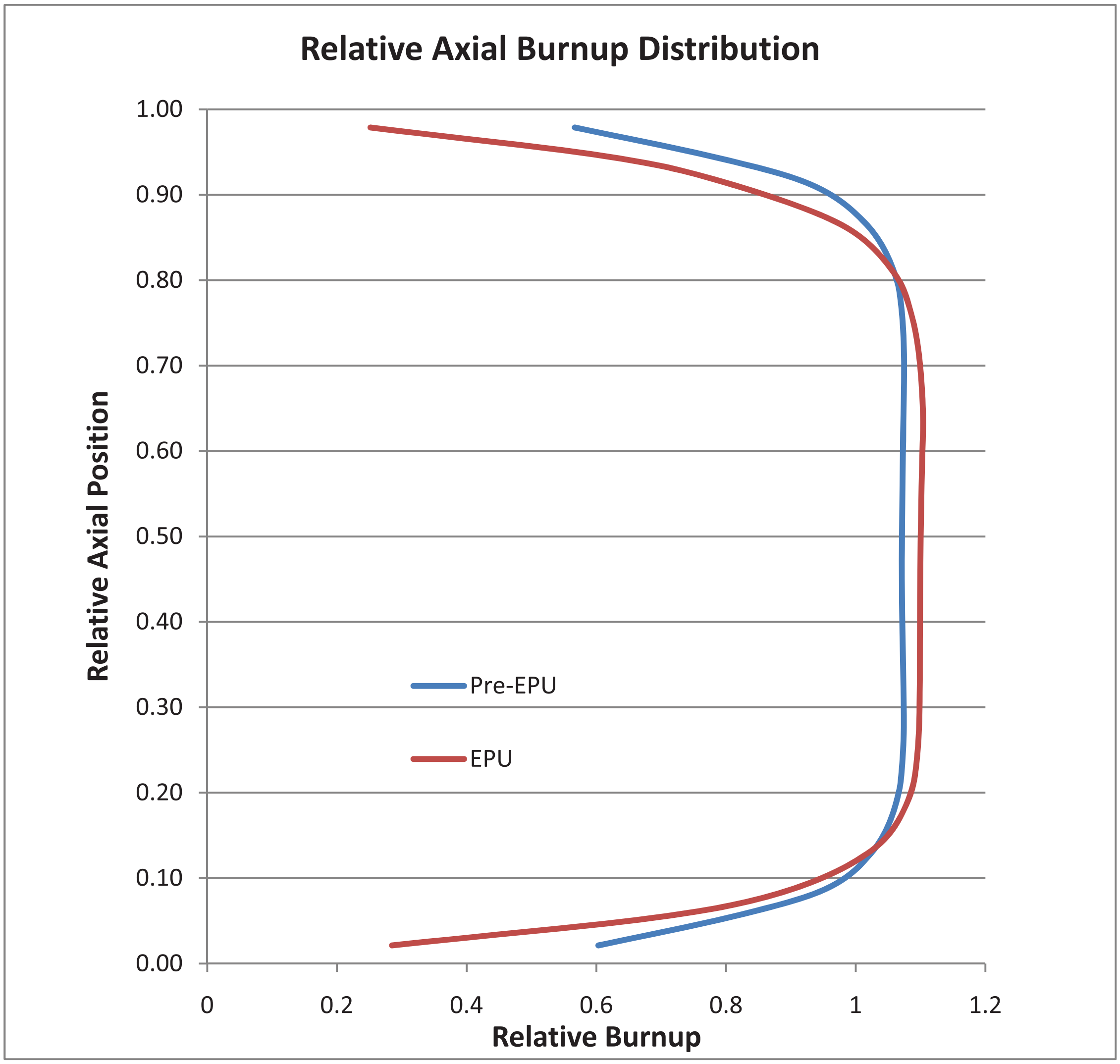


Figure 2.8.6.2-2

Proposed "A" Pool Layout

East

	AA	AB	AC	AD	AE	AF	AG	AH	AK	AL	AM	AN	AO	AP	AR	AS	
34			A	A	A	A	A	A	A	A	A	A	A	A	A	A	34
33	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	33
32	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	32
31	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	31
30	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	30
29	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	29
28	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	28
27	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	27
26	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	26
25	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	25
24	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	24
23	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	23
22	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	22
21	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	21
20	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	20
19	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	19
North																	
18	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	18
17	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	17
16	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	16
15	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	15
14	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	14
13	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	A	13
12	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	12
11	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	11
10	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	10
09	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	09
08	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	08
07	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	07
06	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	06
05	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	05
04	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	04
03	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	03
02	B	F	B	F	B	F	B	F	B	F	B	F	B	F	B	F	02
01	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	01
	AA	AB	AC	AD	AE	AF	AG	AH	AK	AL	AM	AN	AO	AP	AR	AS	

NOTES:

1. Spent fuel categories A, B and F are defined in ITS Bases 3.7.15. The category depends on initial enrichment and fuel burnup.
2. Notation "E" represents an empty water cell.

Figure 2.8.6.2-3
Proposed "B" Pool Layout

East Wall

	BA	BB	BC	BD	BE	BF	BG	BH	BK	BL	BM	BN	BO	BP	BR	BS	BT	
B59	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	BP	E	B59
B58	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B58
B57	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B57
B56	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B56
B55	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B55
B54	B	B	E	E	E	E	E	E	B	B	B	B	B	B	B	BP	E	B54
B53	B	B	E	F	E	E	F	E	B	B	B	B	B	B	B	BP	E	B53
B52	B	B	E	E	E	E	E	E	B	B	B	B	B	B	B	BP	E	B52
B51	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B51
B50	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B50
B49	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B49
B48	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B48
North																		
B47	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B47
B46	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B46
B45	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B45
B44	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B44
B43	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B43
B42	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B42
B41	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B41
B40	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B40
B39	B	B	B	B	B	B	B	B	B	B	B	B	B	B	B	BP	E	B39
B38	B	B	B	B	B	B	B	B	B	B	E	E	E	E	E	E	E	B38
B37	BP	BP	BP	BP	BP	BP	BP	BP	BP	B	E	F	E	E	E	E	E	B37
B36	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	E	B36
	BA	BB	BC	BD	BE	BF	BG	BH	BK	BL	BM	BN	BO	BP	BR	BS	BT	

NOTES:

1. Spent fuel categories B, BP and BE are defined in ITS Bases 3.7.15. The category depends on initial enrichment and fuel burnup.
2. Notation "E" represents an empty water cell.

2.8.7 Additional Reactor Systems

2.8.7.1 Loss of Decay Heat Removal at Mid-loop

2.8.7.1.1 Regulatory Evaluation

NRC Generic Letter (GL) 88-17, "Loss of Decay Heat Removal," (Reference 1) identified actions to be taken by all PWR licensees to preclude Loss of Decay Heat Removal (DHR) during non-power operations. These actions included operator training and the development of procedures and hardware modifications, as necessary, to prevent the Loss of Decay Heat Removal during reduced reactor coolant inventory operations, to mitigate accidents before they progress to core damage, and to control radioactive material if a core damage accident should occur. Procedures and administrative controls were required to address reduced inventory operations to ensure that for applicable plant designs, all hot legs are not blocked by nozzle dams unless a vent path is provided that is large enough to prevent pressurization and loss of water from the reactor vessel. Instrumentation was required to provide continuous core exit temperature and reactor water level indication. Sufficient equipment was required to be maintained in an operable or available status so as to mitigate the Loss of DHR or Loss of Reactor Coolant System (RCS) inventory should such an event occur during mid-loop or reduced inventory conditions.

<u>CR-3 Current Licensing Basis</u>

The adequacy of the CR-3 design and the actions taken in response to GL 88-17 are described in the CR-3 Improved Technical Specifications (ITS) Sections 3.4.7 and 3.9.5 and the associated Bases. The following actions have been taken by CR-3 to conform to the recommendations made by the NRC in GL 88-17:

- Thermal-hydraulic analyses have been performed to form the basis for the required operator actions, which have been implemented in procedures and administrative controls, and for the equipment required to be available for providing core cooling in the event DHR cooling is lost.
- Two independent temperature indications and water level indicators have been provided to monitor mid-loop operation conditions.
- During periods of reduced inventory, one additional offsite power source and one additional emergency diesel generator (EDG) are required to be maintained available.
- Multiple methods for filling the RCS, and maintaining pool temperature have been identified in CR-3 administrative instructions and operating procedures.

2.8.7.1.2 Technical Evaluation

<u>Introduction</u>

The Loss of DHR at reduced RCS inventory conditions was evaluated to determine the time to reach saturation, the RCS inventory boil-off rate, the minimum makeup rate required to match boil-off, and the time to reach 200°F. The RCS loops are considered "not-filled" when reactor coolant water level is drained down, as might be the case for refueling or maintenance on the RCS.

Following shutdown, the reactor core is cooled by forced flow maintained by the DHR System. If the DHR System becomes inoperable, forced flow will cease and the primary coolant in the reactor region will begin to heat up.

- If the RCS is open, the coolant will heat up to the saturation temperature corresponding to the reactor building pressure, and evaporate to the containment. The system is considered open if any of the following conditions exist:
    - RV head is removed,
    - One 5-inch upper hand hole removed from each OTSG,
    - Other openings in the RCS pressure boundary equivalent to 7-inch diameter (40 square inches). Openings in the pressurizer are excluded.
        - Handholes left OPEN, or
        - Handholes connected to ventilation systems with offset ventilation adapters.
- If the RCS is closed, the coolant will initially heat up to the saturation temperature corresponding to the RCS pressure. Additional heating will cause the RCS to pressurize until relief valves open. The lowest pressure for significant pressure relief is the PORV low temperature overpressure protection (LTOP) pressure. The RCS is considered closed if none of the preceding open conditions exist, or if the status of the RCS pressure boundary is unknown.

In either scenario, RCS coolant will boil off, and if uncompensated, has the potential to uncover the core.

The condition of the RCS when fuel is in the reactor vessel and the RCS water level is lower than the top of the flow area at the junction of the hot legs with the reactor vessel is defined as mid-loop. At CR-3, mid-loop corresponds to an RCS water level of 129 feet 6 inches.

Restrictions are placed on mid-loop operations at CR-3 prior to entering reduced inventory conditions which is defined as an RCS inventory that results in a reactor vessel water level lower than three feet below the RV flange (≤ 132 feet).

Reductions in RCS inventory are termed reduced reactor vessel inventory operations, and result in additional concerns based on operating experience. Equipment redundancy and surveillance tests during this configuration ensure that subcooling is maintained in the reactor vessel. Validation Ref: ITS 3.4.6 and 3.4.7 SRs and associated Bases.

<u>Description of Analyses and Evaluations</u>

The CR-3 specific design analysis, applicable sections of the CR-3 ITS, and administrative and operating procedures applicable to reduced inventory operations have been reviewed with respect to the EPU. The evaluation of the Loss of DHR at Mid-loop operation incorporated the following analysis input and assumptions.

- All heat transfer processes between liquid volumes are adiabatic. No credit is taken for heat transfer outside volume boundaries, such as through ambient losses. This is a conservative assumption as this will minimize the time to core uncovery.

- The heat capacities of most metal components in the RCS are not included in the calculations. This is a conservative assumption as it yields a minimum time to boil and time to core uncovery.

- 20% of the steam generator tubes are modeled as plugged. Maximum tube plugging is conservative as it minimizes the volume and the time to boil/uncover.

- In the time to boil calculations, some liquid volumes, such as the volume within the control rod guide tubes, are conservatively modeled as steel since the product of its density and specific heat is slightly less than that of water.

- For conditions where the water level is below the flange elevation (135 ft), it is conservatively assumed that the time to boil will be based on the available heat sinks in the core region (liquid and metal).

- For conservatism, the largest proposed fuel loading at the EPU conditions was assumed. The fuel isotopic compositions were calculated in the source term calculation for bounding fuel enrichments.

- The enrichment yielding the highest decay heat was used in the analysis as it is conservative for the time to boil calculations.

- Results are calculated for a Loss of DHR occurring 24 hours after reactor shutdown for a full (177 assembly) core. Alternate decay times ranging from 1 hour to 60 days after shutdown are evaluated using a decay heat correction factor.

The acceptance criteria for the Loss of DHR at Mid-loop includes maintaining core cooling and protecting the reactor core until DHR can be returned to service. The design analysis provides time estimates to reach saturation conditions. The results of this analysis are utilized by CR-3 plant staff in procedural guidance for responding to a Loss of Decay Heat Removal at Mid-loop.

The time to boil, time to reach 200°F, time to saturation, and time to core uncovery are determined as a function of initial liquid level and initial bulk fluid temperature. The time to core uncovery is sub-divided into the following parts:

- Time to Boil (t_b)

This is the time for boiling to occur locally in the core region following the loss of forced flow.

- Time to reach 200°F (t_r)

When the time to local boiling (t_b) is reached and steam exits the core region, the voids are filled with liquid from the upper plenum, hot legs, cold legs, and/or the refueling canal. The temperature of this liquid is approximately the bulk fluid temperature. Therefore, credit is taken to heat this fluid to 200°F. This is the time required by the liquid not included in the time to boil calculations to reach 200°F. The time to 200°F is used as a benchmark in the CR-3 operating procedures for when steam release can be conservatively expected.

- Time to Saturate (t_s)

Time to saturate denotes the time interval required by the liquid in the remaining components (those not included in the time to boil calculations) to reach saturation.

- Time to Boil-Off

This time interval denotes the time required for all liquid volumes to change phase from liquid to steam.

- Time to Core Uncovery

This is the sum of the all the time intervals defined above.

Results

As a result of the EPU, the residual core decay heat at a given time after shutdown will increase in approximate proportion to the power increase due to the EPU. The EPU increase in decay heat will reduce the time to boil and the time to core uncovery following a postulated Loss of DHR cooling. Analyses have been performed to determine the time to boil, time to reach 200°F, the time to reach saturation, time to core uncovery, and makeup and boil-off rates. The analysis results for the EPU will be incorporated into the CR-3 operating procedures, and operator training as discussed in Section 2.11, Human Performance.

The range of results calculated for a decay time of 24 hours, at the pre-EPU and the post-EPU conditions, is shown in Table 2.8.7.1-1. The results are adjusted in plant procedures to reflect decay times from 1 day to 60 days after shutdown using a decay heat correction factor. Instrumentation that provides continuous core exit temperature and reactor water level indication is unchanged for the EPU.

2.8.7.1.3 Conclusion

Existing instrumentation that is used to monitor the RCS level, water temperature, and DHR performance during previous and current mid-loop operation is sufficient and no additional instrumentation is required for monitoring mid-loop operation at the EPU conditions. The CR-3 design analyses that form the basis for the existing guidance provided to the operators for mid-loop operation have been revised to incorporate the impact of the higher EPU decay heat on existing curves and tables. A reassessment of existing plant operational curves and operator actions will be performed consistent with the requirements of GL 88-17. All plant specific changes to the CR-3 operational procedural guidance required to support operation at the EPU conditions will be implemented prior to going to mid-loop operation. These actions ensure that CR-3 will continue to conform to the recommendations made by the NRC in GL 88-17 with respect to Loss of Decay Heat Removal at Mid-loop.

2.8.7.1.4 References

1. US NRC Generic Letter 88-17, “Loss of Decay Heat Removal,” October 17, 1988.

Table 2.8.7.1-1 – Description of Calculated Results for Decay Time = 24 hours

Title of Tables for CR-3 Operating Procedures	**RCS Pressure, psia**	**Initial RCS Temp. Range, °F**	**RCS Water Level Range, feet**	**Pre-EPU Results, Time Range (minutes) [2]**	**Post-EPU Results, Time Range (minutes) [2]**
Time to Boil with Reactor Head Off	14.7 psia	60 – 200°F	125 – 173 ft	0.96 – 219.81 min	0.72 – 164.93 min
Time to Boil with RCS Vented	14.7 psia	60 – 200°F	125 – 173 ft	0.96 – 32.48 min	0.72 – 24.37 min
Time to Boil with RCS Intact	457 psia [1]	60 – 450°F	125 – 173 ft	0.61 – 31.86 min	0.46 – 23.93 min
Time to 200°F with Reactor Head Off	14.7 psia	60 – 190°F	125 – 173 ft	0.86 – 224.03 min	0.65 – 168.41 min
Time to 200°F with RCS Vented, Nozzle Dams Installed	14.7 psia	60 – 190°F	125 – 173 ft	0.86 – 53.01 min	0.65 – 39.85 min
Time to 200°F with RCS Vented, Nozzle Dams Not Installed	14.7 psia	60 – 190°F	125 – 173 ft	0.86 – 66.38 min	0.65 – 49.90 min
Time to 200°F with RCS Intact	457 psia [1]	60 – 190°F	125 – 173 ft	0.85 – 76.08 min	0.65 – 58.02 min
Time to Saturation with Reactor Head Off	14.7 psia	60 – 200°F	125 – 173 ft	1.03 – 243.26 min	0.77 – 182.53 min
Time to Saturation with RCS Vented, Nozzle Dams Installed	14.7 psia	60 – 200°F	125 – 173 ft	1.03 – 57.56 min	0.77 – 43.19 min
Time to Saturation with RCS Vented, Nozzle Dams Not Installed	14.7 psia	60 – 200°F	125 – 173 ft	1.03 – 72.08 min	0.77 – 54.08 min
Time to Saturation with RCS Intact	457 psia [1]	60 – 450°F	125 – 173 ft	0.66 – 216.25 min	0.49 – 162.53 min
Time to Core Uncovery with Reactor Head Off	14.7 psia	60 – 200°F	125 – 173 ft	6.63 – 1681.89 min	4.97 – 1261.88 min
Time to Core Uncovery with RCS Vented, Nozzle Dams Installed	14.7 psia	60 – 200°F	125 – 173 ft	6.63 – 314.31 min	4.97 – 254.64 min
Time to Core Uncovery with RCS Vented, Nozzle Dams Not Installed	14.7 psia	60 – 200°F	125 – 173 ft	6.63 – 444.34 min	4.97 – 333.37 min
Time to Core Uncovery with RCS Intact	457 psia [1]	60 – 450°F	125 – 173 ft	4.44 – 516.99 min	3.32 – 387.93 min

(1) RCS pressure for calculations with RCS intact corresponds to the PORV setpoint for LTOP protection in Section 2.8.4.3.

(2) The minimum time in each range occurs for the highest initial temperature and lowest level. The maximum time occurs for the lowest initial temperature and highest level. Results are calculated for 10°F temperature increments and 1 ft level increments over the specified ranges.

Table 2.8.7.1-2 – Vent Diameter as a function of time at constant ΔP

Time (hr)	Vent Diameter (in) Pre-EPU 0.25 psid	Vent Diameter (in) Pre-EPU 21.41 psid	Vent Diameter (in) Post-EPU 0.25 psid	Vent Diameter (in) Post-EPU 21.41 psid
10	22.76	6.19	40.32	13.69
20	20.77	5.65	24.05	8.17
50	18.14	4.93	21.03	7.14
100	16.08	4.37	18.40	6.25
240	13.45	3.66	15.18	5.15

2.9 Source Term for Radiological Consequences Analyses

2.9.1 Source Term for Radwaste System Analyses

2.9.1.1 Regulatory Evaluation

CR-3 reviewed the radioactive source term associated with the EPU to ensure the adequacy of the sources of radioactivity used by CR-3 as input to calculations and to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The CR-3 review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the FSAR related to Liquid Waste Management Systems and Gaseous Waste Management Systems.

The NRC's acceptance criteria for Source Term for Radwaste System Analyses are based on:

- GDC-60, insofar as it requires that the plant design include a means to control release of radioactive effluents;
- 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; and
- 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet "as low as reasonably achievable" criterion.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR Part 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment, insofar as it requires that the plant design include a means to control release of radioactive effluents, [GDC-60].

Additionally, FSAR Chapter 11 provides criteria for the liquid, gaseous, and solid waste systems insofar as it requires that these systems are designed with sufficient capacity to collect, process, and release effluents in a controlled manner in accordance with 10 CFR Part 20 and 10 CFR Part 50 Appendix I during normal operation.

2.9.1.2 Technical Evaluation

Introduction

The normal plant operational source terms establish the long-term concentrations of principal radionuclides in the plant fluid streams as input for subsequent prediction of the expected release of radioactive materials from various effluent streams. The fluid streams of the plant are the reactor primary coolant and the secondary steam generator water and steam. The normal operations source terms serve as input to assessments of the projected normal plant effluent released to the environment.

The principle cycle design parameters used in the calculation of tritium generation are described in Table 2.9.1-1. The values for the release of ternary fission tritium from the fuel through clad defects and clad diffusion are based on the expected release values based on experimental results for zircaloy based cladding.

The assumptions and input parameters that served as the basis for the determination of the primary and secondary radiation sources are summarized in Table 2.9.1-2. The results of the normal plant operation source calculations serve as input to establish the long-term, expected concentrations of principal radionuclides in plant effluents. The results of the effluent activity and concentration calculations are summarized in Section 2.10.1, Occupational and Public Radiation Doses.

Description of Analyses and Evaluations

At the EPU conditions, the calculations assume a core thermal power of 3014 MWt with an additional power uncertainty of 0.4% (3026.1 MWt total). Other inputs and assumptions impacted by the EPU are noted in Table 2.9.1-2.

The normal or expected activity concentrations in the primary and secondary sides were based on the methodology of American Nuclear Society Standard ANSI/ANS-18.1-1999 (Reference 2). The methodology applies adjustment factors to a set of "reference value" concentrations if plant parameters deviate from a prescribed set of nominal values. Application of this Standard is consistent with the methodology included in Revision 1 of the PWR-GALE code (Reference 3) that is considered by the NRC in its review of expected plant radioactive effluents for all light water reactor (LWR) plants. Normal sources for CR-3 are established by appropriate scaling by thermal power and other pertinent EPU parameters as outlined in the Standard. The methodology also considers a "Y" factor, defined as the ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system to the purification system (not including the boron recovery system).

Tritium generation in the Reactor Coolant System was calculated with the application of ORIGEN2 based on the EPU parameters. The analyses included tritium generation from (1) neutron reactions with soluble boron, (2) neutron reactions with lithium used for pH control in the reactor coolant, (3) neutron capture by natural deuterium in the reactor coolant, (4) ternary fission with release to the coolant through fuel clad defects, (5) ternary fission with release to the coolant by diffusion through the clad material, and (6) neutron reaction with ^{3}He (daughter product of ^{3}H decay).

Results

A summary of the results of the tritium generation analysis is given in Table 2.9.1-3. The calculated total tritium generation rates are consistent with the CR-3 plant effluent releases during recent years. The major source of tritium is the B-10 (n, 2α) He-3 reaction with the soluble boron in the coolant. The EPU value is considerably higher than those in the current FSAR. This is expected since the boron concentrations in the reactor coolant for the EPU design are significantly higher than those of earlier core designs.

The calculated primary and secondary radiation sources are summarized in Table 2.9.1-4. The results of the normal plant operation source calculations served as input to establish the long-term concentrations of principal radionuclides in the fluid streams of the plant for subsequent application in estimating the expected release of radioactive materials from various effluent streams (as discussed in Section 2.10.1).

2.9.1.3 Conclusion

CR-3 has performed an assessment of the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of radionuclides are appropriate for the evaluation of the radioactive waste management systems. CR-3 further concludes that the proposed radioactive source term meets the requirements of FSAR Section 1.4.70, 10 CFR Part 20 and 10 CFR Part 50 Appendix I. Therefore, CR-3 finds the proposed EPU acceptable with respect to Source Term for Radwaste System Analyses.

2.9.1.4 References

1. ASME Steam Tables, Sixth Edition.

2. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors" and ANS/ANSI-18.1, ERRATA, December 1, 2005.

3. USNRC, NUREG-0017, Rev. 0, Calculation of Releases of Radioactive Materials in Gases & Liquid Effluents from Pressurized Water Reactors, April 1976.

Table 2.9.1-1: Parameters Used in the Calculation of Tritium Production in the Reactor Coolant – Assumptions

Parameter	EPU Value
Mass of U per FA	489.52 kg
Core thermal power	3026.1 MWth
U-235 Enrichment	4.95%
Full Power cycle length	685 days
Fuel clad OD	0.43 in.
Active fuel length	143 in.
Fuel rod volume	20.76653 in^3
No. fuel rods/FA	208
FA rod volume/FA	4319.437 in^3
FA pitch	8.587 in.
FA volume	10544.33 in^3
FA coolant volume	6224.893 in^3
Coolant density	0.71671 gm/cc
FA coolant mass	73109.94 gm
Core coolant mass	1.29405E+07 gm
Boron coolant concentration	981.75 ppm
Boron-10 isotopic abundance	19.78%
Li coolant concentration	3 ppm
Li^6 fraction	0.10%
Percent of ternary fission tritium diffusing through clad	0.50
Percent of ternary fission tritium released through clad defects	0.25

Table 2.9.1-2: ANSI/ANS-18.1-1999 Normal Source Input Parameters

Parameter	Symbol	Element Class [1]	CR-3 EPU	CR-3 Pre-EPU	OTSG Ref. Plant [2]	Reference
Thermal Power (MWt)	P	N/A	3026.1	2609	3400	Note 3.
Steam flow rate (lbm/hr)	FS	N/A	12.86E+6	10.98E+07	1.50E+07	
Volume of water in RCS (including pressurizer) ft^3	N/A	N/A	11511	11511	N/A	
RCS Pressure (psia)	N/A	N/A	2170	2170	N/A	
RCS Average Temperature (Tave) (ºF)	N/A	N/A	582	580	N/A	
RCS Specific Volume @ Tave (ft^3/lbm)	N/A	N/A	0.02245	0.02240	N/A	ASME Tables (Reference 1)
Mass of water in RCS (lbm)	WP	N/A	5.128E+05	5.139E+05	5.50E+05	Calculated (Note 7)
Mass of water in SGs (lbm)	WS	N/A	58068 * 2	51634 * 2	1.00E+05	
Normal letdown flow rate (gpm)	N/A	N/A	45	45	75	Note 10
Density of Letdown water (lbm/ft^3) at T = 120 ºF	N/A	N/A	61.71	61.71	N/A	ASME Tables (Reference 1)
RCS letdown purification flow (lbm/hr) (normal)	FD	N/A	2.227E+4	2.227E+4	3.70E+04	Calculated
RCS letdown flow rate for boron control (lbm/hr)	FB	N/A	5.00E+02	0	5.00E+02	ANSI/ANS-18.1-1999 & Errata, Table 2 (Reference 2)
Purification system cation demineralizer flow (lbm/hr)	FA	N/A	0	0	3.70E+03	Note 5
Steam generator blow down flow rate (total) (lbm/hr)	FBD	N/A	Note 8	Note 8	N/A	ANSI/ANS-18.1-1999 & Errata, Table 3 (Reference 2)
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system	NBD	N/A	Note 8	Note 8	N/A	ANSI/ANS-18.1-1999 & Errata, Table 3 (Reference 2)
Ratio of condensate demineralizer flow rate to total steam flow	NC	N/A	1.0	1.0	6.50E-01	

Table 2.9.1-2: ANSI/ANS-18.1-1999 Normal Source Input Parameters (cont'd)

Parameter	Symbol	Element Class [(1)]	CR-3 EPU	CR-3 Pre-EPU	OTSG Ref. Plant [(2)]	Reference
Fraction of noble gas activity in letdown stream which is not returned to RCS (excluding boron control system)	Y	N/A	0	0	0.000	ANSI/ANS-18.1-1999 & Errata, Table 2 (Reference 2)
Fraction of material removed in passing through the cation demineralizers	NA	1, 2, 4, 5	0	0	0.000	
		3	0.00	0.00	0.900	
		6	0.00	0.00	0.900	
Fraction of material removed in passing through the purification demineralizers (Note 4)	NB	1, 4, 5	0	0	0.000	
		2	0.99	0.99	0.990	
		3	0.2857	0.00	0.500	
		6	0.99	0.99	0.980	
Fraction of activity removed in passing through the condensate demineralizers	NX	1, 4, 5	0	0	0.000	ANSI/ANS-18.1-1999 & Errata, Table 9 (Reference 3)
		2, 6	0.90	0.9999	0.900	
		3	0.50	0.50	0.500	
Ratio of concentration in steam to that in the steam generator	NS	1	N/A	N/A	N/A	ANSI/ANS-18.1-1999 & Errata, Table 9 (Reference 2)
		2, 3, 4, 5, 6	1.0	1.0	1.0	
Primary Coolant Removal Rates (hr^{-1})	R_n	1	9.75E-02		9.0E-04	ANSI/ANS-18.1-1999 & Errata, Table 9 (Reference 2) (Note 9)
		2	4.30E-02		6.7E-02	
		3	1.31E-02		3.7E-02	
		4	0.0		0.0	
		5	n/a		n/a	
		6	4.30E-02 9.75E-04 (Mo, Y)		6.6E-02	
Secondary Coolant Removal Rates (hr^{-1})	r_n	1	n/a		n/a	ANSI/ANS-18.1-1999 & Errata, Table 9 (Reference 2) (Note 9)
		2	n/a		n/a	
		3	99.66		88 [(9)]	
		4	55.37		49 [(9)]	
		5	n/a		n/a	
		6	99.66		88 [(9)]	

Notes:

1. Class of Elements per ANSI/ANS-18.1-1999 are:

1 = Noble Gases
2 = Br, Iodines
3 = Rb, Cs
4 = N-16
5 = Tritium
6 = Others

2. ANSI/ANS-18.1-1999 (Reference 2) provides reference plant data for OTSG plants.

3. Based on a RTP = 3014 MWt + 0.04% uncertainty.

4. The DF for MU demineralizer has a DF = 100 except for Mo and Y which have a removal fraction of 0.00 and 1.4 for Cs and Rb.

5. The value FA was assumed to be 0% of FD.

6. Conservative assumptions based on reference plant described in Reference 5.

7. Calculated by dividing the volume of the RCS (including the pressurizer) by specific volume of the RCS at Tave. The Reactor Coolant System (including the pressurizer and surge lines) volume at hot conditions of 11511 ft^3 is specified in the CR-3 FSAR.

8. Steam generator blowdown is used for cleanup of the secondary coolant. However, blowdown is only used at $\leq$ 15% RTP. Also, blowdown is not modeled for plants with OTSGs (Reference 2). Therefore for conservatism, blowdown was neglected.

9. Removal rates for each element class are calculated using ANSI/ANS-18.1-1999 methodology (Table 9, Reference 2) for the primary and secondary coolants. The removal rates for reference plant were recalculated using Table 9 methodology and are consistent with NUREG-0017, Rev. 0, Table 2-6 [Reference 3].

10. Based on normal letdown flow rate, the density of water at 120 °F, and the application of appropriate conversion factors.

Table 2.9.1-3: Calculation of Tritium Production in the Reactor Coolant

Tritium Source	EPU RC Activity, Ci	Pre-EPU RC Activity, Ci
Ternary Fission - Clad defects	79	179
Ternary Fission - Clad diffusion	158	
Soluble Lithium - Li^6 (n, α) H^3, Li^7 (n, nα) H^3	136	144
Boron 10 - B^{10} (n, 2α) H^3	1304	132
Deuterium - H^2 (n, γ) H^3	9	--
Total	**1686**	**455**

Table 2.9.1-4: Crystal River Unit 3 – EPU Normal Plant Operations Sources (µCi/gm)

ANSI/ANS Element Class	Radionuclide	Primary Coolant (µCi/gm)	Secondary Coolant Steam (µCi/gm)
1	KR 85M	1.527E-01	3.783E-08
	KR 85	3.829E-01	9.243E-08
	KR 87	1.623E-02	3.784E-09
	KR 88	1.718E-02	4.229E-09
	XE 131M	6.833E-01	1.637E-07
	XE133M	6.651E-02	1.662E-08
	XE133	2.740E-02	6.611E-09
	XE135M	1.241E-01	3.005E-08
	XE135	8.106E-01	2.002E-07
	XE137	3.245E-02	7.903E-09
	XE138	5.822E-02	1.447E-08
2	BR84	1.554E-02	1.367E-08
	I131	2.876E-03	1.744E-09
	I132	6.120E-02	5.233E-08
	I133	3.249E-02	2.275E-08
	I134	9.815E-02	8.349E-08
	I135	6.088E-02	4.853E-08
3	RB88	1.832E-01	4.574E-07
	CS134	9.983E-05	1.213E-10
	CS136	2.133E-03	2.729E-09
	CS137	1.433E-04	1.668E-10
4	N16	4.000E+01	1.000E-06
5	H 3	1.000E+00	1.000E-03
6	NA 24	5.646E-02	8.340E-08
	CR 51	4.508E-03	5.231E-09
	MN 54	2.344E-03	2.654E-09
	FE 55	1.759E-03	2.047E-09
	FE 59	4.376E-04	5.080E-10
	CO 58	6.722E-03	7.582E-09
	CO 60	7.770E-04	9.098E-10
	ZN 65	7.471E-04	8.340E-10
	SR 89	2.043E-04	2.350E-10
	SR 90	1.759E-05	2.047E-11
	SR 91	1.099E-03	1.592E-09
	Y91M	4.732E-04	7.363E-10
	Y91	2.250E-04	9.098E-12
	Y93	7.758E-03	7.052E-09

Table 2.9.1-4: Crystal River Unit 3 – EPU Normal Plant Operations Sources (μCi/gm) (cont'd)

ANSI/ANS Element Class	Radionuclide	Primary Coolant (μCi/gm)	Secondary Coolant Steam (μCi/gm)
6	ZR 95	5.698E-04	6.596E-10
	NB 95	4.079E-04	4.701E-10
	MO 99	4.076E-02	1.061E-08
	TC 99M	5.140E-03	7.583E-09
	RU103	1.093E-02	1.289E-08
	RU106	1.319E-01	1.516E-07
	AG110M	1.904E-03	2.199E-09
	TE129M	2.767E-04	3.184E-10
	TE129	2.373E-02	3.870E-08
	TE131M	1.931E-03	2.502E-09
	TE131	7.449E-03	1.140E-08
	TE132	2.344E-03	2.881E-09
	BA140	1.873E-02	2.199E-08
	LA140	3.300E-02	4.246E-08
	CE141	2.184E-04	2.502E-10
	CE143	3.635E-03	4.701E-09
	CE144	5.714E-03	6.596E-09
	W187	3.150E-03	4.246E-09
	NP239	2.976E-03	3.488E-09

2.9.2 Radiological Consequence Analyses

2.9.2.1 Regulatory Evaluation

CR-3 reviewed the Design Basis Accident (DBA) Radiological Consequence Analyses. The Radiological Consequences Analyses reviewed are the LOCA, Fuel Handling Accident (FHA), Control Rod Ejection accident (REA), Main Steam Line Break (MSLB), Steam Generator Tube Rupture (SGTR), and Locked-Rotor Accident (LRA). The CR-3 review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used for the calculation of total effective dose equivalent (TEDE).

The NRC's acceptance criteria for Radiological Consequence Analysis using the Alternative Source Term are identified in:

- 10 CFR Part 50.67, insofar as it sets standards for radiological consequences of postulated accidents; and
- GDC-19, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR Part 50.2, for the duration of the accident.

<u>CR-3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR Part 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable design, construction, and operation of CR-3.

The following are applicable criteria:

- FSAR Section 1.4.11, Control Room, insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE, as defined in 10 CFR Part 50.2, for the duration of the accident [GDC-19]; and
- FSAR 1.4.70, Control Of Releases Of Radioactivity To The Environment, insofar as it sets standards for radiological consequences of postulated accidents, [10 CFR Part 50.67].

2.9.2.2 Technical Evaluation

<u>Introduction</u>

CR-3 performed Design Basis Accident (DBA) Radiological Consequence Analyses using the guidance in Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. The assumptions are the same as those provided in Reference 1 except as discussed in each event below. The radiological consequence analyses include the following:

- Main Steam Line Break (MSLB),
- Locked-Rotor Accident (LRA),
- Control Rod Ejection Accident (REA),
- Steam Generator Tube Rupture (SGTR),
- Loss of Coolant Accident (LOCA) ,
- Fuel Handling Accident (FHA),
- Letdown Line Rupture (LLR)

CR-3 analysis for each accident considered:

- The sequence of events;
- Models, assumptions, and parameter values used in calculation of the total effective dose equivalent (TEDE).

These results are summarized in Table 2.9.2-1 along with the dose acceptance criteria.

Description of Analyses and Evaluations

Common Input Parameters and Assumptions and Acceptance Criteria

The assumptions and input described in this section are common to all analyses. They are consistent with Reference 1 except as noted. Accident specific input and assumptions are described in detail in the sections that follow.

Core Radionuclide Inventory for CR-3 EPU was determined using the ORIGEN-2 computer code along with extended burnup libraries. A maximum core average radionuclide inventory was calculated from a parametric evaluation with a fuel enrichment of 3.5-5.0 wt% Uranium-235 and burnup steps ranging between 5 and 47 GWD/MTU. The maximum activity for each radionuclide from the parametric cases was selected to provide a maximum core average inventory. The core inventory is listed in Table 2.9.2-2. Pre-EPU core inventory is based on the PWR default library for AST source from RADTRAD (References 6 and 7).

Primary Coolant Concentrations for design-basis applications were developed using the ORIGEN-2 core radionuclide inventory and the escape rate coefficients listed in NUREG-0017, Rev. 0 (Reference 2, Table 2-11) based on more recent experiments shown in Table 2.9.2-3. The Reactor Coolant System (RCS) activity levels were calculated based on continuous reactor coolant purification at a rate of one reactor system volume per day with a zero removal efficiency for Kr, Xe, Mo, H3, and Y and a 99% removal efficiency for all other nuclides except Cs and Rb which have a removal efficiency of 28.6% until equilibrium concentration is achieved (pre-EPU source term did not credit cleanup of Cs and Rb per FSAR Section 11.2.1.3, System Design Evaluation).

Calculations of activity released from the fuel and equilibrium RCS concentrations were determined using a FORTRAN-77 computer code, ELISA2-4 which solves the differential equations for a three-member radioactive chain for release from the fuel to the coolant, removal from the coolant by purification and bleed, and collection on a resin or in a holdup tank. The fission product activity levels in the reactor coolant at the end of the equilibrium core cycle with defective fuel are shown in Table 2.9.2-4.

Except where otherwise noted, the iodine concentrations for EPU analyses were conservatively adjusted to an RCS specific activity limit of 0.35 μCi/gm Dose Equivalent I-131 (DE I-131), rather than 1.0 μCi/gm DE I-131 used in the Reference 1. The proposed Improved Technical Specification change will impose a more restrictive Dose Equivalent I-131 limit of 0.25 μCi/gm (ITS Section 3.4.15).

Secondary Coolant Concentrations were calculated based on the primary-to-secondary leakage of 150 gpd/SG and RCS concentrations. The iodine concentrations were adjusted to the ITS limit of 4.5E-04 μCi/gm DE I-131. Secondary side activities were neglected in the pre-EPU accident analyses and in the EPU SGTR licensing basis analysis presented here.

Dose Conversion Factors (DCFs) for the TEDE offsite and control room dose calculations were extracted from Federal Guidance Reports 11 and 12 (FGR 11 and FGR 12) (References 3 and 4). The DCFs are unchanged from those used in Reference 1.

Atmospheric Dispersion Factors (χ/Q) and Breathing Rates for the EAB, LPZ, and control room are presented in Table 2.9.2-5. The offsite (EAB and LPZ) and TSC atmospheric dispersion factors have been updated from those previously submitted in the Reference 1 based on five years of recent meteorological data and were evaluated using PAVAN and ARCON96. The Main Control Room (MCR) atmospheric dispersion factors are the same as previously submitted in the Reference 1. The MCR atmospheric dispersion factors were calculated using CR-3 site specific meteorological data collected at original licensure (1975) using the Murphy/Campe methodology. The critical meteorological parameter, 95 percentile average wind speed, used for the MCR atmospheric dispersion factor, was verified as conservative in comparison to the 95 percentile average wind speed for the 5 year period from 2003 through 2007.

Control Room radiological analysis parameters and assumptions are summarized in Table 2.9.2-6. Iodine removal efficiencies have been revised from those of the Reference 1. Control room ventilation system filter iodine removal efficiencies are in accordance with Improved Technical Specifications SR 3.7.12 and Regulatory Guide 1.52.

Iodine Spiking is modeled as follows:

> **Pre-Accident Spike**: A reactor transient has occurred prior to the postulated accident and has raised the primary coolant iodine concentration to 21 μCi/gm DEI-131, which is higher than the Technical Specification 48-hour Action limit of 15 μCi/gm DEI-131, allowed for iodine spiking (60 times the proposed limit of 0.25 μCi/gm DE I-131). Pre-EPU raised the primary coolant iodine concentration to the maximum temporary value of 60 μCi/gm DEI-131 permitted by the technical specifications.
>
> **Concurrent Spike:** The primary system transient associated with the accident causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model which assumes the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times (MSLB and LLR) greater than the release rate corresponding to the iodine concentration at an equilibrium value of 0.35 μCi/gm DEI-131, which is higher than the proposed Technical Specification limit of 0.25 μCi/gm DEI-131, allowed for continuous operation. The concurrent iodine spike is not considered if fuel damage is postulated. The assumed iodine spike duration is 8 hours.

Where there are differences in parameters assumed between the radiological and thermal-hydraulics analyses, a conservative value is used to evaluate the Radiological Consequences.

Acceptance Criteria

The CR-3 EPU accident analyses were evaluated to ensure offsite dose and control room dose meet the dose acceptance criteria for offsite doses and control room habitability in 10 CFR Part 50.67.

Description of Analysis and Evaluation

MSLB Radiological Consequences

The MSLB accident considered is the complete severance of a 24-inch main steam line between the steam generator and the turbine outside containment. The Radiological Consequences of a break inside containment are bounded by the consequences of a break outside containment. Thus, only the MSLB outside of containment is considered with regard to the Radiological Consequences. The affected steam generator will rapidly depressurize and release the initial contents of the steam generator to the environment. A reactor trip occurs on high flux specified in the CR-3 EPU MSLB thermal-hydraulic analyses. The affected steam generator pressure is assumed to reach the setpoint for actuation of the reactor trip, turbine trip, and main steam line isolation. High pressure injection (HPI) is assumed to start after Engineered Safeguards Actuation System (ESAS) actuation. At low steam generator pressure, the Emergency Feedwater Initiation and Control (EFIC) would actuate Emergency Feedwater and Main Steam Line Isolation (MSLI) and Main Feedwater Isolation (MFWI). Actuation of the MSLI and MFWI would isolate the affected SG. For this analysis, a single failure of the EFIC trip of Main Feedwater pump is assumed per FSAR Section 14.2.2.1.4 (Case II). This allows more Feedwater to enter the affected SG until the Main Feedwater pump suction valve closure. This results in more negative reactivity insertion following the reactor trip. Following closure of the main Feedwater pump suction valve, the affected SG is assumed to blowdown until it is dry and depressurized. The break is assumed to be downstream of the MSSV; therefore, the affected SG would continue to blowdown releasing the remaining inventory through the break. In addition, a loss of offsite power (LOOP) is assumed concurrent with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by release of steam directly to the environment. Cooldown via the unaffected SG is assumed to continue for the duration of the accident.

No fuel damage is postulated to occur as the result of an MSLB since there is no return to criticality or departure from nucleate boiling (DNB) postulated to occur as discussed in Section 2.8.5.0, Non-LOCA Analyses Introduction, and Section 2.8.3, Thermal and Hydraulic Design. For the MSLB, two iodine spiking cases are considered. The first assumes that a pre-accident iodine spike occurred just before the event and the Reactor Coolant System (RCS) iodine inventory is at 21 μCi/gm Dose Equivalent I-131). The second case assumes the event initiates a concurrent iodine spike. No credit is taken for the operator to isolate the affected steam generator. The unaffected steam generator is then used for cool down for the duration of the accident. This is conservatively assumed to be 30 days. The steam from the unaffected steam generator is assumed to be released, with no holdup, directly from the secondary side. The affected steam generator is assumed to boil dry instantly, releasing the entire liquid inventory and entrained radionuclides through the steam line break to the environment.

Analysis Parameters and Assumptions

The major assumptions and parameters used in the MSLB analysis are itemized in Table 2.9.2-11. The radioactivity transport model is shown in Figure 2.9.2-1.

A primary-to-secondary leak rate of 150 gpd (at cold conditions) is assumed to the unaffected SG, and accident-induced leakage of 1 gpm to the affected SG, for duration of the event (30 days). The affected SG is assumed to boil dry instantly, and remain dry for the duration of the event. The unaffected SG is used for cooldown for duration of the event. Primary-to-secondary leakage, into the affected and unaffected SG, is released directly to the environment, with no credit for retention via partitioning. The elemental and organic iodide and alkalis release assumes a partition of 1.0. No credit is taken for release of the unaffected steam generator via the condenser. Noble gas activity, leaked to the SGs, is assumed to be directly released to the environment.

Comparison to Reference 1

The analysis of the MSLB radiological consequences, for the EPU, is consistent with the analytical methods presented in Regulatory Guide 1.183. Pre-accident and concurrent iodine spikes were modeled for the EPU; however, iodine spiking was not modeled in the pre-EPU. The source term for the pre-EPU conditions was based on an RCS primary coolant concentration for 1% failed fuel fraction. This supersedes the analysis presented in Reference 1 with changes made to reflect the increased power and Regulatory Guide 1.183 clarification per RIS 2006-04. Specific changes between the EPU analysis and pre-EPU analysis include:

- Revised primary and secondary coolant initial water mass and nuclide activity shown in Table 2.9.2-7 and Table 2.9.2-8 based on:
    - RCS iodine concentration limit was lowered from 1.0 uCi/gm DE I-131 to 0.35 μCi/gm DE I-131
    - Noble gas and alkali RCS activity was based on 1% failed fuel fraction (pre-EPU RCS activity was based on 1% failed fuel, alkalis were not considered)
- Evaluated pre-accident and concurrent iodine spikes for the EPU (pre-accident and concurrent iodine spikes were not modeled in the pre-EPU analysis):
    - Pre-accident spike iodine RCS concentration of 21 μCi/gm DE I-131 shown in Table 2.9.2-9
    - Concurrent iodine spike shown in Table 2.9.2-12 based on iodine appearance rate with increased letdown flow rate in accordance with NSAL 00-0004 (Reference 5)
- included dose contribution from secondary side release (pre-EPU analysis neglected secondary side dose contribution)
- assumed a primary-to-secondary leakage of 150 gpd to the unaffected SG and accident induced leakage of 1 gpm to the affected SG in accordance with Improved Technical Specifications Sections 3.4.12 and 5.6.2.10.b.2 (pre-EPU analysis assumed 1 gpm to the affected SG only)

- revised steam release rate from SGs, transfer pathways (no credit taken for condenser following reactor trip), and duration (pre-EPU assumed release through the condenser (with a DF = 10,000) from unaffected SG)

- revised offsite atmospheric dispersion factors shown in Table 2.9.2-5 to reflect use of five years of recent meteorological data evaluated using PAVAN .

Results

The results of the MSLB dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Description of Analysis and Evaluation

LRA Radiological Consequences

The accident considered is the instantaneous seizure of a reactor coolant pump rotor (i.e., a Locked Rotor Accident), which causes a rapid reduction in the flow through the affected RCS loop, and initiation of reactor trip. The flow imbalance creates localized temperature and pressure changes in the core. These differences are predicted to lead to localized boiling and fuel damage. The main condenser is unavailable, and the plant is cooled down by releases of steam directly to the environment. A simultaneous LOOP is assumed. A maximum allowable failed fuel fraction of 2.77% is utilized for the Locked Rotor Accident to ensure that the dose was within 90% of the dose acceptance criteria.

Analysis Assumptions and Parameters

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-13. The radioactivity transport model is shown in Figure 2.9.2-2. Fuel rods producing 2.77% of the core power are assumed to experience departure from nucleate boiling (DNB), and are therefore assumed to release their gap activity into the RCS. The radionuclides released from the fuel are assumed to be instantaneously and homogeneously mixed in the RCS and transported to the secondary side via primary-to-secondary leakage. It is assumed that differential pressure/temperature conditions in the steam generators during an LRA do not result in primary to secondary leakage in excess of the limiting conditions of operations (LCO). Therefore, the primary to secondary coolant leakage is assumed to be at the ITS limit of 150 gpd/steam generator. The leakage is assumed to be instantaneously released to the atmosphere, without holdup or plateout in the secondary side, via the ADVs and/or MSSVs.

Steam carryover fractions from the unaffected steam generator is 100% for iodines and alkalis (i.e., no credit is taken for partitioning in the steam generator). Noble gas activity, leaked into the SGs, is assumed to be directly released to the environment without mixing. The steam releases, from the steam generators, continues until the Decay Heat Removal (DHR) System can be used to complete the cooldown. This is conservatively assumed to be 30 days (duration of the accident).

Comparison to Reference 1

The locked rotor analysis for Radiological Consequences was not performed prior to the EPU since there was no postulated failed fuel. At the EPU conditions, the LRA results in failed fuel; therefore, the radiological consequences of the LRA are evaluated. As discussed in Section 2.8.3, Thermal and Hydraulic Design, current operating limits bound the MAP limits for the LRA. However, a maximum allowable failed fuel fraction for the LRA is addressed to provide future flexibility in the event that the operating limits or MAP limits do not preclude failed fuel. The analysis of the LRA Radiological Consequences is consistent with the analytical methods specified in Regulatory Guide 1.183, Appendix G (Reference 2) at increased power. The analysis modeled the following:

- revised core nuclide inventory shown in Table 2.9.2-2, consistent with the EPU
- revised primary and secondary coolant initial water mass and nuclide activity shown in Table 2.9.2-7 and Table 2.9.2-8 based on:
    - RCS iodine concentration limit was lowered from 1.0 uCi/gm DE I-131 to of 0.35 μCi/gm DE I-131 for halogens, and
    - the noble gas and alkali RCS activity based on 1% failed fuel fraction f (pre-EPU RCS activity was based on 1% failed fuel, alkalis were not considered)
- assumed primary-to-secondary leakage of 150 gpd per steam generator in accordance with Improved Technical Specification Section 3.4.12
- assumed radial peaking factor of 1.80 to account for differences in the radial power distribution
- revised steam release from the SGs and duration
- maximum allowable accident induced failed fuel fraction
- revised offsite atmospheric dispersion factors shown in Table 2.9.2-5 to reflect use of five years of recent meteorological data evaluated using PAVAN.

Results

The results of the LRA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Description of Analysis and Evaluation

REA Radiological Consequences

The accident considered is the mechanical failure of a casing of the control rod drive mechanism, which results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected due to the reactivity spike. This failure breaches the reactor pressure vessel head resulting in a LOCA to the containment. A reactor trip occurs, safety injection actuates, and a loss of offsite power (LOOP) occurs concurrently with the reactor trip. As this LOOP renders the main condenser unavailable, the plant is cooled down by releases of steam directly to the environment. The release to the environment is assumed to occur through two separate pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through the steam generators.

The actual dose from a REA is a composite of the two pathways. However, the dose from each pathway is conservatively modeled independently of the other.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-14. The radioactivity transport model is shown in Figure 2.9.2-3 and Figure 2.9.2-4.

The radiological impact associated with the postulated REA at the EPU conditions was evaluated in accordance with Regulatory Guide 1.183, Appendix H guidance. For the primary containment release pathway, 56.6% of the fuel cladding is assumed to fail, releasing the radionuclide inventory in the fuel rod gap. It was further assumed that 10% of the core inventory of radioiodine and 10% noble gas is in the fuel rod gap. A radial peaking factor of 1.80 was applied. In addition, localized heating is assumed to cause 4% of the fuel rods which experience clad failure (i.e., 2.26% of all the fuel rods in the core) to melt, releasing 100% of the noble gases and 25% of the radioiodine contained in the melted fuel to the containment. For the secondary release case, 12.6% of the fuel rods are assumed to fail releasing the radionuclide inventory in the fuel rod gap, and 0.5% of the fuel rods in the core are assumed to melt. For this case 100% of the noble gases and 50% of the radioiodine contained in the melted fuel are released to the secondary. The maximum clad failure fraction and fuel melt fraction for each pathway were calculated to yield 90% of the dose acceptance criteria at the critical receptor for the REA. The maximum clad failure fraction and fuel melt fraction calculated for the REA bounds the expected clad failure fraction in Section 2.8.5.4.6, Spectrum of Rod Ejection Accidents. For the REA, the secondary side release pathway was found to be bounding.

The containment leakage case assumes that radionuclides released from the fuel are instantaneously and homogeneously mixed in the containment free volume. In addition, the containment leaks at the ITS value of 0.25% volume per day for the first 24 hours and 0.125 % volume per day for days 2 through 30.

No credit is taken for containment spray operation or filtration as a radionuclide removal mechanism. However, the natural deposition model in RADTRAD (Powers model at 10% probability) (References 6 and 7) is credited.

Secondary side release via primary-to-secondary side leakage is assumed for the REA. This assumes 150 gpd per steam generator is released in accordance with the ITS limit for primary-to-secondary leakage during normal operation and the Reference 1. The atmospheric release consists of the secondary side coolant halogens, the primary coolant radionuclide inventory of noble gases and halogens, and gap inventory released to the RCS from 12.6% clad failures, and inventories released to RCS from fuel overheat/melt. This source term is assumed to be instantly released to the environment without credit for holdup or plateout on the secondary side.

Comparison to Reference 1

The analysis of the REA Radiological Consequences is consistent with the analytical methods and assumptions presented in Regulatory Guide 1.183, Appendix F. This analysis supersedes the analysis presented in Reference 1 with changes made to reflect the increased power, inclusion a the pre-REA primary and secondary source term, revised steam releases and durations, revised failed fuel and melted fuel fractions, and revised offsite atmospheric dispersion factors. Specific changes between the EPU analysis and the pre-EPU analysis include:

- revised core nuclide inventory shown in Table 2.9.2-2, consistent with the EPU
- Pre-REA source term revised primary and secondary coolant initial water mass and nuclide activity per the EPU shown in Table 2.9.2-7 and Table 2.9.2-8 based on (pre-REA source term not modeled for pre-EPU conditions):
    - o RCS iodine concentration limit was lowered from 1.0 uCi/gm DE I-131 to 0.35 μCi/gm DE I-131 for halogens, and
    - o the noble gas RCS activity based on 1% failed fuel fraction (pre-EPU noble gas RCS activity was based on 1% failed fuel)
- revised steam releases and durations
- revised failed and melted fuel fractions (pre-EPU 14% clad failure assumed, fuel melt not considered)
- revised offsite atmospheric dispersion factors shown in Table 2.9.2-5 to reflect use of five years of recent meteorological data evaluated using PAVAN .

Results

The results of the REA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Description of Analysis and Evaluation

SGTR Radiological Consequences

The accident considered is the double ended rupture of a single steam generator tube resulting in the transfer of reactor coolant to the ruptured steam generator. The primary-to-secondary break flow through the ruptured tube following a SGTR results in radioactive contamination of the secondary system. Following the SGTR, the reactor pressure decreases until a reactor trip occurs on low reactor coolant pressure. This is followed by turbine trip and safety injection actuation.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-15. The radioactivity transport model is shown in Figure 2.9.2-5.

The dose calculation is simplistic and conservative. No credit is taken for decay from time of accident. No credit for retention of iodine in the steam generators is taken. Iodine in the RCS is assumed to be at a concentration consistent with 1% failed fuel and not at the much lower Technical Specification limit. EAB dose assumes break flow for two hours. No credit is taken for the expected isolation time of the affected steam generator in calculating the LPZ dose, which is based on a conservative isolation time of 24 hours. Break flow for the full 24 hours is assumed to be a constant bounding value of 60.38 lbm/s, which is approximately 50% greater than the average break flow up to isolation of the affected steam generator. Primary-to-secondary leakage is assumed to be 1 gpm into the intact SG.

Comparison to Reference 1

This analysis of the SGTR Radiological Consequences does not follow the guidance of Regulatory Guide 1.183, Appendix F, but is consistent with the current licensing basis and the SGTR (pre-EPU) analysis presented as part of the original Alternate Source Term submittal and approved in License Amendment No. 199. Like the EPU analysis, the pre-EPU licensing basis analysis did not assume LOOP or iodine spiking, but assumed a source term based on an RCS primary coolant concentration for 1% failed fuel fraction for noble gases and iodines, a conservative constant break flow of 60.38 lbm/s, and a conservative existing primary-to-secondary leakage of 1 gpm. To assure continued conservatism for EPU conditions the release duration was increased from 8 hours (as used in the pre-EPU analysis) to 24 hours.

Results

The results of the SGTR dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Description of Analysis and Evaluation

LOCA Radiological Consequences

The LOCA accident considered is double-ended rupture of a Reactor Coolant System (RCS) pipe resulting in a loss of reactor coolant to the reactor building at a rate in excess of the capability of the Makeup (MU) System. Activity from the core is released to the containment and then to the environment by containment leakage, leakage from the Emergency Core Cooling System (ECCS) as it recirculates sump solution outside the containment, or Reactor Building Purge.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-16. The radioactivity transport model is shown in Figure 2.9.2-6.

Fission products released to the containment atmosphere, following the postulated LOCA, are mitigated by two processes:

(1) Reactor Building Spray System (BS) (discussed in Section 2.5.3.1, Fission Product Control Systems and Structures) removal

(2) Radioactive decay

The BS is designed to provide containment cooling and fission product removal following the postulated LOCA. The BS consists of two trains. Each train consists of a pump, two spray headers, and associated valves. Each train of BS is independently capable of delivering 1,000 gpm of borated water from the Borated Water Storage Tank (BWST) into the containment atmosphere. The spray pumps are automatically started on Engineered Safety Actuation System (ESAS) on high reactor building pressure signal. Reactor Building Spray is assumed to actuate at 0.0344 hours (124 seconds) into the accident.

Release to the environment via ECCS component leakage is postulated to occur. Recirculation loop circulates contaminated sump water outside of the containment, where system leakage is assumed to provide a path for the release of radionuclides to the environment. The assumed leakage rate of 0.04 gallons per hour is two times the program value, consistent with the guidance provided in Regulatory Guide (RG) 1.183. It is assumed that there is no back-leakage into the BWST during the recirculation phase via the check valves in the lines from the BWST per FSAR Section 6.1.2.1.2.

For the Engineered Safety Feature (ESF) component leakage pathway, it is conservatively assumed that all of the radioiodine and non-iodine particulate released from the fuel is instantaneously moved to the containment sump water, and that noble gases remain in the containment atmosphere. It is further assumed that the radioiodine chemical form in the sump water, at the time of recirculation is 97% elemental iodine and 3% organic, in line with RG 1.183. The total iodine in the leaked fluid is assumed to become airborne and leak to the environment based on an ECCS leak rate from the containment sump of 0.04 gallons per hour (2 x ECCS leakage rate) in accordance with CR-3 procedures. Ten percent of the sump water is assumed to flash to steam in accordance with RG 1.183 guidance. All other radionuclides are assumed to be retained in the recirculating liquid in the liquid phase.

Activity released to containment is assumed to be released to the environment at a constant rate of 0.25% per day per ITS for the first 24 hours and 0.125% per day thereafter crediting iodine removal via the Reactor Building Spray (see Section 2.5.3.1, Fission Product Control Systems and Structures). Fission product removal via the Reactor Building Sprays is ensured by maintaining the sump pH greater than 7 by use of Trisodium Phosphate dodechydrate (TSP-C) during the recirculation mode (after switchover from the BWST to the emergency sump).

No release is assumed to occur as a result of back-leakage via the check valves to the BWST.

RB purge via the mini-purge may be in operation during normal operating conditions to purge carbon dioxide from containment. Therefore, this pathway was considered for the LOCA analysis. Reactor Building purge is assumed to take place for the first 5 seconds of the LOCA at sonic velocity calculated using methodology in Crane manual (Reference 8) via two mini-purge lines prior to closure of the containment isolation valves. All of the primary coolant activity is assumed to be released to the environment via this release pathway. No credit is taken for filtration.

Release of radioactivity from containment due to hydrogen purge is not considered for the EPU conditions since hydrogen purge is not credited for hydrogen concentration mitigation.

All releases are assumed be at ground level release from the Reactor Building.

Comparison to Reference 1

The analysis of the LOCA radiological consequences is consistent with the analytical methods and assumptions presented in Regulatory Guide 1.183, Appendix A. This analysis supersedes the analysis presented in Reference 1 with changes made to reflect the increased power, proposed ITS RCS specific activity limit, release pathways, and revised offsite and Technical Support Center (TSC) atmospheric dispersion factors. Specific changes between the EPU analysis and the pre-EPU analysis include:

- revised core inventory shown in Table 2.9.2-2 consistent with the EPU.
- revised primary initial water mass and nuclide activity shown in Table 2.9.2-7 per the EPU based on:

- o RCS iodine concentration limit was lowered from 1.0 uCi/gm DE I-131 to 0.35 μCi/gm DE I-131 for halogens, and
- o the noble gas and alkali RCS activity based on 1% failed fuel fraction (pre-EPU RCS activity was based on 1% failed fuel, alkalis were not considered)

- included Reactor Building purge for 5 seconds via the mini-purge lines (dose contribution due to Reactor Building purge via the mini-purge lines was not included in pre-EPU analysis)
- hydrogen purge was not considered since the hydrogen purge system is no longer credited during LOCA for hydrogen concentration mitigation (pre-EPU analysis included dose contribution due to hydrogen purge (starting at 14 days into LOCA))
- revised cutoff times for the respective elemental and particulate iodine removal coefficients evaluated at the EPU conditions
- revised offsite and TSC atmospheric dispersion factors shown in Table 2.9.2-5 to reflect use of five years of recent meteorological data evaluated using PAVAN and ARCON96.

Results

The results of the LOCA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Description of Analysis and Evaluation

FHA Radiological Consequences

The FHA assumes the dropping of a spent fuel assembly during refueling, and it is postulated to occur either inside the containment or in the fuel storage (auxiliary) building. The dropped assembly may strike the fuel storage rack, the reactor vessel flange, shallow end of the fuel transfer canal, or another fuel assembly. The FHA inside containment bounds the FHA in the fuel building. Therefore, only the results for the FHA inside containment will be presented.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-17. The radioactivity transport model is shown in Figure 2.9.2-7.

A fuel assembly drop analysis was performed using computer code ANSYS. The fuel assembly drop analysis determined that 43 fuel rods of the 208 fuel rods in the dropped fuel assembly fail. Therefore, the total number of damaged fuel rods is conservatively assumed to be equivalent to one-half (½) assembly (104 fuel rods). The damaged assembly is assumed to have the highest inventory of radionuclides of all the assemblies in the core (i.e., application of a radial peaking factor of 1.80 to fuel assembly inventory). The radionuclide inventory, in the gaps of the damaged fuel rods, is assumed to be instantaneously released to the water. Fission products released from the damaged fuel are decontaminated by passage through the overlaying water inside containment or spent fuel pool depending on their physical and chemical form. A decay time of 72 hours prior to moving irradiated fuel was assumed for both the FHA in the containment and in the spent fuel pool.

No decontamination is assumed for noble gases. An effective water pool decontamination factor of 113 and 167 is assumed for radioiodine inside containment and spent fuel pool, respectively, and an infinite DF (100% retention) is assumed for alkalis using the Burley paper methodology referenced in Regulatory Guide 1.183, Appendix B. The FHA analysis assumes that 100% of the radionuclides, becoming airborne within the containment or fuel storage building, are released to the environment in two hours. No credit is taken for filtration or dilution. In addition, no credit is taken for cleanup via the Control Room Emergency Ventilation System (CREVs). The Control Room Emergency Ventilation System is not required to be available during fuel movement. The non-LOCA gap release fractions in Table 3 of the Regulatory Guide 1.183 were doubled except for Xe-133 and its precursors to account for Linear Heat Generation Rate (LHGR) criterion of 6.3 kW/ft for the peak average rod > 54 GWd/MTU. The gap release fractions used are based on industry experience at Calvert Cliffs (Reference 9) and Ginna submittal (Reference 10) calculated using ANSI/ANS-5.4-1982 methodology (Reference 11).

Comparison to Reference 1

The analysis of the FHA Radiological Consequences is consistent with the analytical methods and assumptions presented in Regulatory Guide 1.183, Appendix B. This analysis supersedes the analysis presented in Reference 1 with changes made to reflect the increased power, revised water depths, revised pool DF, revised gap release fractions, revised number of failed rods for the FHA, and revised offsite atmospheric dispersion factors. Specific changes between the EPU analysis and pre-EPU analysis include:

- revised core inventory shown in Table 2.9.2-2 consistent with the EPU
- revised decontamination factor of 113 based on a water depth of 20 feet above the failed fuel (inside containment) and a decontamination factor 167 based on a water depth of 22 feet above the failed fuel (in the spent fuel pool) using the Burley paper methodology
- assumed ½ fuel assembly fails due to fuel assembly drop
- revised gap fractions Table 3 of Reg. Guide 1.183 (i.e., doubled non-LOCA gap fractions, except for Xe-133 and its precursors, which are tripled to account for the LHGR criterion of 6.3 kW/ft for peak average rod > 54 GWD/MTU being exceeded for EPU.)
- revised duration of release to environment of 2 hrs
- revised offsite atmospheric dispersion factors shown in Table 2.9.2-5 to reflect use of five years of recent meteorological data evaluated using PAVAN.

Results

The results of the FHA dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

Description of Analysis and Evaluation

Letdown Line Rupture Radiological Consequences

The Letdown Line Rupture/Break is assumed to occur in the Auxiliary Building resulting in release directly to the environment without mitigation by the Auxiliary Building ventilation system filtration or holdup. The letdown line break flow is in the Auxiliary Building, at 120°F and 14.7 psia. No flashing is expected to occur at these conditions, however, a 10% flashing fraction is applied per Reg. Guide 1.183, Appendix A, Section 5.5. Although, the LLR is not addressed in the Regulatory Guide 1.183, the LLR accident is included in the FSAR and as such is part of the current licensing basis. Therefore, the LLR is included for completeness.

Analysis Parameters and Assumptions

The major assumptions and parameters used in this analysis are itemized in Table 2.9.2-18. The radioactivity transport model is shown in Figure 2.9.2-8.

The radiological consequences of a postulated Letdown Line Rupture (LLR) outside containment were evaluated. A break in fluid-bearing lines that penetrate the reactor containment may result in the release of radioactivity to the environment. There are lines contained within the Makeup and Purification (MU) System and the Decay Heat Removal (DHR) System that penetrate the containment. No instrument lines penetrate containment. The most severe pipe rupture is the rupture of the letdown line. A rupture of a high energy line outside containment is not considered credible. However, for the purposes of this analysis, a failure of the 2-1/2" letdown line outside containment downstream of the outboard containment isolation valve MUV-49 and upstream of the letdown control valves is considered. The letdown line rupture was modeled using the RELAP5/MOD2 B&W at RCS conditions for the EPU. In the RELAP5 model, the letdown integrated flow was determined based on a letdown line break located between the B Steam Generator outlet piping and the cold leg. In the RELAP5 model, this line represents the letdown pipe from the cold leg to the MUV-49 valve located in the Auxiliary Building. For the radiological analysis, the letdown break flow at RCS conditions will be applied to the location outside containment for conservatism.

The LLR is not addressed in Regulatory Guide 1.183. However, Regulatory Guide 1.183, Appendix E guidance for the Main Steam Line Break was applied to the LLR analysis. The present analysis also incorporates the clarifications provided in NRC Regulatory Issue Summary (RIS 2006-04, namely the inclusion of alkalis (in addition to the halogens and noble gases) in the radiological evaluations of the LLR. Two alternative accident scenarios were postulated, as follows:

- An LLR with a pre-accident iodine spike, where a reactor transient had occurred prior to the postulated accident raising the primary coolant concentration to 21 μCi/gm DE I-131 in Table 2.9.2-9.

- An LLR with an accident-induced concurrent iodine spike of 0.133-hr duration, where the iodine spike corresponds to an increase in the design-basis iodine appearance rate into the primary coolant by a factor of 500 in Table 2.9.2-19.

Comparison to Reference 1

The analysis of the LLR radiological consequences is consistent with the analytical methods and assumptions presented in Reference 1 except for changes made to reflect the increased power, LLR timing, Dose Equivalent I-131, revised letdown line break flow and duration, and revised offsite atmospheric dispersion factors. Specific changes between the EPU analysis and the pre-EPU analysis include:

- revised primary coolant activity shown in Table 2.9.2-7 per the EPU based on:
    - RCS iodine concentration limit was lowered from 1.0 uCi/gm DE I-131 to 0.35 µCi/gm DE I-131 for halogens, and
    - the noble gas and alkali RCS activity based on 1% failed fuel (pre-EPU RCS activity was based on 1% failed fuel, alkalis were not considered)
- modeled pre-accident and concurrent iodine spikes:
    - pre-accident spike iodine RCS concentration of 21 µCi/gm DE I-131 (pre-EPU based on 60 µCi/gm DE I-131) shown in Table 2.9.2-9
    - iodine appearance rate for concurrent iodine spike shown in Table 2.9.2-19 and increased letdown flow rate per NASL-00-0004 letter (Reference 8)
- revised break flow and duration (pre-EPU assumed manual isolation of break, no credit taken for automatic isolation on Emergency Safeguards Actuation System (ESAS) signal)
- revised offsite atmospheric dispersion factors shown in Table 2.9.2-5 to reflect use of five years of recent meteorological data evaluated using PAVAN .

Results

The results of the LLR dose calculations, and the applicable dose acceptance criteria, are presented in Table 2.9.2-1.

2.9.2.3 Conclusions

CR-3 has evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concludes that analyses have adequately accounted for the effects of the proposed EPU. CR-3 further concludes that the plant site and the dose-mitigating ESFs remain acceptable with respect to the Radiological Consequences of postulated DBAs since the calculated total effective dose equivalent (TEDE) at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the control room meet the exposure guideline values specified in 10 CFR Part 50.67 and FSAR Section 1.4.11. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Radiological Consequences of Design Basis Accidents.

2.9.2.4 References

1. CR-3 – Issuance of Amendment Regarding Alternative Source Term and Control Ventilation System, September 2001.

2. US NRC NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," Rev. 0, April 1976.

3. EPA 520/1-88-020, Federal Guidance Report No. 11, Limiting Values of Radionuclides Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, ORNL, September 1988.

4. EPA 402-R-93-081, Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil, ORNL, September 1993.

5. Westinghouse Nuclear Safety Advisory Letter NSAL-00-004, Non-Conservatisms in Iodine Spiking Calculations, March 2000.

6. S. L. Humphries et al., "RADTRAD: a Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, US NRC, April 1998.

7. NUREG/CR-6604, Supplement 2, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal And Dose Estimation", October 2002.

8. "Flow of Fluids Through Valves, Fittings, and Pipe", Crane, 1988.

9. US NRC SER, Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2 – Amendment RE: Implementation of Alternative Radiological Source Term, August 29, 2007 (Accession No. ML072130521).

10. R. E. Ginna Nuclear Power Plant, License Amendment Request: Revision of Technical Specification 3.9.3 to Allow Refueling Operations inside Containment with Both Personnel Hatch Air Lock Doors Open Under Administrative Control, and Obtain Regulatory Review of the Supporting Dose Analysis, December 4, 2008 (Accession No. ML0834560051).

11. American Nuclear Society, ANSI/ANS-5.4-1982, Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel, November 1982.

12. ASME Steam Tables, Sixth Edition

13. Palo Verde Nuclear Generating Station Units 1, 2 and 3, Docket Nos. 50-528, 50-529 and 50-530 -Issuance of Amendments Re: Replacement of Steam Generators and Uprated Power Operations and Associated Administrative Changes (NRC Accession No. ML053130275, 11/16/2005).

14. ANS/ANSI-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors" and ANS/ANSI-18.1, ERRATA, December 1, 2005.

Table 2.9.2-1: Summary of CR-3 EPU TEDE Doses and Acceptance Criteria, rem TEDE

Description		EAB	LPZ	MCR	TSC
MSLB	Pre-accident iodine spike	2.82E-02 (25)	2.39E-02 (25)	9.14E-02 (5)	n/a
	Concurrent iodine spike	1.49E-01 (2.5)	1.00E-01 (2.5)	3.96E-01 (5)	
LRA (with 2.77% clad failure)		1.29 (2.5)	1.27 (2.5)	4.49 (5)	n/a
REA (Primary Containment Rel. Pathway)	Bounded by REA with secondary side release.				
REA (Secondary Side Release, bounding)	Clad failure (CF) only [1]	2.82 (6.3)	1.33 (6.3)	4.49 (5)	n/a
	CF + fuel melt (equal to 4% CF) [2]	2.83 (6.3)	1.34 (6.3)	4.50 (5)	
SGTR [3][4]		0.16 (25)	0.19 (25)	[4]	n/a
LOCA		12.79 (25)	1.79 (25)	3.49 (5)[4]	3.02 (5)
FHA (½ FA fails, no CREVs; 72 hrs decay)	Inside containment, 20 ft water depth [5]	1.012 (6.3)	0.118 (6.3)	4.906 (5)	n/a
	Spent Fuel Pool, 22 ft water depth	0.724 (6.3)	0.084 (6.3)	3.352 (5)	
LLR (LOOP)	Pre-accident iodine spike	0.15 (25)	0.02 (25)	0.36 (5)	n/a
	Concurrent iodine spike	0.06 (2.5)	0.01(2.5)	0.12 (5)	
LLR (no LOOP)	Pre-accident iodine spike	0.15 (25)	0.02 (25)	0.71 (5)	n/a
	Concurrent iodine spike	0.06(2.5)	0.01 (2.5)	0.24 (5)	

Notes:

(1) REA with secondary side release dose is for 14.6% CF.

(2) REA with secondary side release dose is for 12.6% CF + 0.5% fuel overheat/melt.

(3) Licensing Basis SGTR results.

(4) A radiological evaluation of SGTR, using the methods of Regulatory Guide 1.183, confirms that control room habitability LOCA dose bound SGTR dose.

(5) Bounds FHA in Spent Fuel Pool area.

Table 2.9.2-2: CR-3 Normalized Core Inventory

Nuclide	EPU Activity (Ci/MWth)	Pre-EPU Activity (Ci/MWth) (2)
Kr-83m (1)	4.250E+03	
Kr-85m	9.785E+03	9.181E+03
Kr-85	3.899E+02	1.960E+02
Kr-87	1.961E+04	1.678E+04
Kr-88	2.771E+04	2.269E+04
Xe-133m (1)	1.708E+03	
Xe-133	5.624E+04	5.372E+04
Xe-135m (1)	1.101E+04	
Xe-135	2.045E+04	1.008E+04
Xe-138 (1)	4.997E+04	
I-131	2.745E+04	2.540E+04
I-132	3.913E+04	3.743E+04
I-133	5.638E+04	5.370E+04
I-134	6.282E+04	5.893E+04
I-135	5.241E+04	5.063E+04
Rb-86	8.909E+01	1.496E+01
Rb-88 (1)	2.798E+04	
Cs-134	8.751E+03	3.425E+03
Cs-136	2.416E+03	1.042E+03
Cs-137	4.203E+03	1.915E+03
Sb-127	3.371E+03	2.208E+03
Sb-129	9.474E+03	7.820E+03
Te-127m	4.534E+02	2.823E+02
Te-127	3.348E+03	2.132E+03
Te-129m	1.394E+03	1.935E+03
Te-129	9.329E+03	7.341E+03
Te-131m	4.094E+03	3.707E+03
Te-132	3.840E+04	3.690E+04
Sr-89	3.496E+04	2.844E+04
Sr-90	3.118E+03	1.535E+03
Sr-91	4.488E+04	3.656E+04
Sr-92	4.640E+04	3.805E+04

Table 2.9.2-2: CR-3 Normalized Core Inventory (cont'd)

Nuclide	EPU Activity (Ci/MWth)	Pre-EPU Activity (Ci/MWth) (2)
Ba-139	5.248E+04	4.976E+04
Ba-140	5.046E+04	4.924E+04
Mo-99	5.056E+04	4.830E+04
Tc-99m	4.425E+04	4.169E+04
Ru-103	4.659E+04	3.598E+04
Ru-105	3.513E+04	2.340E+04
Ru-106	2.156E+04	8.175E+03
Rh-105	3.208E+04	1.621E+04
Ce-141	4.739E+04	4.476E+04
Ce-143	4.706E+04	4.352E+04
Ce-144	3.701E+04	2.697E+04
Pu-238	1.851E+02	2.902E+01
Pu-239	1.203E+01	6.545E+00
Pu-240	2.055E+01	8.254E+00
Pu-241	4.963E+03	1.390E+03
Np-239	6.758E+05	5.120E+05
Y-90	3.250E+03	1.647E+03
Y-91	4.266E+04	3.465E+04
Y-92	4.653E+04	3.819E+04
Y-93	5.069E+04	4.320E+04
Zr-95	4.970E+04	4.377E+04
Zr-97	4.818E+04	4.562E+04
Nb-95	4.967E+04	4.138E+04
La-140	5.069E+04	5.032E+04
La-141	4.805E+04	4.615E+04
La-142	4.739E+04	4.449E+04
Pr-143	4.689E+04	4.273E+04
Nd-147	1.874E+04	1.911E+04
Am-241	5.800E+00	9.181E-01
Cm-242	1.999E+03	3.514E+02
Cm-244	2.866E+02	2.056E+01

Notes:

(1) Radionuclides not in RADTRAD (References 6 and 7) default library. Additional radionuclides included in LOCA and FHA analyses updated RATRAD library.

(2) Based on RADTRAD default PWR nuclide inventory.

Table 2.9.2-3: CR-3 Escape Rate Coefficients for Fission Product Release

Element	Escape Rate Coefficient, sec^{-1}	
	Pre-EPU [(1)]	Extended Power Uprate [(2)]
Xe	$1.0x10^{-7}$	$6.5x10^{-8}$
Kr	$1.0x10^{-7}$	$6.5x10^{-8}$
I	$2.0x10^{-8}$	$1.3x10^{-8}$
Br	$2.0x10^{-8}$	$1.3x10^{-8}$
Cs	$2.0x10^{-8}$	$1.3x10^{-8}$
Rb	$2.0x10^{-8}$	$1.3x10^{-8}$
Mo	$4.0x10^{-9}$	$2.0x10^{-9}$
Te	$5.0x10^{-9}$	$1.0x10^{-9}$
Sr	$2.0x10^{-10}$	$1.0x10^{-11}$
Ba	$2.0x10^{-10}$	$1.0x10^{-11}$
Zr	$1.0x10^{-11}$	$1.6x10^{-12}$
Ce, and other rare earths	$1.0x10^{-11}$	$1.6x10^{-12}$

Notes:

(1) FSAR Table 11-1, Escape Rate Coefficients for Fission Product Release

(2) Taken from NUREG-0017, Rev. 0, Table 2-11.

Table 2.9.2-4: CR-3 Primary Coolant Activity for LOCA

Nuclide	EPU Concentration (μCi/gm) (1, 2)	Pre-EPU Concentration (μCi/gm) (3)
Kr-83m	4.289E-01	4.58E-01
Kr-85m	1.915E+00	2.26E+00
Kr-85	1.000E+01	1.95E+01
Kr-87	1.092E+00	1.17E+00
Kr-88	3.437E+00	3.62E+00
Xe-131m		2.52E+00
Xe-133m	3.885E+00	4.22E+00
Xe-133	2.672E+02	4.00E+02
Xe-135m	5.875E-01	4.28E-01
Xe-135	1.113E+01	1.13E+01
Xe-138	5.178E-01	6.92E-01
I-131	2.801E-01	8.20E-01
I-132	7.789E-02	3.15E-01
I-133	3.516E-01	9.82E-01
I-134	3.775E-02	1.09E-01
I-135	1.690E-01	4.00E-01
Rb-86	3.648E-02	
Rb-88	3.489E+00	3.63E+00
Cs-134	3.989E+00	4.06E+02
Cs-136	9.475E-01	5.98E+00
Cs-137	1.922E+00	1.20E+02
Cs-138		1.02E+00
Sb-127	4.978E-05	
Sb-129	3.483E-05	
Te-127m	4.881E-03	
Te-127	1.639E-02	
Te-129m	1.480E-02	
Te-129	1.581E-02	
Te-131m	2.889E-02	
Te-132	3.451E-01	
Sr-89	5.843E-03	5.87E-02
Sr-90	3.377E-04	6.18E-03
Sr-91	1.808E-03	3.30E-02
Sr-92	7.263E-04	1.08E-02

Table 2.9.2-4: CR-3 Primary Coolant Activity for LOCA (cont'd)

Nuclide	EPU Concentration (μCi/gm) (1, 2)	Pre-EPU Concentration (μCi/gm) (3)
Ba-137m		1.12E+02
Ba-139	4.499E-04	1.15E-01
Ba-140	5.195E-03	9.27E-02
Mo-99	4.052E+00	7.81E+00
Tc-99m	2.580E+00	
Ru-103	7.940E-04	
Ru-105	1.320E-04	
Ru-106	3.730E-04	4.65E-01
Rh-105	4.237E-04	
Ce-141	8.080E-04	
Ce-143	5.487E-04	
Ce-144	6.399E-04	3.65E-03
Pu-238	3.207E-06	
Pu-239	2.091E-07	
Pu-240	3.562E-07	
Pu-241	8.601E-05	
Np-239	9.123E-03	
Y-90	5.073E-04	7.34E-03
Y-91	1.964E-02	3.09E-01
Y-92	8.989E-04	
Y-93	5.438E-04	
Zr-95	8.525E-04	
Zr-97	4.284E-04	
Nb-95	8.605E-04	
La-140	2.109E-03	
La-141	1.639E-04	
La-142	7.216E-05	
Pr-143	8.003E-04	
Nd-147	3.063E-04	
Am-241	1.009E-07	
Cm-242	3.450E-05	
Cm-244	4.968E-06	

Notes:

(1) RCS Activity (i) = RCS Concentration (i) (μCi/gm) x Mass RCS (2.326E+08 gm) x 1.0E-06 (Ci/μCi)

(2) At EPU conditions, the RCS primary coolant concentration for iodines is 0.35 μCi/gm Dose Equivalent I-131. Other radionuclides are assumed to be at 1% failed fuel RCS primary coolant concentrations.

(3) At pre-EPU conditions, the RCS primary coolant concentrations for iodines is 1.00 μCi/gm Dose Equivalent I-131. The other radionuclides are assumed to be at 1% failed fuel RCS primary coolant concentrations.

Table 2.9.2-5: CR-3 Atmospheric Dispersion Factors, Breathing Rates & Occupancy

EPU Atmospheric Dispersion Factors (sec/m^3)				
Time Period	**EAB**	**LPZ**	**Control Room**	**TSC**
0-2 hrs	1.65E-4	1.92E-05	9.00E-04	3.50E-04
2-8 hrs	N/A	8.63E-06		2.34E-04 (1)
8-24 hrs		5.10E-06	5.31E-04	9.43E-05
24-96 hrs		2.21E-06	3.38E-04	7.48E-05
96-720 hrs		6.65E-07	1.49E-04	5.15E-05
Breathing Rates				
Time Period	**EAB**	**LPZ**	**Control Room**	**TSC**
0-2 hrs	3.50E-04	3.5E-04	3.50E-04	3.50E-04
2-8 hrs	N/A	3.5E-04	3.50E-04	3.50E-04
8-24 hrs		1.80E-04	3.50E-04	3.50E-04
24-720 hrs		2.30E-04	3.50E-04	3.50E-04
Occupancy Factors				
Time Period	**EAB**	**LPZ**	**Control Room**	**TSC**
0-24 hrs	1.0 (0-2 hrs)	1.0	1.0	1.0
24-96 hrs	N/A	1.0	0.6	0.6
96-720 hrs		1.0	0.4	0.4
Pre- EPU Atmospheric Dispersion Factors (sec/m^3)				
Time Period	**EAB**	**LPZ**	**Control Room**	**TSC**
0-2 hrs	1.60E-4	1.40E-05	9.00E-04	9.30E-04
2-8 hrs	N/A			
8-24 hrs		1.50E-06	5.31E-04	5.49E-04
24-96 hrs		7.70E-07	3.38E-04	3.57E-04
96-720 hrs		4.50E-07	1.49E-04	1.54E-04
Pre-EPU Breathing Rates				
Time Period	**EAB**	**LPZ**	**Control Room**	**TSC**
0-2 hrs	3.50E-04	3.47E-04	3.47E-04	3.47E-04
2-8 hrs	N/A	3.47E-04	3.47E-04	3.47E-04
8-24 hrs		1.75E-04	1.75E-04	3.47E-04
24-720 hrs		2.32E-04	2.32E-04	3.47E-04
Pre-EPU Occupancy Factors				
Time Period	**EAB**	**LPZ**	**Control Room**	**TSC**
0-24 hrs	1.0 (0-2 hrs)	1.0	1.0	1.0
24-96 hrs	N/A	1.0	0.6	0.6
96-720 hrs		1.0	0.4	0.4

Note:

(1) TSC – 0 to 8 hr value, assumed to apply 2-8 hrs.

Table 2.9.2-6: CR-3 Control Room Ventilation Parameters

Parameter	Value	Reference
Volume of Control Complex	364,900 ft^3	
Free Volume of MCR proper	88,000 ft^3	

Control Room Ventilation Operation

Condition	Inleakage			Exhaust		
	Flow Rate	Iodine Removal Efficiency		Flow Rate	Iodine Removal Efficiency	
Control Room						
Pre-Isolation	5700 cfm	Aerosol	0%	5700 cfm	Aerosol	0%
		Elemental	0%		Elemental	0%
		Organic	0%		Organic	0%
Post-Isolation	1000 cfm	Aerosol	0%	1000 cfm	Aerosol	0%
		Elemental	0%		Elemental	0%
		Organic	0%		Organic	0%

Condition	Recirculation		
	Flow Rate	Iodine Removal Efficiency	
Pre-isolation	37,800 cfm	Aerosol	0%
		Elemental	0%
		Organic	0%
Post-isolation	37,800 cfm	Aerosol	95% (1)
		Elemental	90%
		Organic	90%

Control Room Emergency Ventilation Isolation and Operation Times

Accident	Normal Ventilation Isolation Time	CREVs Actuation Time
LOCA, CREA	0 secs	0.5 hrs
Letdown Line Rupture	0.1 hrs	0.6 hrs
Main Steam Line Break	0.1 hrs	0.6 hrs
Locked Rotor	LOOP (0.0 hrs)	LOOP + 30 min
FHA	n/a	n/a

Note:

(1) For the MSLB, LLR, and SGTR accidents, the recirculation filter/removal efficiency is set to 90% for all iodine species. This was done to accommodate concurrent iodine spike, where each iodine isotope was treated as a separate group. RADTRAD (References 6 and 7) does not provide the option to specify filter efficiencies for separate groups.

Table 2.9.2-7: CR-3 Source Term for RCS Unspiked Iodine, Alkali, & Noble Gases Releases – Non-LOCA

EPU Values		
Radionuclide	**RCS Concentration (µCi/gm)**	**Total RCS Activity (Ci) (1)**
Noble Gases & Halogens at 1% Failed fuel Fraction		
Kr-85m	1.915E+00	4.454E+02
Kr-85	1.000E+01	2.326E+03
Kr-87	1.092E+00	2.540E+02
Kr-88	3.437E+00	7.994E+02
Xe-133	2.672E+02	6.215E+04
Xe-135	1.113E+01	2.589E+03
I-131	3.751E+00	8.725E+02
I-132	9.929E-01	2.309E+02
I-133	4.482E+00	1.043E+03
I-134	4.812E-01	1.119E+02
I-135	2.154E+00	5.010+02
Halogens at 0.35 µCi/gm DE I-131 (2)		
I-131	2.801E-01	6.516E+01
I-132	7.789E-02	1.812E+01
I-133	3.516E-01	8.179E+01
I-134	3.775E-02	8.781E+00
I-135	1.690E-01	3.931E+01
Alkalis at Design Basis (1% Failed Fuel Fraction)		
Rb-86	3.648E-02	8.485E+00
Cs-134	3.989E+00	9.278E+02
Cs-136	9.475E-01	2.204E+02
Cs-137	1.922E+00	4.471E+02

Table 2.9.2-7: CR-3 Source Term for RCS Unspiked Iodine, Alkali, & Noble Gases Releases – Non-LOCA (cont'd)

Pre- EPU Values (3)		
Radionuclide	**RCS Concentration (µCi/gm)**	**Total RCS Activity (Ci)**
Noble Gases at 1% Failed fuel Fraction		
Kr-85m	2.26E+00	5.26E+02
Kr-85	1.95E+01	4.54E+03
Kr-87	1.17E+00	2.72E+02
Kr-88	3.62E+00	8.42E+02
Xe-133	4.00E+02	9.30E+04
Xe-135	1.13E+01	2.63E+03
Halogens at 1% Failed fuel Fraction		
I-131	5.18E+00	1.21E+03
I-132	1.99E+00	4.63E+02
I-133	6.20E+00	1.44E+03
I-134	6.90E-01	1.61E+02
I-135	2.53E+00	5.89E+02
Halogens at 1.0 µCi/gm DE I-131		
I-131	8.20E-01	1.92E+02
I-132	3.15E-01	7.35E+01
I-133	9.82E-01	2.29E+02
I-134	1.09E-01	2.54E+01
I-135	4.00E-01	9.36E+01
Alkalis at Design Basis (1% Failed Fuel Fraction)		
Rb-86	---	---
Cs-134	4.06E+02	9.44E+04
Cs-136	5.98E+02	1.39E+05
Cs-137	1.20E+02	2.79E+04

Notes:

(1) RCS Activity (i) = RCS Concentration (i) (µCi/gm) x Mass RCS (2.326E+08 gm) x 1.0E-06 (Ci/µCi)

(2) At EPU conditions, the RCS primary coolant concentrations for iodines is 0.35 µCi/gm Dose Equivalent I-131, except for the licensing basis SGTR analysis which assumes 1% failed fuel for iodines. Other radionuclides are assumed to be at 1% failed fuel RCS primary coolant concentrations.

(3) At the pre-EPU conditions, the RCS primary coolant concentrations for halogens are at 1.00 µCi/gm Dose Equivalent I-131, except for the licensing basis SGTR analysis which assumes 1% failed fuel for iodines. The noble gases are assumed to be at 1% failed fuel RCS primary coolant concentrations (alkalis were not considered) with the exception of the Letdown Line Rupture. The LLR that assumes the RCS primary coolant concentrations are at Technical Specification limit of 1.00 µCi/gm Dose Equivalent I-131 for both halogens and noble gases (equivalent to 0.16% failed fuel fraction).

Table 2.9.2-8: CR-3 EPU Source Term for Secondary-Side Iodine and Alkali Releases – Non-LOCA

Radionuclide	SG Secondary Side Concentration (1, 2, 3) (μCi/gm)	Total Activity in both SGs (Ci)
Halogens at 4.5E-04 μCi/gm DE I-131		
I-131	3.603E-04	1.898E-02
I-132	1.003E-04	5.284E-03
I-133	4.517E-04	2.380E-02
I-134	4.826E-05	2.542E-03
I-135	2.170E-04	1.143E-02
Alkalis at Design Basis (1% Failed Fuel Fraction)		
Rb-86	5.912E-07	3.114E-05
Cs-134	6.465E-05	3.406E-03
Cs-136	1.536E-05	8.092E-04
Cs-137	3.115E-05	1.641E-03

Notes:

(1) Secondary Activity (i) = Secondary Concentration (i) (μCi/gm) x Mass Secondary Side (2.634E+07 gm) x 1.0E-06 (Ci/μCi)

(2) At EPU conditions, the secondary coolant concentrations for halogens are at 4.5E-04 μCi/gm Dose Equivalent I-131. The other radionuclides are assumed to be at 1% failed fuel RCS primary coolant concentrations.

(3) For the pre-EPU accident analyses, the secondary side coolant activity was neglected.

Table 2.9.2-9: CR-3 Source Term for RCS Pre-accident Iodine Spike at 20.65 µCi/gm – Non-LOCA

EPU Values		
Radionuclide	**RCS Concentration (µCi/gm)**	**Total RCS Activity (Ci)**
Halogens at 20.65 µCi/gm DE I-131 (1)		
I-131	1.653E+01	3.845E+03
I-132	4.596E+00	1.069E+03
I-133	2.075E+01	4.825E+03
I-134	2.227E+00	5.181E+02
I-135	9.970E+00	2.319E+03
Pre-EPU		
Halogens at 60 µCi/gm DE I-131 (2)		
I-131	4.92E+01	1.15E+04
I-132	1.89E+01	4.42E+03
I-133	5.89E+01	1.38E+04
I-134	6.54E+00	1.53E+03
I-135	2.40E+01	5.61E+03

Notes:

(1) At the EPU conditions, RCS total DEI-131 concentration is 21 µCi/gm, corresponding to the summation of entries in Table 2.9.2-7 and Table 2.9.2-9.

(2) At the pre-EPU conditions, RCS total DEI-131 concentration is 60 µCi/gm for the Letdown Line Rupture only.

Table 2.9.2-10: CR-3 EPU Activity Available for Release for Non-LOCA Accidents

Nuclide	Activity Released FHA (1 Failed Fuel Assembly) (Ci) (2)	Activity Released REA Due to 1% Clad Failure (Ci) (2)	Activity Released REA Due to 1% Fuel Melt (Ci) (2, 3)	Activity Released LRA Due to 1% Clad Failure (Ci) (2)
Kr-83m (1)	1.308E+04	2.31E+04	2.08E+05	2.31E+04
Kr-85m	3.011E+04	5.33E+04	4.80E+05	5.33E+04
Kr-85	2.400E+03	2.12E+03	1.91E+04	4.25E+03
Kr-87	6.036E+04	1.07E+05	9.61E+05	1.07E+05
Kr-88	8.527E+04	1.51E+05	1.36E+06	1.51E+05
Xe-133m (1)	7.885E+03	9.30E+03	8.37E+04	1.90E+05
Xe-133	2.596E+05	3.06E+05	2.76E+06	1.67E+03
Xe-135m(1)	3.387E+04	6.00E+04	5.40E+05	1.40E+04
Xe-135	6.294E+04	1.11E+05	1.00E+06	4.60E+05
Xe-138 (1)	1.538E+05	2.72E+05	2.45E+06	6.00E+04
I-131	1.352E+05	1.50E+05	5.98E+05	1.11E+05
I-132	1.204E+05	2.13E+05	8.52E+05	2.70E+05
I-133	2.602E+05	3.07E+05	1.23E+06	2.72E+05
I-134	1.933E+05	3.42E+05	1.37E+06	2.31E+04
I-135	1.613E+05	2.85E+05	1.14E+06	4.28E+04
Rb-86	6.580E+02	0.00E+00	0.00E+00	5.27E+04
Rb-88 (1)	2.066E+05	0.00E+00	0.00E+00	7.12E-03
Cs-134	6.463E+04	0.00E+00	0.00E+00	8.87E+03
Cs-136	1.784E+04	0.00E+00	0.00E+00	2.39E+05
Cs-137	3.105E+04	0.00E+00	0.00E+00	2.13E+05

Notes:

(1) Radionuclides not in RADTRAD (References 6 and 7) default library. Additional radionuclides included in LOCA and FHA analyses updated RATRAD library.

(2) At pre-EPU conditions, the activity released corresponds to the following:

- FHA: pre-EPU normalized activity (Ci) (Table 2.9.2-2) x 2619 MWth x 1/177
- CREA: pre-EPU normalized activity (Ci) (Table 2.9.2-2) x 2619 MWth x 1%

(Note: For pre-EPU conditions, fuel melt was not evaluated.)

- LRA: This accident was not evaluated since there was no failed fuel postulated at pre-EPU conditions for the LRA.

(3) Exclusive of gap activity for REA limiting scenario (secondary release).

Table 2.9.2-11: CR-3 EPU Double-Ended Rupture of One Steam Line Radiological Consequences

<table>
<tr><th>Item</th><th colspan="2">Parameter</th><th colspan="2">Value</th><th>Basis</th></tr>
<tr><td colspan="6">Source Term</td></tr>
<tr><td>1</td><td colspan="2">Core thermal power (1)</td><td colspan="2">3026.1 MWth
(2619 MWth)</td><td></td></tr>
<tr><td rowspan="3">2</td><td rowspan="3">Radionuclide concentration limits</td><td>RCS iodines</td><td colspan="2">0.35 µCi/gm DE I-131
(1% failed fuel for pre-EPU)</td><td rowspan="2"></td></tr>
<tr><td>RCS alkalis</td><td colspan="2">Based on 1% failed fuel
(neglected for pre-EPU)</td></tr>
<tr><td>SG secondary-side coolant</td><td colspan="2">4.5E-04 µCi/gm
DE I-131
(neglected for pre-EPU)</td><td>ITS Section 3.7.16</td></tr>
<tr><td rowspan="3">3</td><td rowspan="3">Primary coolant activity (Ci)</td><td>Noble gases</td><td colspan="2" rowspan="3">See Table 2.9.2-7</td><td rowspan="3"></td></tr>
<tr><td>Iodines</td></tr>
<tr><td>Alkalis</td></tr>
<tr><td rowspan="3">4</td><td rowspan="3">Secondary side coolant activity (Ci)</td><td>Noble gases</td><td colspan="2" rowspan="3">See Table 2.9.2-8
(neglected for pre-EPU)</td><td rowspan="3"></td></tr>
<tr><td>Iodines</td></tr>
<tr><td>Alkalis</td></tr>
<tr><td rowspan="2">5</td><td colspan="2" rowspan="2">Iodine spiking (no iodine spiking assumed in pre-EPU analysis)</td><td>Pre-Accident</td><td>21 µCi/gm
DE I-131
(see Table 2.9.2-7 & Table 2.9.2-9, additive)</td><td>60 x 0.35 µCi/gm DE I-131, which is greater than ITS Section 3.4.15 limit</td></tr>
<tr><td>Concurrent</td><td>500-fold increase for 8 hr duration (see Table 2.9.2-12)</td><td>Reg. Guide 1.183, Appendix E</td></tr>
<tr><td>6</td><td colspan="2">Concurrent Iodine Spike RCS Activity & Fractional Release</td><td colspan="2">See Table 2.9.2-12
(no iodine spike assumed in pre-EPU analysis)</td><td></td></tr>
<tr><td rowspan="3">7</td><td colspan="2" rowspan="3">Chemical form of iodine species in primary coolant</td><td>Aerosol</td><td>0</td><td rowspan="3">Reg. Guide 1.183, Appendix E, Section 4</td></tr>
<tr><td>Elemental</td><td>97%</td></tr>
<tr><td>Organic</td><td>3%</td></tr>
<tr><td rowspan="4">8</td><td rowspan="4">RCS (including pressurizer)</td><td rowspan="2">Total Volume</td><td colspan="2">8218.9 ft^3 (cold)</td><td>0.714 g/cc x 11511 ft^3</td></tr>
<tr><td colspan="2">11511 ft^3 (hot)</td><td></td></tr>
<tr><td>RCS Mass</td><td colspan="2">5.127E+05 lbm
(2.326E+08 gm)</td><td></td></tr>
<tr><td>Density at 2170 psia & 582 °F</td><td colspan="2">0.714 gm/cc
(0.709 gm/cc pre-EPU)</td><td>ASME Steam Tables (Reference 11)</td></tr>
<tr><td rowspan="3">9</td><td rowspan="3">OTSG *</td><td>Secondary Water Volume</td><td colspan="2">1219.8 ft^3/SG</td><td>Calculated value:
58068 lbm / 47.606 lbm/ft^3</td></tr>
<tr><td>Secondary Water Mass</td><td colspan="2">58068 lbm/SG
(2.634E+07 gm/SG)</td><td></td></tr>
<tr><td>Secondary water density at 964 psia & 525 °F</td><td colspan="2">47.606 lbm/ft^3
(50 lbm/ft^3 pre-EPU)</td><td>ASME Steam Tables (Reference 11)</td></tr>
</table>

Table 2.9.2-11: CR-3 EPU Double-Ended Rupture of One Steam Line Radiological Consequences (cont'd)

Item	Parameter	Value		Basis
10	Dose conversion factors	FGR11&12.inp		RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)
Variables Concurrent Spike Iodine Appearance Rate				
11a	Nominal RCS letdown flow rate	75 gpm		
11b	Conservative RCS letdown flow rate used in current analysis for iodine appearance rate	113 gpm		Based on guidance in NSAL-00-004 (Nominal + 50%) (Reference 5)
11c	RCS letdown flow rate mass basis (at 120 °F, 14.7 psia, 61.71 lbm/ft^3)	5.593E+04 lbm/hr		Calculated value: 113 (gpm) * 61.71 (lbm/ft^3) * 60 (min/hr) / 7.481 (gal/ft^3)
11d	RCS letdown flow for boron recovery (yearly avg.)	500 lbm/hr		ANSI/ANS-18.1-1999 (Reference 14)
11e	Demineralizer purification efficiency for the removal of halogens (Br and I)	0.99		FSAR Section 11.1
11f	Iodine decay constants (for use in the computation of the appearance rates during iodine spiking)	RADTRAD Table 1.4.3.2-2		RADTRAD (References 6 and 7)
Primary-to-Secondary Side Releases				
12	Unaffected SG (nominal)	0-720 hrs	150 gpd (1.393E-02 cfm)	ITS Section 3.4.12
	Affected SG (accident initiated)	0-720 hrs	1 gpm (1.337E-01 cfm)	ITS Section 5.6.2.10.b.2 & RIS 2007-020 (Reference 17)
		Time	**Pre-EPU**	
		0-2 hrs	802 lbm	
		2-8 hrs	2791.44 lbm	
		8-24 hrs	7443.85 lbm	
		24-96 hrs	28379.68 lbm	
13	Partition coefficient for halogens in SG	1		Secondary-side liquid and steam concentrations assumed to be the same (on mass basis)
14	Steam carryover	100%		Conservative assumption
15	Halogen and alkali depletion due to plateout on internal surfaces and steam lines	None credited		Conservative assumption
16	Release pathway (unaffected SG)	Via ADVs or SRVs (via condenser with DF = 10,000 pre-EPU)		No credit for release via condenser before reactor trip.

Table 2.9.2-11: CR-3 EPU Double-Ended Rupture of One Steam Line Radiological Consequences (cont'd)

Item	Parameter	Value		Basis
17	Primary-to-secondary leakage duration	720 hrs (96 hrs pre-EPU)		Conservative assumption
18	Primary-to-secondary leakage assumed to mix simultaneously with secondary coolant in unaffected SG	All		Conservative assumption
Secondary Side Releases *				
19	Unaffected SG	0-720 hrs	1.0E+12 cfm	Conservative assumption. No holdup assumed in secondary side.
	Affected SG	0-720 hrs	1.0E+12 cfm	
	Pre-EPU (density = 50 lbm/ft^3)	0-2 hrs	200866 lbm	
* Secondary side releases are neglected in pre-EPU analyses. Duration of release is assumed to be 8 hrs.				
Control Room Parameters				
20	Volume of Control Complex	364,900 ft^3		
21	Free Air Volume, MCR proper	88,000 ft^3		
22	Pre-Isolation normal ventilation	Flow rate	5700 cfm	Automatic isolation on reactor trip on high flux at 6.87 secs and 1 minute damper closure time.
		Filtration	None	
		Duration	0.0-0.1 hrs	
23	Post-Isolation normal ventilation	Flow rate	1000 cfm	
		Filtration	None	
		Duration	0.1-720 hrs	
24	Recirculation Flow	Flow rate	37,800 cfm	Technical Specifications B 3.7.12 and FSAR Table 9-16 (Manual actuation at 30 minutes after isolation)
		Filtration	95% (aerosol) 90% (organic & elemental)	
		Duration	0.6-720 hrs	
25	Exposure interval	720 hrs		Reg. Guide 1.183, Section 4.2.6
		Dose Locations		
26	Atmospheric Dispersion Factors	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
27	Breathing Rates & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
28	Occupancy Factors & Exposure Intervals	EAB	See Table 2.9.2-5	Reg. Guide 1.183, Sects. 4.1.3 and 4.2.6
		LPZ		
		MCR/TSC		
		Dose Acceptance Criteria – Pre-Accident Iodine Spike		
29	Exclusion Area Boundary (EAB)	25 rem TEDE		Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)	25 rem TEDE		
	Main Control Room (MCR)	5 rem TEDE		
		Dose Acceptance Criteria – Concurrent Iodine Spike		
30	Exclusion Area Boundary (EAB)	2.5 rem TEDE		Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)	2.5 rem TEDE		
	Main Control Room (MCR)	5 rem TEDE		

Table 2.9.2-11: CR-3 EPU Double-Ended Rupture of One Steam Line Radiological Consequences (cont'd)

Item	Parameter	Value	Basis
Dose Results – Pre-Accident Iodine Spike (21 µCi/gm)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.0282 rem
Low Population Zone (LPZ) Dose		TEDE	0.0239 rem
Main Control Room (MCR) Dose		TEDE	0.0914 rem
Dose Results - Concurrent Iodine Spike (initially at 0.35 µCi/gm)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.149 rem
Low Population Zone (LPZ) Dose		TEDE	0.100 rem
Main Control Room (MCR) Dose		TEDE	0.396 rem
Dose Results – Pre-EPU (no iodine spiking assumed)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.0047 rem
Low Population Zone (LPZ) Dose		TEDE	0.0031 rem
Main Control Room (MCR) Dose		TEDE	---

Note:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

Table 2.9.2-12: Iodine Release into RCS from Concurrent Iodine Spike – MSLB

Radionuclide	Total Adjusted 0 to 8 hr Release (Ci) (1)	Fractional Release (2)	
		0 – 2 hrs	2 – 8 hrs
I-131	2.982E+04	2.473E-01	7.527E-01
I-132	9.808E+04	1.010E-01	8.990E-01
I-133	5.328E+04	2.261E-01	7.739E-01
I-134	5.672E+05	2.773E-02	9.723E-01
I-135	5.126E+04	1.819E-01	8.181E-01

Notes:

(1) Reverse-decay adjustment, for use with RADTRAD (References 6 and 7) only.

(2) Fractional release input into release fraction file for concurrent iodine spike case for MSLB for 500 spiking factor.

Table 2.9.2-13: CR-3 EPU Locked Rotor Accident Radiological Consequences

<table>
<tr><th>Item</th><th colspan="2">Description</th><th>Value</th><th>Basis</th></tr>
<tr><td colspan="5">General</td></tr>
<tr><td>1</td><td colspan="2">Dose conversion factors</td><td>FGR11&12.inp</td><td>RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)</td></tr>
<tr><td colspan="5">Source Term</td></tr>
<tr><td>2</td><td colspan="2">Core thermal power (1)</td><td>3026.1 MWt</td><td></td></tr>
<tr><td>3</td><td colspan="2">Peak assembly radial peaking factor</td><td>1.8</td><td>COLR</td></tr>
<tr><td>4</td><td colspan="2">Maximum Allowable Failed Fuel Fraction</td><td>2.77%</td><td></td></tr>
<tr><td>5</td><td colspan="2">Normalized Core inventory (Ci/MWth)</td><td>See Table 2.9.2-2</td><td></td></tr>
<tr><td rowspan="7">6</td><td rowspan="7">Fuel rod activity gap fractions for LRA (non-fuel overheat/melt) (2)</td><td>I-131</td><td>0.16</td><td rowspan="7"></td></tr>
<tr><td>I-133</td><td>0.15</td></tr>
<tr><td>Kr-85</td><td>0.20</td></tr>
<tr><td>Xe-133 & Xe-133m</td><td>0.15</td></tr>
<tr><td>Other Noble Gases</td><td>0.10</td></tr>
<tr><td>Other Halogens</td><td>0.10</td></tr>
<tr><td>Alkalis</td><td>0.24</td></tr>
<tr><td rowspan="3">7</td><td rowspan="3">RCS radionuclide concentration limits</td><td>Iodines</td><td>0.35 μCi/gm
DE I-131</td><td rowspan="3">Reg. Guide 1.183, Section 3.1</td></tr>
<tr><td>Noble gases</td><td>Based on 1% failed fuel</td></tr>
<tr><td>Others</td><td>Not controlled</td></tr>
<tr><td rowspan="3">8</td><td rowspan="3">Primary coolant concentration (μCi/gm)</td><td>Noble gases</td><td rowspan="3">See Table 2.9.2-7</td><td rowspan="3"></td></tr>
<tr><td>Iodines</td></tr>
<tr><td>Alkalis</td></tr>
<tr><td rowspan="2">9</td><td colspan="2" rowspan="2">Primary to secondary leak rate, assumed cold (lasting until shutdown cooling is initiated)</td><td>150 gpd/SG</td><td>Technical Specifications, Section 3.4.12</td></tr>
<tr><td>0.5%/day</td><td></td></tr>
<tr><td>10</td><td colspan="2">Duration of release to secondary coolant</td><td>30 days</td><td>Conservative assumptions</td></tr>
<tr><td rowspan="2">11</td><td rowspan="2">ITS limits for SG secondary-side conc.</td><td>Halogens</td><td>4.5E-04 μCi/gm
DE I-131</td><td rowspan="2">Technical Specifications, Section 3.4.16</td></tr>
<tr><td>Others</td><td>Not controlled</td></tr>
<tr><td rowspan="3">12</td><td rowspan="3">Secondary side coolant concentration (μCi/gm)</td><td>Noble gases</td><td rowspan="3">See Table 2.9.2-8</td><td rowspan="3"></td></tr>
<tr><td>Iodines</td></tr>
<tr><td>Alkalis</td></tr>
<tr><td colspan="5">Note: The Locked Rotor Accident was not considered at pre-EPU conditions since there was no fuel damage postulated.</td></tr>
</table>

Table 2.9.2-13: CR-3 EPU Locked Rotor Accident Radiological Consequences (cont'd)

Item	Description		Value		Basis
13	Fraction of gap activity released to RCS (instantaneous release, uniform mixing)		100%		Reg. Guide 1.183, Appendix H, Section 1 (The release from overheated fuel is conservatively assumed to be same as that from melted fuel.)
14	Chemical composition of halogens released to atmosphere	Elemental	97%		Reg. Guide 1.183, Appendix H, Section 5
		Organic	3%		
Coolant Volumes & Masses					
15	RCS (including pressurizer)	Total Volume	8218.9 ft^3 (cold)		0.714 g/cc x 11511 ft^3
			11511 ft^3 (hot)		
		RCS Mass	5.127E+05 lbm (2.326E+08 gm)		
		Density at 2170 psia & 582°F	0.714 gm/cc		ASME Steam Tables (Reference 11)
16	OTSG	Secondary Water Volume	1219.8 ft^3/SG		58068 lbm / 47.606 lbm/ft^3
		Secondary Water Mass	58068 lbm/SG (2.634E+07 gm/SG)		
		Secondary water density at 964 psia & 525°F	47.606 lbm/ft^3		ASME Steam Tables (Reference 11)
System Response and Plant Cooldown					
17	Overall steaming rate for plant cooldown (for accident duration)		1E12 cfm		Conservative value, equivalent to no holdup in secondary side.
			1E10 %/day		
18	Time for RCS coolant temperature to reach 212°F		30 days		Conservative assumption
Control Room Variables					
19	Exposure interval		30 days		Reg. Guide 1.183, Section 4.2.6
20	Volume of Control Complex		364,900 ft^3		
21	Free Air Volume of MCR proper		88,000 ft^3		
22	Manual actuation of control room recirculation filters		0.5 hours		
23	Recirculation filter efficiency		Aerosol	95%	
			Elemental	90%	
			Organic	90%	
24	Recirculation flow rate		37,800 cfm		
25	Unfiltered inleakage		1000 cfm		
Dose Locations					
26	Atmospheric Dispersion Factors	EAB	See Table 2.9.2-5		
		LPZ			
		MCR/TSC			

Table 2.9.2-13: CR-3 EPU Locked Rotor Accident Radiological Consequences (cont'd)

Item	Description		Value	Basis
27	Breathing Rates & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
28	Occupancy Factors & Exposure Intervals	EAB	See Table 2.9.2-5	Reg. Guide 1.183, Sections 4.1.3 and 4.2.6
		LPZ		
		MCR/TSC		
Dose Acceptance Criteria				
29	Exclusion Area Boundary (EAB)		2.5 rem TEDE	Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)		2.5 rem TEDE	
	Main Control Room (MCR)		5 rem TEDE	
Dose Results for the Locked Rotor Accident (2.77% Clad Failure)				
Exclusion Area Boundary (EAB) Dose			TEDE	1.29 rem
Low Population Zone (LPZ) Dose			TEDE	1.27 rem
Main Control Room (MCR) Dose			TEDE	4.49 rem

Notes:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

(2) Gap fractions from Table 3 of Regulatory Guide 1.183 were doubled except for Xe-133 and its precursors which were tripled. This was done since the linear heat generation rate exceeded 6.3 kW/ft for fuel exceeding 54 GWD/MTU per footnote 11 of Table 3 based on industry precedence at Calvert Cliffs (Reference 9) and Ginna (Reference 10).

Table 2.9.2-14: CR-3 EPU Control Rod Ejection Accident Radiological Consequences

Item	Parameter	Value		Basis
General				
1	Core thermal power (1)	3026.1 MWth (2619 MWth pre-EPU)		
2	RCS radionuclide activity	See Table 2.9.2-7 (noble gases & iodines only)		
3	Secondary side radionuclide activity	See Table 2.9.2-8 (iodines only)		
4	Normalized Core Nuclide Inventory Values - Containment & ECCS Leakage Cases (Ci/MWth)	See Table 2.9.2-2		
5	Fraction of core inventory in gap (non-fuel melt/overheat) (2)	Halogens	10%	Reg. Guide 1.183, Appendix H
		Noble Gases	10%	
6	Radial Peaking Factor	1.80		COLR
7	Gap activity available for release	See Table 2.9.2-10		
8	RCS (including pressurizer) at 2170 psia & 582 °F	Total Volume	8218.9 ft^3 (cold)	0.714 gm/cc/1 gm/cc x 11511 ft^3
			11511 ft^3 (hot)	
		RCS Mass	5.127E+05 lbm	
			2.326E+08 gm	
		Density	0.714 gm/cc (0.709 gm/cc pre-EPU)	ASME Steam Tables (Reference 11)
9	Dose Conversion Factors	fgr11&12.inp		RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)
Primary Containment Leakage Pathway				
10	Clad Failure (CF) only, % core	60.5% (14% CF pre-EPU)		Selected to yield 90% of dose acceptance criteria at critical receptor.
11	CF + fuel melt (equal to 4% CF), % core	56.6% CF /2.26% FM (fuel melt not evaluated at pre-EPU conditions)		
12	Containment volume	2.0E+06 ft^3		
13	Fraction of gap activity released to containment (instantaneous release, uniform mixing)	100%		Reg. Guide 1.183, Appendix H, Section 1 (The release from overheated fuel is conservatively assumed to be same as that from melted fuel.)
14	Fraction of overheated/ melted-fuel inventory released to containment	Halogens and Alkalis	25% (n/a pre-EPU)	
		Noble gases	100% (n/A pre-EPU)	
15	Containment spray removal credit of iodine	None		

Table 2.9.2-14: CR-3 EPU Control Rod Ejection Accident Radiological Consequences (cont'd)

<table>
<tr><th>Item</th><th colspan="2">Parameter</th><th colspan="2">Value</th><th>Basis</th></tr>
<tr><td>16</td><td colspan="2">Aerosol natural deposition</td><td colspan="2">10% Powers' model</td><td>Reg. Guide 1.183, RADTRAD, Table 2.2.2.1 (Reference 6 and 7)</td></tr>
<tr><td rowspan="2">17</td><td colspan="2" rowspan="2">Containment Leak Rate</td><td>0-24 hrs</td><td>0.25%/day</td><td>ITS Section 3.6.1</td></tr>
<tr><td>24 – 720 hrs</td><td>0.125%/day</td><td>Reg. Guide 1.183, Appendix A, Section 3.7.1</td></tr>
<tr><td rowspan="3">18</td><td colspan="2" rowspan="3">Iodine Chemical Fraction – Containment Air</td><td>Aerosol</td><td>0.95</td><td rowspan="3">Reg. Guide 1.183, Section 3.5</td></tr>
<tr><td>Elemental</td><td>0.0485</td></tr>
<tr><td>Organic</td><td>0.0015</td></tr>
<tr><td colspan="6">Secondary Side Release</td></tr>
<tr><td>19</td><td colspan="2">Clad Failure (CF) only, % core</td><td colspan="2">14.6% CF
(14% CF for pre-EPU)</td><td rowspan="2">(Selected to yield 90% of the dose acceptance criteria at the critical receptor.)</td></tr>
<tr><td>20</td><td colspan="2">CF + fuel melt (FM) (equal to 4% CF), % core</td><td colspan="2">12.6% CF / 0.5% FM
(FM neglected for pre-EPU)</td></tr>
<tr><td rowspan="3">21</td><td rowspan="3">OTSG
(not modeled pre-EPU)</td><td colspan="2">Secondary Water Volume</td><td>1219.8 ft^3/SG</td><td>58068 lbm / 47.606 lbm/ft^3</td></tr>
<tr><td colspan="2">Secondary Water Mass</td><td>58068 lbm/SG (2.634E+07 gm/SG)</td><td></td></tr>
<tr><td colspan="2">Secondary water density at 964 psia & 525°F</td><td>47.606 lbm/ft^3</td><td>ASME Steam Tables (Reference 11)</td></tr>
<tr><td>22</td><td colspan="2">Primary to secondary leak rate, assumed cold (lasting until shutdown cooling is initiated)</td><td colspan="2">0.5%/day</td><td>Based on 150 gpd/SG per ITS Section 3.4.12</td></tr>
</table>

Table 2.9.2-14: CR-3 EPU Control Rod Ejection Accident Radiological Consequences (cont'd)

Item	Parameter	Value		Basis
23	Holdup or removal in SG	none (instantaneous release)		Conservative assumption
24	Release path	Direct release to environment via ADV (assume loss of condenser)		Conservative assumption
25	Duration of release to secondary coolant	72 hours		
26	Iodine Chemical Fraction – water	Aerosol	0.00	Reg. Guide 1.183, Appendix H
		Elemental	0.97	
		Organic	0.03	
Control Room Parameters				
27	Volume of Control Complex	364,900 ft^3		
28	Free Air Volume of MCR proper	88,000 ft^3		
29	Post-Isolation normal ventilation	Flow rate	1000 cfm	Automatic isolation on LOOP at t = 0
		Filtration	None	
		Duration	0.0-720 hrs	
30	Recirculation Flow	Flow rate	37,800 cfm	Technical Specifications B 3.7.12 and FSAR Table 9-16 (Manual Actuation at 30 minutes after isolation)
		Filtration	95% (aerosol) 90% (organic & elemental)	
		Duration	0.5-720 hrs	
31	Exposure interval	720 hrs		Reg. Guide 1.183, Section 4.2.6
Dose Locations				
32	Atmospheric Dispersion Factors	EAB	See Table 2.9.2-5	
		LPZ		
		MCR		
33	Breathing Rates & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
		MCR		
34	Occupancy Factors & Exposure Intervals	EAB	See Table 2.9.2-5	Reg. Guide 1.183, Sects. 4.1.3 and 4.2.6
		LPZ		
		MCR		
35	Dose conversion factors		FGR11&12.inp	RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)
Dose Acceptance Criteria [(3)]				
36	Exclusion Area Boundary (EAB)		6.3 rem TEDE	Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)		6.3 rem TEDE	
	Main Control Room (MCR)		5 rem TEDE	

Table 2.9.2-14: CR-3 EPU Control Rod Ejection Accident Radiological Consequences (cont'd)

Item	Parameter			Value	Basis
Dose Results for Control Rod Ejection Accident					
Fuel Damage	**TEDE Dose (rem)**			**Clad Failure Percentage**	**Full-Core Percentage of Fuel-Rod Overheat/Melt**
	EAB	**LPZ**	**MCR**		
PRIMARY CONTAINMENT LEAKAGE PATHWAY					
Clad failure (CF) only	5.67	1.50	3.79	60.5 %	N/A
CF + fuel melt (equal to 4% CF)	5.66	1.49	3.77	56.6%	2.26%
SECONDARY SIDE RELEASE (4)					
Clad failure (CF) only	2.82	1.33	4.49	14.6 %	N/A
CF + fuel melt (equal to 4% CF)	2.83	1.34	4.50	12.6 %	0.5%
Dose Results for Control Rod Ejection Accident – pre-EPU					
Fuel Damage	**TEDE Dose (rem)**			**Clad Failure Percentage**	**Full-Core Percentage of Fuel-Rod Overheat/Melt**
	EAB	**LPZ**	**MCR**		
PRIMARY CONTAINMENT LEAKAGE PATHWAY					
Clad failure (CF) only	1.03	0.754	0.754	14.0 %	N/A
SECONDARY SIDE RELEASE					
Clad failure (CF) only	2.10	0.819	3.49	14.0 %	N/A

Notes:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

(2) Gap fractions used in accordance with Reg. Guide 1.183, Appendix H.

(3) The dose acceptance criteria for MCR are for each release pathway independently (Primary Containment Leakage or Secondary Side Leakage).

(4) The secondary-side release scenario is bounding for the CREA, leading to a lower acceptable clad failure and fuel overheat/melt.

Table 2.9.2-15: CR-3 EPU Steam Generator Tube Rupture Radiological Consequences

Item	Parameter	Value	Basis
Source Term			
1	Core thermal power [1]	3026.1 MWth (2619 MWth pre-EPU)	
2	Break flow	60.38 lbm/s constant	Conservative Licensing Basis Value
3	Existing Primary to Secondary Leakrate	1 gpm	Conservative Licensing Basis Value
4	Secondary side coolant activity	Neglected	Licensing Basis Assumption
5	Duration of Steaming (MSSVs)	120 seconds	Conservative Assumption
6	Removal mechanisms – Decay, plateout, etc	No decay or iodine retention assumed in SGs	Conservative Assumption
7	Time to isolate affected SG used in dose calculation	24 hours	Conservative Assumption
8	Condenser DF for iodines	10,000	NUREG-1228
9	Reactor Coolant Concentration (μCi/gm)	Table 2.9.2-7 Noble Gases & Iodines only	Based on 1% failed fuel
10	Dose conversion factors	fgr11&12	FGR11 & FGR12 (References 3 and 4) and Reg Guide1.183
11	Release Duration and Exposure Time	24 hours	Conservative assumption

Note:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

Table 2.9.2-15: CR-3 EPU Steam Generator Tube Rupture Radiological Consequences (cont'd)

Item	Parameter		Value	Basis
Dose Locations				
12	Atmospheric Dispersion Factors	EAB	See Table 2.9.2-5	
		LPZ		
13	Breathing Rates & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
14	Occupancy Factors & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
Dose Acceptance Criteria (no iodine spiking or LOOP assumed)				
15	Exclusion Area Boundary (EAB)		2.5 rem TEDE	
	Low Population Zone (LPZ)		2.5 rem TEDE	
	Main Control Room (MCR)		5 rem TEDE	

Dose Results for the SGTR – EPU (no iodine spiking or LOOP assumed)		
Exclusion Area Boundary (EAB) Dose	TEDE	0.16 rem
Low Population Zone (LPZ) Dose	TEDE	0.19 rem
Main Control Room (MCR) Dose	TEDE	Not calculated
Dose Results for the SGTR – pre-EPU (no iodine spiking or LOOP assumed)		
Exclusion Area Boundary (EAB) Dose	TEDE	0.14 rem
Low Population Zone (LPZ) Dose	TEDE	0.046 rem
Main Control Room (MCR) Dose	TEDE	Not calculated

Table 2.9.2-16: CR-3 EPU LOCA Radiological Consequences

<table>
<tr><th>Item</th><th colspan="2">Parameter</th><th colspan="2">Value</th><th>Basis</th></tr>
<tr><td colspan="6">Source Term</td></tr>
<tr><td>1</td><td colspan="2">Core thermal power (1)</td><td colspan="2">3026.1 MWth
(2619 MWth pre-EPU)</td><td></td></tr>
<tr><td>2</td><td colspan="2">Normalized Core Nuclide Inventory Values - Containment & ECCS Leakage Cases (Ci/MWth)</td><td colspan="2">See Table 2.9.2-2</td><td></td></tr>
<tr><td>3</td><td colspan="2">RCS Activity (Ci)</td><td colspan="2">See Table 2.9.2-4</td><td></td></tr>
<tr><td rowspan="9">4</td><td rowspan="9">LOCA Gap Fractions / Release Timing</td><td>Radionuclide Group</td><td>Gap release (30 sec – 0.5 hrs)</td><td>Early In-Vessel (0.5 -1.8 hrs)</td><td rowspan="9">Reg. Guide 1.183, Tables 2 and 4</td></tr>
<tr><td>Noble Gases (Xe, Kr)</td><td>0.05</td><td>0.95</td></tr>
<tr><td>Halogens (I, Br)</td><td>0.05</td><td>0.35</td></tr>
<tr><td>Alkali Metals (Cs, Rb)</td><td>0.05</td><td>0.25</td></tr>
<tr><td>Tellurium Metals (Te, Sb)</td><td>0</td><td>0.05</td></tr>
<tr><td>Ba, Sr</td><td>0</td><td>0.02</td></tr>
<tr><td>Noble Metals (Ru, Rh, Mo, Co)</td><td>0</td><td>0.0025</td></tr>
<tr><td>Cerium Group (La, Zr, Nd, Nb, Pr, Y, Cm, Am)</td><td>0</td><td>0.0005</td></tr>
<tr><td>Lanthanides</td><td>0</td><td>0.0002</td></tr>
<tr><td rowspan="2">5</td><td rowspan="2">Primary Coolant Concentrations</td><td>Iodine</td><td colspan="2">0.35 µCi/gm
DE I-131</td><td rowspan="2">Greater than proposed ITS Sect ion 3.4.15</td></tr>
<tr><td>Noble gases & other</td><td colspan="2">1% failed fuel</td></tr>
<tr><td>6</td><td colspan="2">Primary coolant mass</td><td colspan="2">5.127E+05 lbm
(2.326E+08 gm)</td><td></td></tr>
<tr><td rowspan="2">7</td><td rowspan="2">Dose conversion factors</td><td>Containment & ECCS Leakage</td><td colspan="2">FGR11&12.inp modified to include radionuclides of interest</td><td rowspan="2">RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)</td></tr>
<tr><td>RB Purge</td><td colspan="2">fgr11&12.inp</td></tr>
<tr><td colspan="6">Containment Leakage & Reactor Building Purge</td></tr>
<tr><td>8</td><td colspan="2">Containment Total Volume</td><td colspan="2">2.000E+06 ft^3</td><td></td></tr>
<tr><td>9</td><td colspan="2">Containment Sprayed Volume</td><td colspan="2">1.304E+06 ft^3</td><td></td></tr>
<tr><td>10</td><td colspan="2">Containment Unsprayed Volume</td><td colspan="2">6.96E+05 ft^3</td><td></td></tr>
</table>

Table 2.9.2-16: CR-3 EPU LOCA Radiological Consequences (cont'd)

Item	Parameter		Value		Basis
11	Mixing Rate		Sprayed to Unsprayed Region	2562%/day	2 turnovers of unsprayed volume per hour per Reg. Guide 1.183, Appendix A, Section 3.3
			Unsprayed to Sprayed Region	4800%/day	
12	Containment Leak Rate		0 – 24 hrs	0.25%/day	ITS Section 3.6.1
			24 – 720 hrs	0.125%/day	Reg. Guide 1.183, Appendix A, Section 3.7
13	Spray Removal Initiation Time		124 seconds (0.0344 hrs)		
14	Spray Removal Coefficient		**Duration**	**Lambda**	Containment Spray System is safety related
	Elemental	Before DF = 200	0.0344-3.5 hr	19.81 hr^{-1}	
		After DF = 200	3.5-720 hrs	0 hr^{-1}	
	Aerosol	Before DF = 50	0.0344-5.0 hr	1.98 hr^{-1}	
		After DF = 50	5.0-720 hrs	0.198 hr^{-1}	
	Organic		0-720 hrs	0.0	
15	Iodine Chemical Fraction – Containment Air		Aerosol	0.95	Reg. Guide 1.183, Section 3.5
			Elemental	0.0485	
			Organic	0.0015	
ECCS Leakage					
16	Containment Sump Volume		45,370 ft^3		
17	ECCS Leakage Rate		0.017%/day (2 x 0.02 gpm)		2 x Surveillance procedure limit
18	Iodine Chemical Fraction – Sump Water		Aerosol	0.00	Reg. Guide 1.183, Appendix A, Section 5.6
			Elemental	0.97	
			Organic	0.03	
19	Iodine Flashing Fraction		10%		Reg. Guide 1.183, Appendix A, Section 5.5
20	Non-halogens		Retained in water		Reg. Guide 1.183, Appendix A, Section 5.4
Reactor Building Purge (not modeled for pre-EPU)					
21	Duration of RB purge		5 seconds (0.0014 hrs)		ITS Section B 3.6.1
22	RB Purge Flow Rate (Sprayed + Unsprayed)		0-0014 hrs	31914.1 cfm	
			0.0014-24 hrs	3.47 cfm	
			24-720 hrs	1.74 cfm	

Table 2.9.2-16: CR-3 EPU LOCA Radiological Consequences (cont'd)

Item	Parameter	Value		Basis
Control Room Parameters				
23	Volume of Control Complex	364,900 ft^3		
24	Free Air Volume of MCR proper	88,000 ft^3		
25	Pre-Isolation normal ventilation	Flow rate	0 cfm	Automatic MCR isolation at onset of LOCA.
		Filtration	None	
		Duration	0.0-720 hrs	
26	Post-Isolation normal ventilation	Flow rate	1000 cfm	
		Filtration	None	
		Duration	0.0-720 hrs	
27	Recirculation Flow	Flow rate	37,800 cfm	Technical Specifications B 3.7.12 and FSAR Table 9-16 (Manual Actuation at 30 minutes after isolation)
		Filtration	95% (aerosol) 90% (organic & elemental)	
		Duration	0.5-720 hrs	
28	Exposure interval	720 hrs		Reg. Guide 1.183, Section 4.2.6
Technical Support Center Parameters				
29	Free Air Volume	43,880 ft3		
30	Unfiltered flow	0-1.33 hrs	362 cfm	
		1.33-720	10 cfm	
31	Filtered makeup	0-1.33 hrs	0 cfm	
		1.33-720	625 cfm	
		Filtration	95% (aerosol) 90% (organic & elemental)	
32	Exhaust	0-1.33 hrs	362 cfm	
		1.33-720	635 cfm	
33	Recirculation Flow	Flow rate	2300 cfm	Technical Specifications B 3.7.12 and FSAR Table 9-16 Assumed to take 20 minutes to manually align to emergency mode (after TSC actuation)
		Filtration	95% (aerosol) 90% (organic & elemental)	
		Duration	1.33-720 hrs	
34	Exposure interval	720 hrs		Reg. Guide 1.183, Section 4.2.6

Table 2.9.2-16: CR-3 EPU LOCA Radiological Consequences (cont'd)

Item	Parameter	Value		Basis
Dose Locations				
35	Atmospheric Dispersion Factors	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
36	Breathing Rates & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
37	Occupancy Factors & Exposure Intervals	EAB	See Table 2.9.2-5	Reg. Guide 1.183, Sects. 4.1.3 and 4.2.6
		LPZ		
		MCR/TSC		
Dose Acceptance Criteria				
38	Exclusion Area Boundary (EAB)	25 rem TEDE		Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)	25 rem TEDE		
	Main Control Room (MCR)	5 rem TEDE		
	Technical Support Center (TSC)			
Dose Results				
Exclusion Area Boundary (EAB) Dose		TEDE		12.79 rem
Low Population Zone (LPZ) Dose		TEDE		1.79 rem
Main Control Room (MCR) Dose		TEDE		3.49 rem
Technical Support Center (TSC) Dose		TEDE		3.02 rem
Dose Results – pre-EPU				
Exclusion Area Boundary (EAB) Dose		TEDE		7.59 rem
Low Population Zone (LPZ) Dose		TEDE		1.07 rem
Main Control Room (MCR) Dose		TEDE		2.30 rem
Technical Support Center (TSC) Dose		TEDE		4.71 rem

Note:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

Table 2.9.2-17: CR-3 EPU Fuel Handling Accident Radiological Consequences

Item	Parameter		Value		Basis
Limiting FHA inside containment at 135 ft elevation					
1	Core thermal power (1)		3026.1 MWth (2619 MWth pre-EPU)		
2	Normalized Core Nuclide Inventory Values (Ci/MWth)		See Table 2.9.2-2		
3	Number of assemblies in core		177		FSAR Section 3.16
4	Number of fuel rods per assembly		208		FSAR Section 3.16
5	Peak assembly radial peaking factor		1.80		FSAR Section 3.12.4 (bounds radial peaking anticipated during normal operation listed in COLR)
6	Number of Failed Fuel Assemblies		½ fuel assembly		
7	Fuel rod activity gap fractions (2)	I-131	0.16 (0.08 pre-EPU)		
		I-133	0.15 (0.05 pre-EPU)		
		Kr-85	0.20 (10 pre-EPU)		
		Xe-133 & Xe-133m	0.15 (0.05 pre-EPU)		
		Other Noble Gases	0.1 (0.05 pre-EPU)		
		Other Halogens	0.1 (0.05 pre-EPU)		
		Alkalis	0.24 (neglected pre-EPU)		
8	Minimum Decay Prior to FHA		72 hours		Tech. Specs Section 3.3.15
9	Percent of damaged fuel rod gap activity released		100%		Reg. Guide 1.183, Appendix B
Atmospheric Release Resulting from Postulated FHA in Primary Containment in 20 feet Water Depth					
10	Plant configuration		Open containment		
11	Water depth inside containments		20 ft		Reg. Guide 1.183, Appendix B & ITS Section 3.9.6 (Figure 3.62c) Minimum Water Elevation - Primary Shield Elevation – grid width = 156 – 135 - 8.536/12 = 20.3 feet
12	Overall pool decontamination factor (DF)		Iodine	113	DF calculated per US NRC Burley Paper (Accession No. 8402080322) method (Reference 15)
			Noble Gas	1	
			Alkalis	infinite	
13	Composition of airborne halogens above cavity		Elemental	57%	Reg. Guide 1.183, Appendix B
			Organic	43%	
14	Gap activity available for release		Table 2.9.2-10		
15	Release point to atmosphere		RB Stack		
16	RB filtration		None credited		Conservative assumption

Table 2.9.2-17: Radiological Consequences of a Fuel Handling Accident (cont'd)

Item	Parameter	Value		Basis
Atmospheric Release Resulting from Postulated FHA in Fuel Building < 23 feet Water Depth				
17	Minimum Water depth above top of failed fuel in Fuel Building	22 feet		(Reg. Guide 1.183, Appendix B) (ITS, Section 3.7.13.1) (Figure 3.62c) Minimum Water Elevation - Spent Fuel Pool Bottom Elevation – Length of FA – seating surface height - grid width = 156 – 118.333 - 165.895/12 – 1 – 8.536/12 = 22.13 feet
18	Overall pool decontamination factor (DF)	Iodine	167 (100 pre-EPU)	DF calculated per US NRC Burley Paper (Accession No. 8402080322) method (Reference 15)
		Noble Gas	1	
		Alkalis	infinite	
19	Composition of airborne halogens above cavity	See Item #13		
20	Release point to atmosphere	Auxiliary Building		
21	Exhaust filtration	None Credited		
22	Dose conversion factors	FGR11&12.inp modified to include additional radionuclides		RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)
Control Room Parameters				
23	Volume of Control Complex	364,900 ft^3		
24	Free Air Volume of MCR proper	88,000 ft^3		
25	Control room emergency ventilation availability	Not available		ITS Section 3.7.12
26	Normal ventilation	Flow rate	5700 cfm	
		Filtration	None	
		Duration	720 hrs	
27	Exposure interval	720 hrs		Reg. Guide 1.183, Section 4.2.6
Dose Locations				
28	Atmospheric Dispersion Factors	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
29	Breathing Rates & Exposure Intervals	EAB	See Table 2.9.2-5	
		LPZ		
		MCR/TSC		
30	Occupancy Factors & Exposure Intervals	EAB	See Table 2.9.2-5	Reg. Guide 1.183, Sections 4.1.3 and 4.2.6
		LPZ		
		MCR/TSC		

Table 2.9.2-17: Radiological Consequences of a Fuel Handling Accident (cont'd)

Item	Parameter	Value	Basis
Dose Acceptance Criteria			
31	Exclusion Area Boundary (EAB)	6.3 rem TEDE	Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)	6.3 rem TEDE	
	Main Control Room (MCR)	5 rem TEDE	
Dose Results of FHA Inside Open Containment (20 ft water depth, 72 hr decay, ½ Fuel Assembly Fails & No CREVs)			
Exclusion Area Boundary (EAB) Dose		TEDE	1.012 rem
Low Population Zone (LPZ) Dose		TEDE	0.118 rem
Main Control Room (MCR) Dose		TEDE	4.906 rem
Dose Results of FHA Spent Fuel Pool (22 ft water depth, 72 hr decay, ½ Fuel Assembly Fails & No CREVs)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.724 rem
Low Population Zone (LPZ) Dose		TEDE	0.084 rem
Main Control Room (MCR) Dose		TEDE	3.352 rem
Dose Results of FHA Spent Fuel Pool (22 ft water depth, 72 hr decay, 1Fuel Assembly Fails & No CREVs)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.83 rem
Low Population Zone (LPZ) Dose		TEDE	0.073 rem
Main Control Room (MCR) Dose		TEDE	4.43 rem

Notes:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

(2) Gap fractions from Table 3 of Regulatory Guide 1.183 were doubled except for Xe-133 and its precursors which were tripled. This was done since the linear heat generation rate exceeded 6.3 kW/ft for fuel exceeding 54 GWD/MTU per footnote 11 of Table 3 based on industry precedence at Calvert Cliffs (Reference 9) and Ginna (Reference 10)

Table 2.9.2-18: CR-3 EPU Letdown Line Rupture Radiological Consequences

<table>
<tr><th>Item</th><th colspan="2">Parameter</th><th colspan="2">Value</th><th>Basis</th></tr>
<tr><td colspan="6">Source Term</td></tr>
<tr><td>1</td><td colspan="2">Core thermal power (1)</td><td colspan="2">3026.1 MWth
(2619 MWth pre-EPU)</td><td></td></tr>
<tr><td rowspan="2">2</td><td rowspan="2">Radionuclide concentration limits</td><td>RCS iodines</td><td colspan="2">0.35 µCi/gm DE I-131
(1.0 µCi/gm DE I-131 pre-EPU)</td><td rowspan="2">Greater than proposed Technical Specification 3.4.15</td></tr>
<tr><td>RCS alkalis & noble gases</td><td colspan="2">Based on 1% failed fuel
(0.16% failed fuel pre-EPU)</td></tr>
<tr><td>3</td><td colspan="2">Primary coolant activity (Ci)</td><td colspan="2">See Table 2.9.2-7</td><td></td></tr>
<tr><td rowspan="2">4</td><td colspan="2" rowspan="2">Iodine spiking</td><td>Pre-Accident</td><td>21 µCi/gm DE I-131 (see Table 2.9.2-7 & Table 2.9.2-9, additive) (60 µCi/gm DE I-131 pre-EPU)</td><td>60 x 0.35 µCi/gm DE I-131, greater than proposed ITS Section 3.4.15 limit</td></tr>
<tr><td>Concurrent</td><td>500-fold increase for 478 secs duration (500-fold increase for 19.5 minutes pre-EPU) (see Table 2.9.2-19)</td><td>Reg. Guide 1.183, Appendix E</td></tr>
<tr><td>5</td><td colspan="2">Fractional Release & RCS Activity of Iodine for concurrent Spike</td><td colspan="2">See Table 2.9.2-19</td><td rowspan="2"></td></tr>
<tr><td>6</td><td colspan="2">Failed fuel</td><td colspan="2">None</td></tr>
<tr><td rowspan="3">7</td><td colspan="2" rowspan="3">Chemical form of iodine species in primary coolant</td><td>Aerosol</td><td>0</td><td rowspan="3">Reg. Guide 1.183, Appendix E, Section 4</td></tr>
<tr><td>Elemental</td><td>97%</td></tr>
<tr><td>Organic</td><td>3%</td></tr>
<tr><td>8</td><td colspan="2">Primary coolant mass including pressurizer</td><td colspan="2">5.127E+05 lbm
(2.326E+08 gm)</td><td></td></tr>
<tr><td>9</td><td colspan="2">Dose conversion factors</td><td colspan="2">fgr11&12.inp</td><td>RADTRAD (References 6 and 7) & FGR11 & FGR12 (References 3 and 4)</td></tr>
<tr><td colspan="6">Variables Concurrent Spike Iodine Appearance Rate</td></tr>
<tr><td>10</td><td colspan="2">Variables used in the computation of concurrent iodine spike iodine appearance rate</td><td colspan="2">See Table 2.9.2-11
(MSLB, Items 11a – 11f)</td><td></td></tr>
<tr><td colspan="6">System Response</td></tr>
<tr><td>11</td><td colspan="2">Maximum letdown break mass flow rate</td><td colspan="2">116.4 lbm/sec
(6984.6 lbm/min)</td><td>Calculated value: 116.4 lbm/sec x 60 sec/min = 6984.6 lbm/min</td></tr>
<tr><td>12</td><td colspan="2">Reactor trip time following LLR</td><td colspan="2">319 secs</td><td></td></tr>
</table>

Table 2.9.2-18: CR-3 EPU Letdown Line Rupture Radiological Consequences (cont'd)

Item	Parameter		Value		Basis
13	Time to isolate letdown break		478 secs (19.5 minutes pre-EPU assuming manual termination)		
14	Carryover Fractions	Noble gases	1.00		Reg. Guide 1.183, Appendix E, Section 5.4
		Halogens & alkalis	0.10		Reg. Guide 1.183, Appendix A, Section 5.5
Control Room Parameters					
15	Volume of Control Complex		364,900 ft^3		
16	Free Air Volume of MCR proper		88,000 ft^3		
No LOOP					
17	Pre-Isolation normal ventilation		Flow rate	5700 cfm	
			Filtration	None	
			Duration	720 hrs	
LOOP					
18	Pre-Isolation normal ventilation		Flow rate	5700 cfm	Automatic isolation on LOOP at 319 secs + 60 secs for damper closure
			Filtration	None	
			Duration	0-0.1 hrs (not modeled pre-EPU)	
19	Post-Isolation normal ventilation		Flow rate	1000 cfm	
			Filtration	None	
			Duration	0.1-720 hrs	
20	Recirculation Flow		Flow rate	37,800 cfm	Technical Specifications B 3.7.12 and FSAR Table 9-16 (Manual Actuation at 30 minutes after isolation)
			Filtration	95% (aerosol) 90% (organic & elemental) (90% for all pre-EPU)	
			Duration	0.6-720 hrs (0.5-720 hrs pre-EPU)	
21	Exposure interval		720 hrs		Reg. Guide 1.183, Section 4.2.6
Dose Locations					
22	Atmospheric Dispersion Factors		EAB	See Table 2.9.2-5	
			LPZ		
			MCR/TSC		
23	Breathing Rates & Exposure Intervals		EAB	See Table 2.9.2-5	
			LPZ		
			MCR/TSC		
24	Occupancy Factors & Exposure Intervals		EAB	See Table 2.9.2-5	Reg. Guide 1.183, Sects. 4.1.3 and 4.2.6
			LPZ		
			MCR/TSC		

Table 2.9.2-18: CR-3 EPU Letdown Line Rupture Radiological Consequences (cont'd)

Item	Parameter	Value	Basis
Dose Acceptance Criteria – Pre-Accident Iodine Spike			
25	Exclusion Area Boundary (EAB)	25 rem TEDE	Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)	25 rem TEDE	
	Main Control Room (MCR)	5 rem TEDE	
Dose Acceptance Criteria – Concurrent Iodine Spike			
26	Exclusion Area Boundary (EAB)	2.5 rem TEDE	Reg. Guide 1.183, Table 6 & 10 CFR Part 50.67
	Low Population Zone (LPZ)	2.5 rem TEDE	
	Main Control Room (MCR)	5 rem TEDE	
Dose Results - Pre-Accident Iodine Spike (21 µCi/gm)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.15 rem
Low Population Zone (LPZ) Dose		TEDE	0.02 rem
Main Control Room (MCR) Dose (LOOP)		TEDE	0.36 rem
Main Control Room (MCR) Dose (No LOOP)		TEDE	0.71 rem
Dose Results - Concurrent Iodine Spike (initially at 0.35 µCi/gm)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.06 rem
Low Population Zone (LPZ) Dose		TEDE	0.01 rem
Main Control Room (MCR) Dose (LOOP)		TEDE	0.12 rem
Main Control Room (MCR) Dose (No LOOP)		TEDE	0.24 rem
Pre-EPU Dose Results			
Pre-Accident Iodine Spike (60 µCi/gm)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.614 rem
Low Population Zone (LPZ) Dose		TEDE	0.540 rem
Main Control Room (MCR) Dose (LOOP)		TEDE	0.895 rem
Main Control Room (MCR) Dose (No LOOP)		TEDE	3.24 rem
Concurrent Iodine Spike (initially at 1.0 µCi/gm)			
Exclusion Area Boundary (EAB) Dose		TEDE	0.078 rem
Low Population Zone (LPZ) Dose		TEDE	0.0068 rem
Main Control Room (MCR) Dose (LOOP)		TEDE	0.060 rem
Main Control Room (MCR) Dose (No LOOP)		TEDE	0.339 rem

Note:

(1) Regulatory Guide 1.183 specifies a typical value of 1.02 for instrumentation uncertainty for use in heat balance calculations. Since the implementation and approval of the MUR uprate that installed more accurate secondary calorimetrics (i.e., Leading Edge Flow Meter (LEFM) and Main Steam P/T transmitters, the maximum uncertainty is 0.4%. This corresponds to a rated thermal power of 3026.1 MWt for the EPU analyses.

Table 2.9.2-19: Iodine Release into RCS from Concurrent Iodine Spike Due to Letdown Line Rupture Accident

Radionuclide	Total 0 to 478 sec Release (Ci) (1)	Fractional Release (2)	
		0 - 120 sec	120 - 478 sec
I-131	4.879E+02	2.510E-01	7.490E-01
I-132	5.038E+02	2.473E-01	7.527E-01
I-133	7.754E+02	2.506E-01	7.494E-01
I-134	5.528E+02	2.413E-01	7.587E-01
I-135	5.624E+02	2.497E-01	7.503E-01

Notes:

(1) Reverse-decay adjustment, for use with RADTRAD (References 6 and 7) only.
(2) Fractional release input into release fraction file for concurrent iodine spike case for LLR for 500 spiking factor.

Figure 2.9.2-1: CR-3 EPU Main Steam Line Break Accident – Scenario Diagram

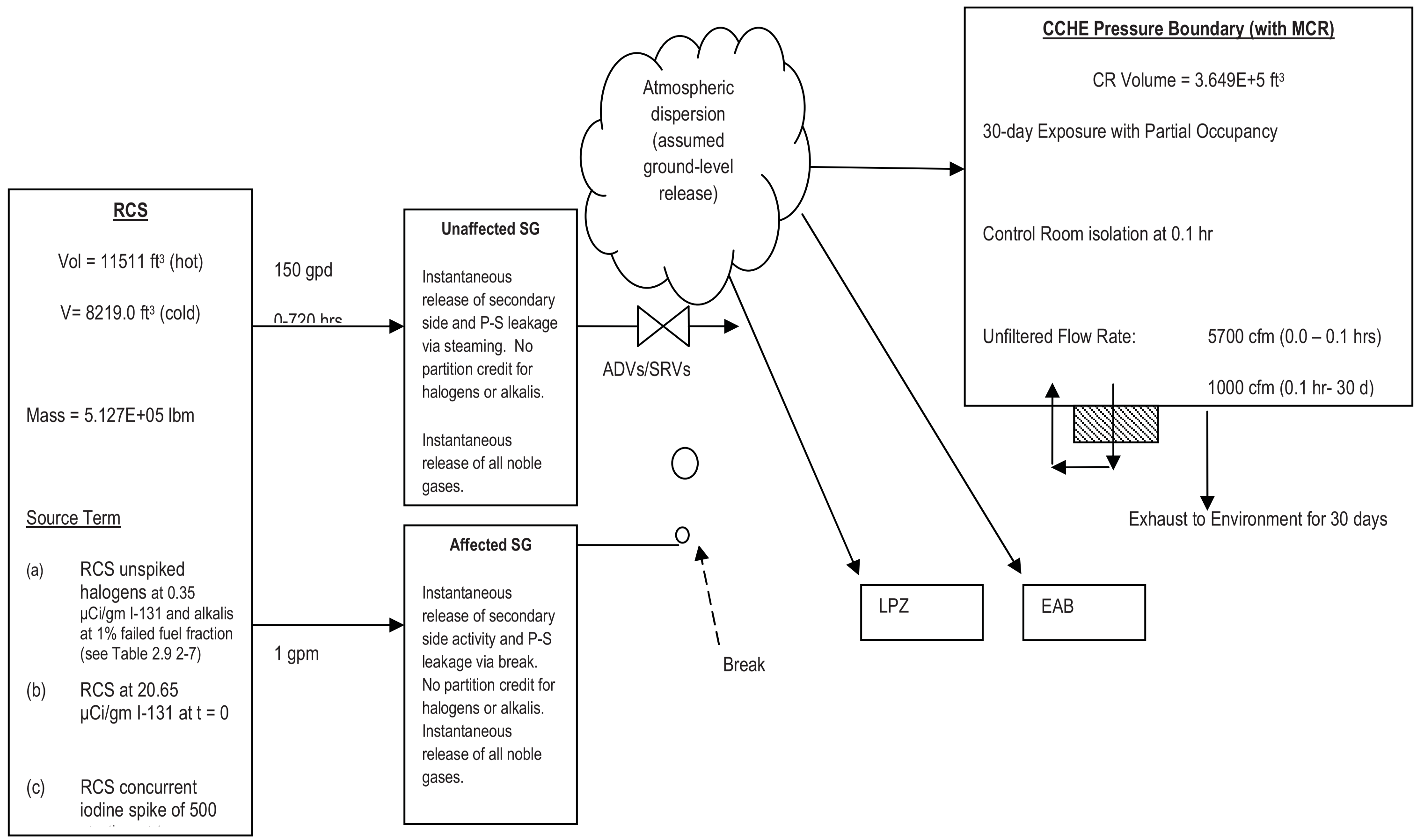


Figure 2.9.2-2 : CR-3 EPU Locked Rotor Accident – Scenario Diagram

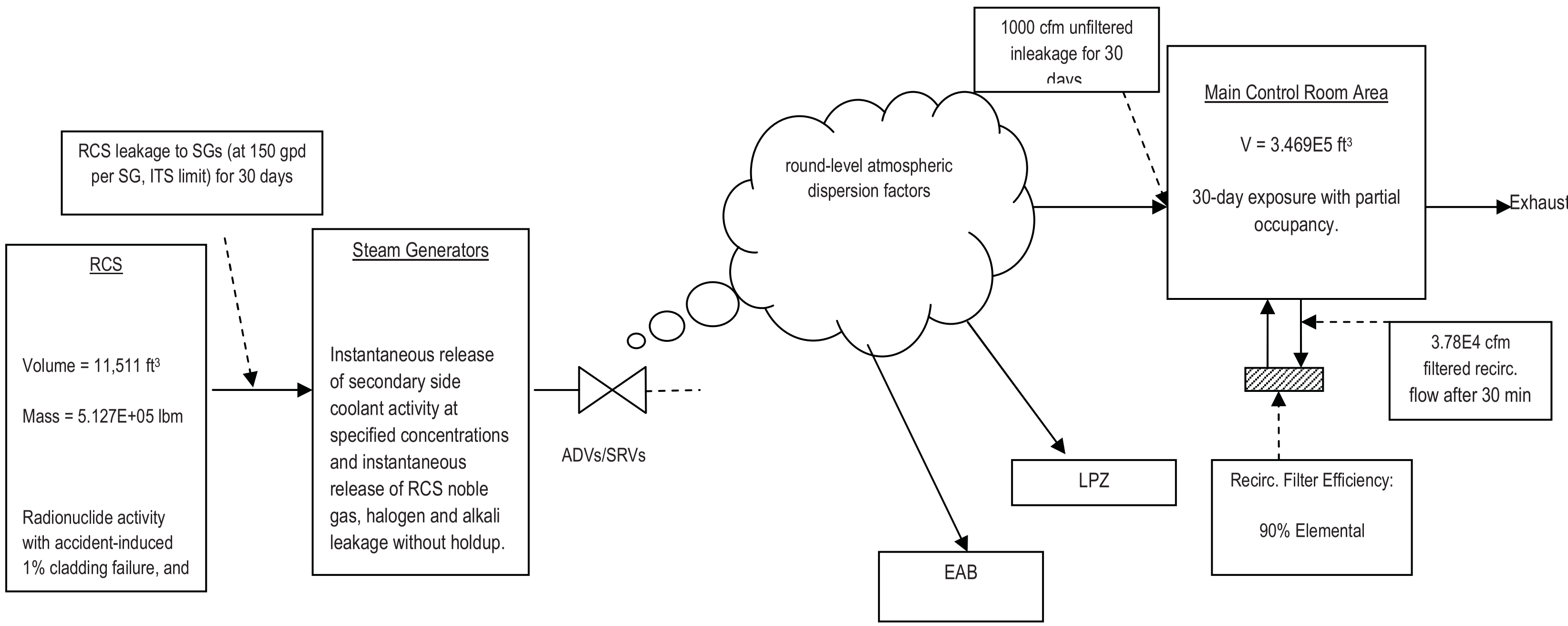


Figure 2.9.2-3 : REA - Scenario Diagram Bounding Release Pathway (Primary Containment Leakage)

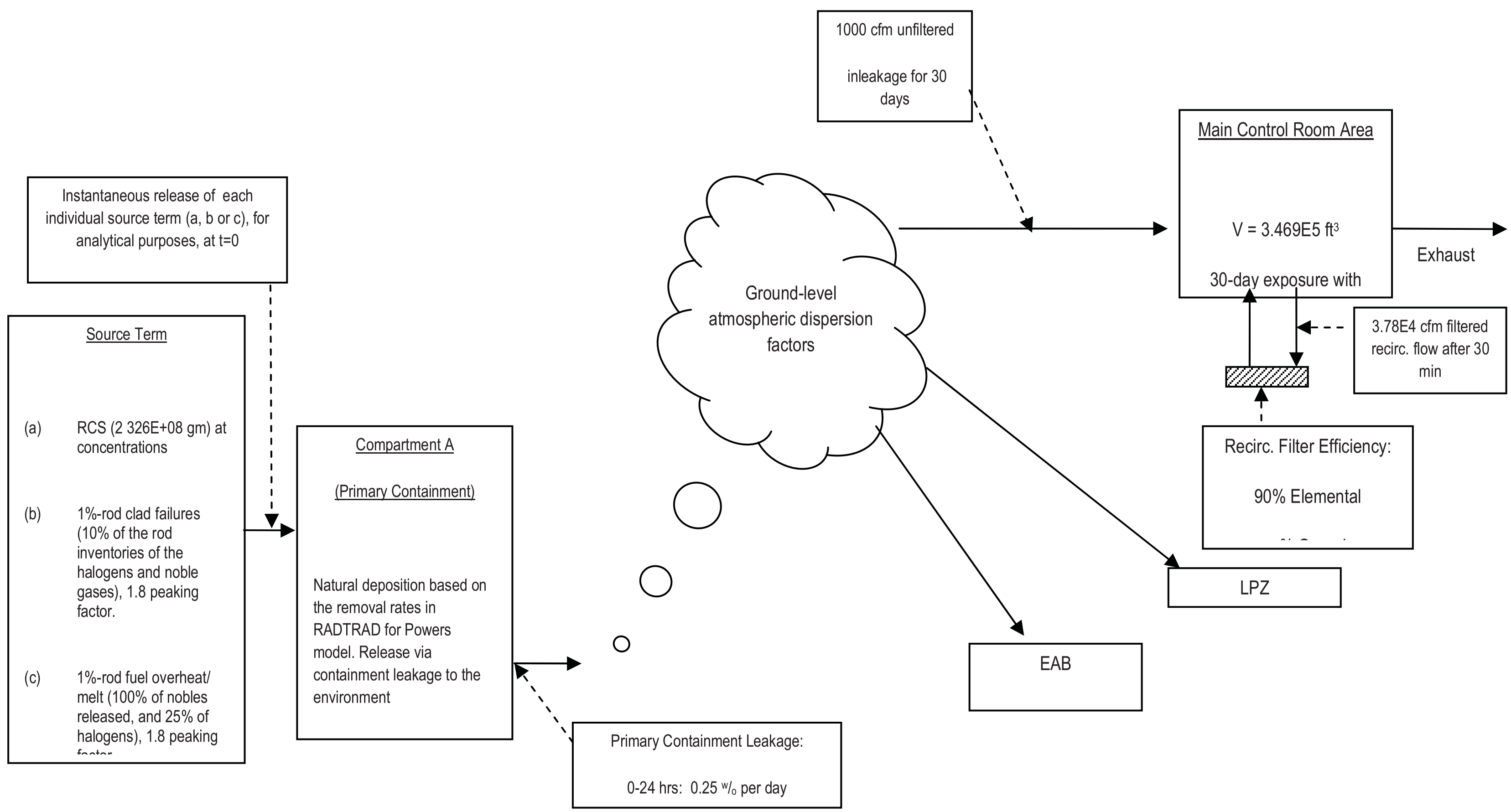


Figure 2.9.2-4: REA - Scenario Diagram Bounding Release Pathway (Secondary Side Leakage)

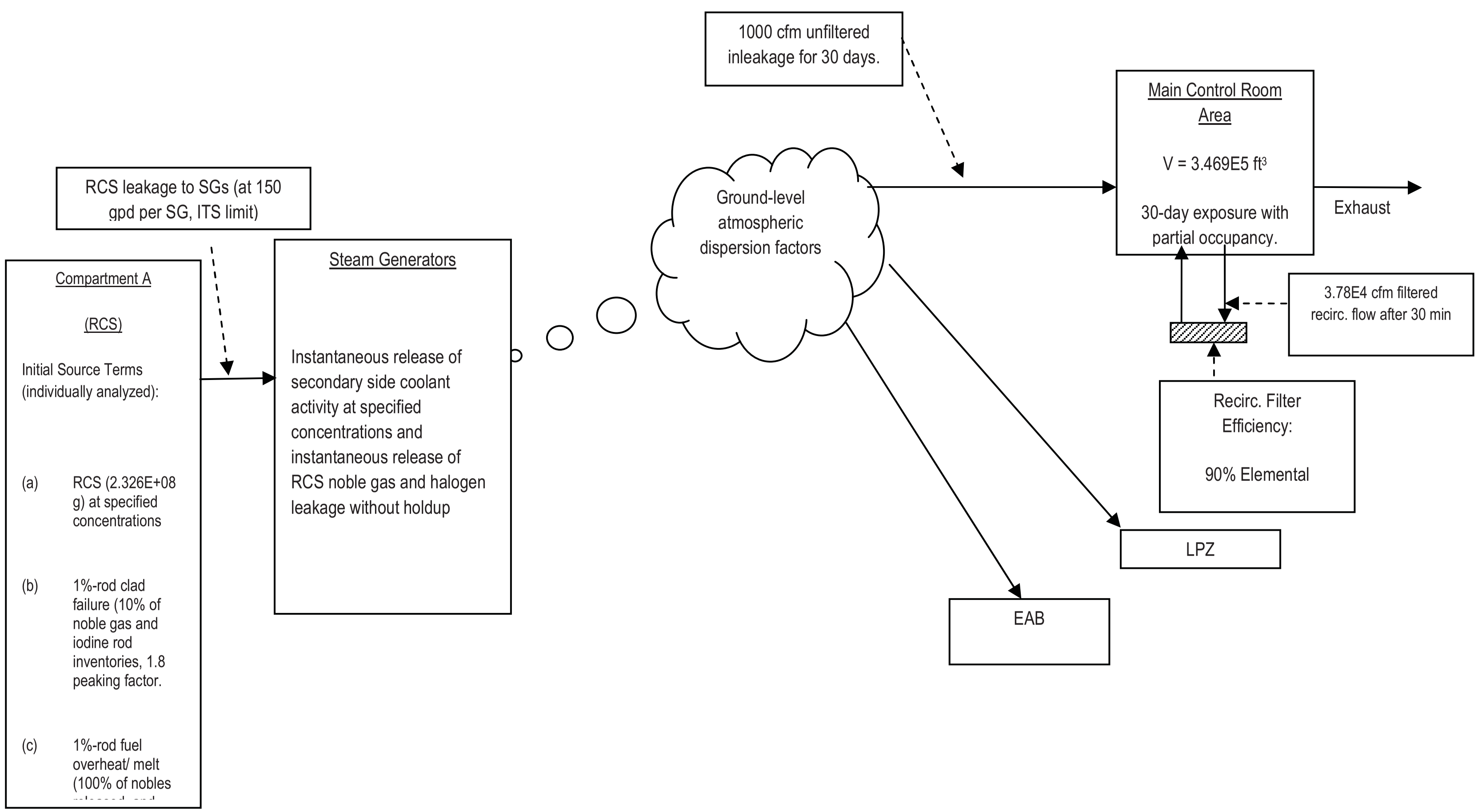


Figure 2.9.2-5: SGTR - Scenario Diagram

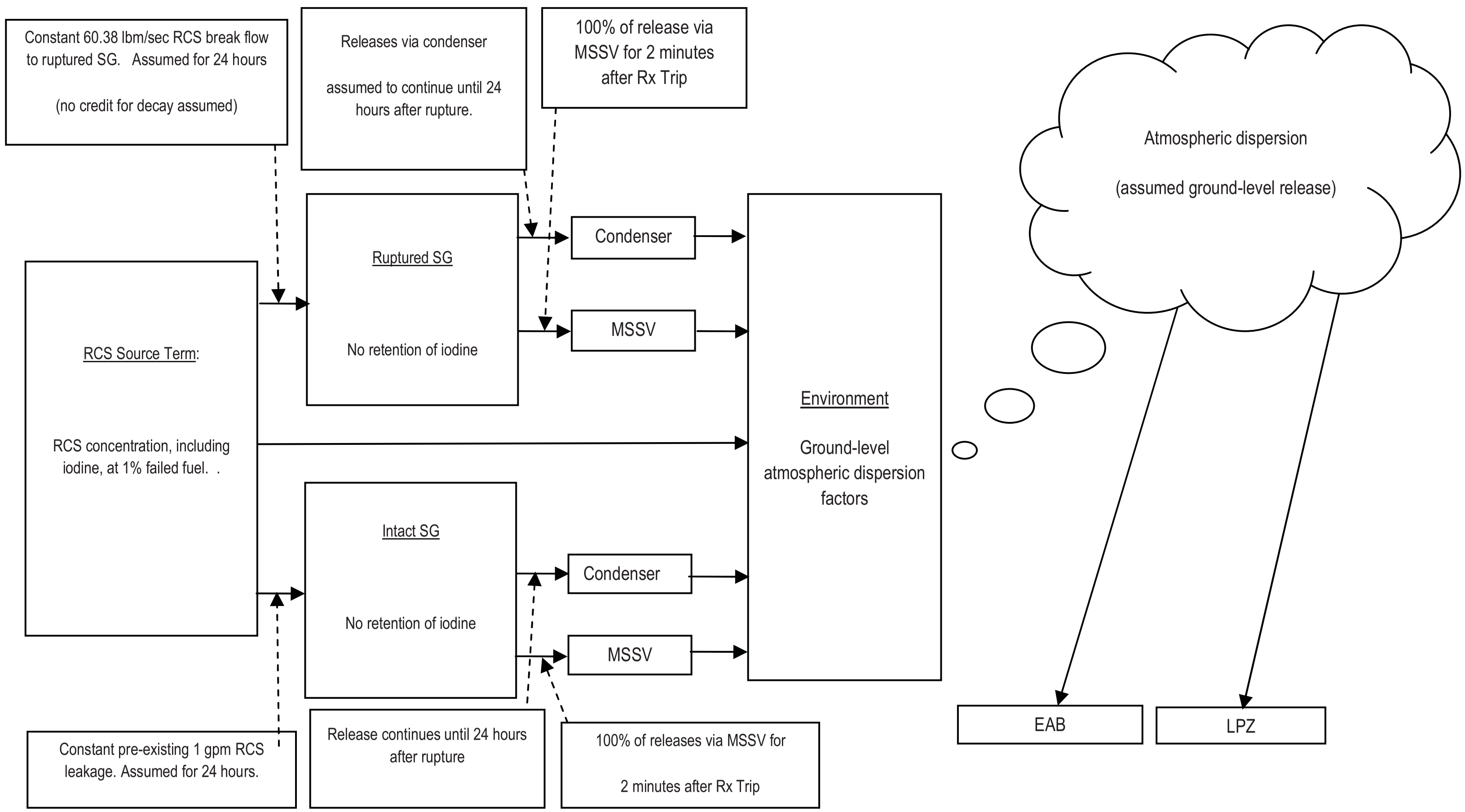


Figure 2.9.2-6: CR-3 EPU Loss of Coolant Accident – Scenario Diagram

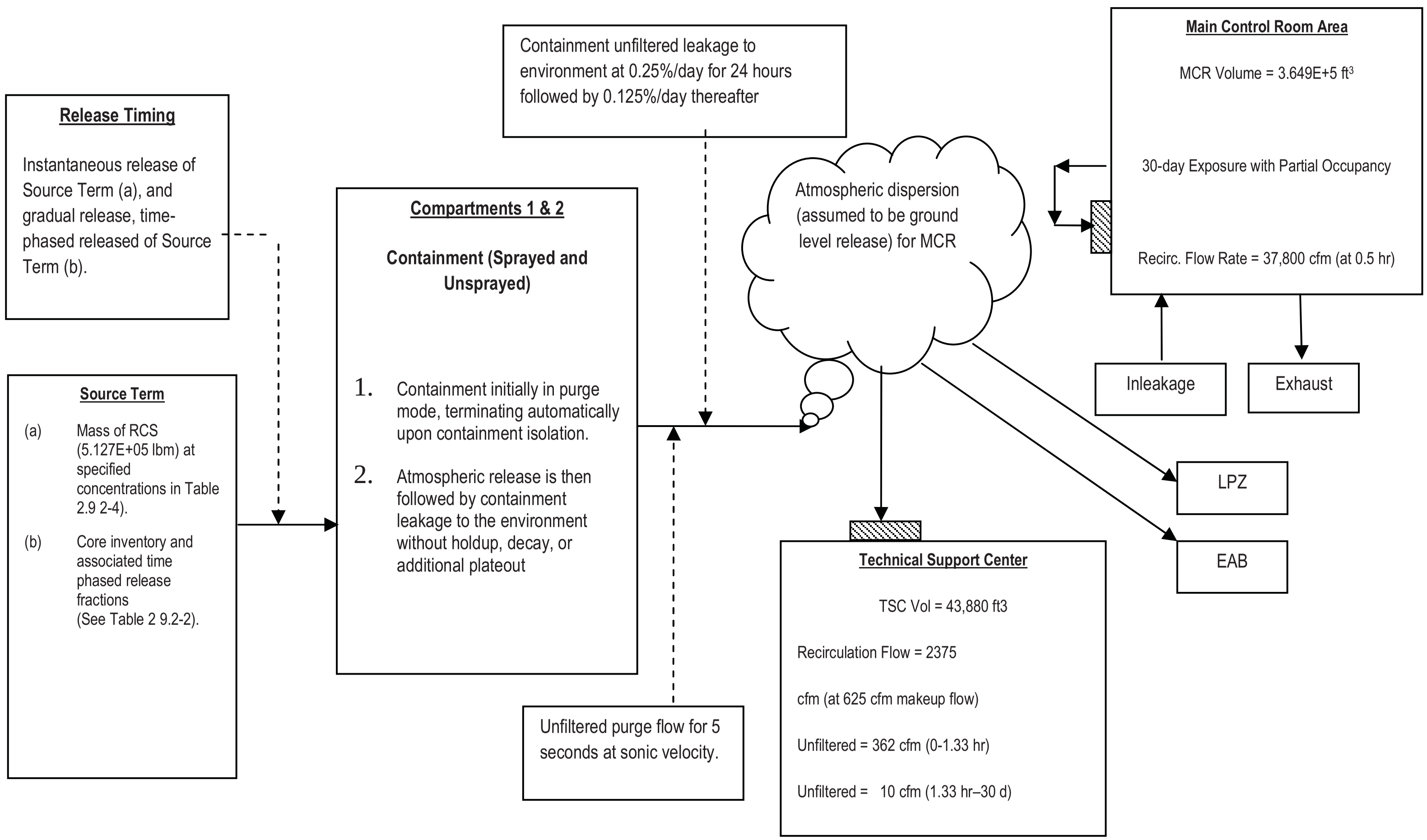


Figure 2.9.2-7: CR-3 EPU Fuel Handling Accident – Scenario Diagram for all Cases without MCR isolation

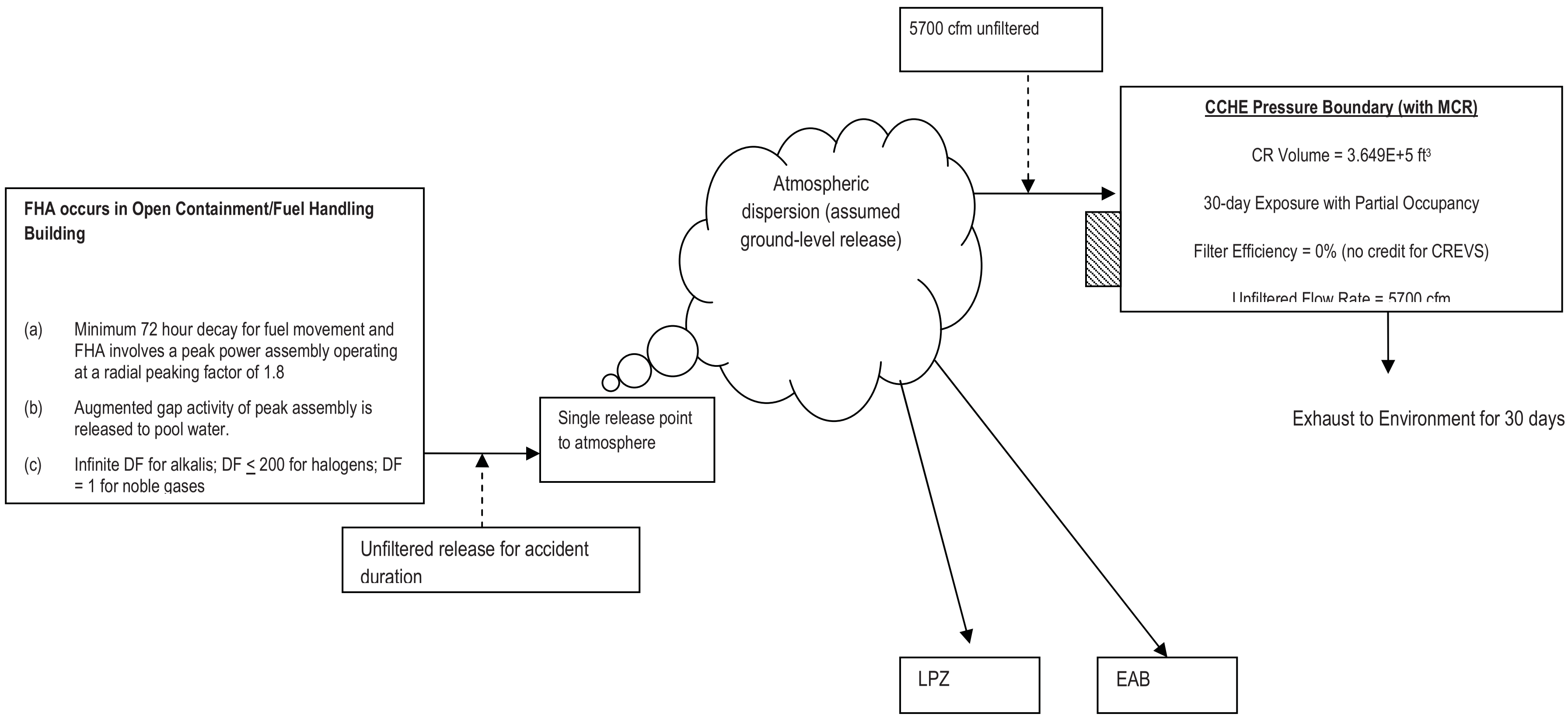
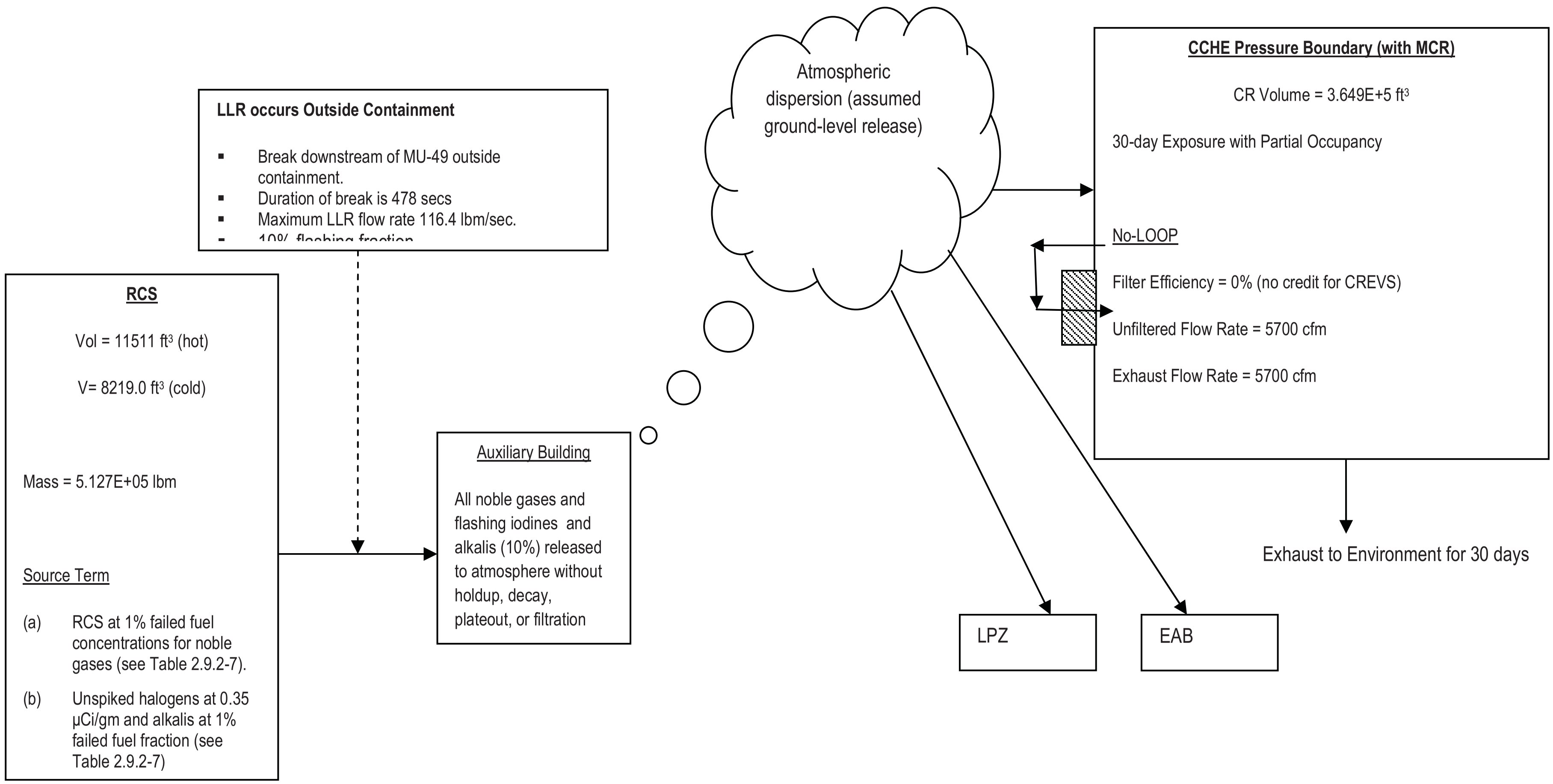


Figure 2.9.2-8: CR-3 EPU Letdown Line Break Accident – Scenario Diagram



2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

2.10.1.1 Regulatory Evaluation

CR-3 conducted its review in this area to ascertain what overall effects the proposed EPU will have on both Occupational and Public Radiation Doses and to determine that CR-3 has taken the necessary steps to ensure that any dose increases will be maintained as low as is reasonably achievable. The CR-3 review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. CR-3 evaluated how personnel doses needed to access plant vital areas following an accident are affected. CR-3 considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary.

The NRC's acceptance criteria for Occupational and Public Radiation Doses are based on:

- 10 CFR 20
- GDC-19

<u>CR–3 Current Licensing Basis</u>

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific design criteria:

- FSAR Section 1.4.11, Control Room, Control room habitability acceptance criteria is evaluated in accordance with the Alternate Source Term per 10 CFR 50.67. [GDC-19];
- FSAR Section 1.4.17, Monitoring Radioactivity Release [10 CFR 20];
- FSAR Section 1.4.68, Fuel and Waste Storage Shielding [10 CFR 20]
- FSAR Section 1.4.70, Control of Releases of Radioactivity to the Environment [10 CFR 20]

Additional details that define the CR-3 current licensing basis with respect to radiation protection of plant personnel and the public include the following:

- FSAR Section 11.3 provides criteria that radiation shielding is designed for operation at the maximum calculated thermal power. Additionally, FSAR Section 11.3 provides criteria for plant shielding necessary for the protection of operating personnel following a reactor accident so that the accident may be terminated without excessive radiation exposure to the operators or to the general public.
- FSAR Section 11.4 provides criteria that the Radiation Monitoring System (RMS) is comprised of three subsystems: Area Gamma Monitoring System, Atmospheric Monitoring System, and

Liquid Monitoring System. Area monitors support the control of radiation exposure to plant personnel. Liquid and some atmospheric monitors are provided to monitor gaseous or liquid process streams and effluent release points to unrestricted areas. Post-Accident Monitoring is provided in accordance with Regulatory Guide 1.97 and NUREG-0737 requirements to give indication of significant radioactive releases.

- FSAR Section 11.1.1 provides criteria for radiation exposure to the public due to normal operation radwaste effluents as determined by compliance with the Offsite Dose Calculation Manual (ODCM). [10 CFR 50, Appendix I, and 40 CFR 190]
- FSAR Section 11.5 provides criteria for the Health Physics and Radiation Safety Procedures that describe the programmatic content and operating philosophy of the Radiation Protection Program. The Radiation Protection Program is based upon a Risk vs. Benefit As Low As Reasonably Achievable (ALARA) methodology, and is designed to minimize the probability of occurrence of health effects within the workforce and members of the general public.

2.10.1.2 Technical Evaluation

The following evaluation is presented in five subsections:

- Normal Operation Radiation Levels and Shielding Adequacy
- Radiation Monitoring Setpoints
- Post-Accident Vital Area Accessibility
- Normal Operation Radioactive Effluents and Annual Dose to the Public
- Ensuring Occupational & Public Radiation Exposures are ALARA

Normal Operation Radiation Levels and Shielding Adequacy

Occupational dose for the normal operation of the plant results from a number of different sources, each impacted differently by the EPU. These sources and the impact of EPU on each are as follows:

<u>Sources in Containment During Power Operation</u>

The primary sources of radiation exposure in containment during power operation are gamma and neutron radiation coming from the reactor, and gamma radiation coming from activation gases, such as N-16, circulating in the primary piping. These radiation levels are approximately linearly related to power level. An increase in power level will cause a corresponding increase in the radiation levels from these sources.

One of the functions of the primary and secondary shields in containment is to reduce the dose rates from these sources to allow for limited access to select areas of the containment during power operation. During the design of the plant, this was accomplished by designing these shields to ensure dose rates at various locations were less than specified zone criteria. Per the FSAR, the zone criteria established for the generally accessible areas in the containment was ≤ 25 mrem/hr. A dose rate criterion of ≤ 25 mrem/hr ensures that access to these areas is possible for reasonable time periods, while maintaining anticipated doses within the criteria of 10 CFR 20.

While the initial shielding design analyses were performed at an assumed power level of 2544 MWt, once a plant has been operating for the length of time that CR-3 has operated, it is more accurate and more practical to evaluate the impact of a change by using actual operating experience. The actual operating data is based on actual dose rates at the locations where access has been previously made.

The following is an estimate, based on completed Radiation Work Permits, of the total worker-dose accumulated in containment during power operation from 2004 through 2008 (five calendar years):

2004 – 1.814 person-rem
2005 – 1.488 person-rem
2006 – 1.643 person-rem
2007 – 2.468 person-rem
2008 – 1.100 person-rem

The total dose has been relatively consistent as each year is between 1 and 2.5 person-rem, with an average of 1.7 person-rem/yr. To evaluate the impact due to EPU, it is conservatively assumed that there will be a 25% increase in this dose. This is based on a 16% assumed increase associated with increased power and an added 9% increase to provide a measure of conservatism. This results in a potential increase of approximately 0.4 person-rem/yr from this source due to EPU. The ALARA cost associated with an increased dose of 0.4 person-rem/yr is insignificant compared to the benefits of operation at the higher power level, and therefore ALARA principles are met.

In regard to the ability to meet the individual dose limit of 10 CFR 20, the following provides the maximum dose to any individual at the plant from 2004 through 2008:

2004 – 0.114 rem
2005 – 1.290 rem
2006 – 0.119 rem
2007 – 1.448 rem
2008 – 0.311 rem

These doses are well below the 5 rem limit of 10 CFR 20. A 25% increase in these doses remains well within the limit.

If an unexpected need for work in the containment occurs in an area where the dose rate is high enough, such that when combined with the anticipated hours required in the area the 10 CFR 20 limits or ALARA criteria would be challenged, then this particular radiation source can be reduced or eliminated by reducing reactor power or temporarily increasing shielding. Therefore, the expected increase in this source due to EPU will never preclude the ability to meet 10 CFR 20 requirements or ALARA guidelines.

<u>Piping/Component Sources</u>

This source constitutes the radiation coming from the fission and activation products that are circulating in liquid and gaseous fluids in piping and components (e.g., heat exchangers and tanks) and the activity removed by filters and demineralizers within these systems. These systems are located throughout the Radiological Control Area. The pre-startup design shielding analyses for this source assumed radioactivity concentrations in these systems based on an assumption that 1% of the fuel experiences cladding degradation.

The Reactor Coolant System (RCS) concentrations, based on this 1% cladding degradation assumption, are significantly greater than the concentrations allowed by the Improved Technical Specifications (ITS)

3.4.15. Therefore, the maximum possible dose rates from this source will be significantly less than the design values, even if operating at the ITS limit. Actual source terms are expected to be significantly less than ITS limits. EPU will not impact this conclusion, as the RCS concentrations must still be maintained less than the ITS limit, which is being reduced as part of this license amendment request (refer to ITS changes shown in Attachment 2, Operating License and Technical Specification Changes (Markup)).

Even though the effects of a higher power level will be to create more fission and activation products within the core, the concentration and mixture of radionuclides in the RCS, and other systems outside the RCS such as liquid waste, is much more sensitive to the number and type of fuel rod cladding defects than it is to power level. Due to the improvements in fuel design and operating practices, it is expected that the average number of degraded fuel rods in the future will be less than past history. Additionally, the implementation of other ALARA practices over the years, such as the use of macroporous resins in the makeup and purification system has reduced the radionuclide concentrations in the RCS. Therefore, even with the power increase associated with EPU, the concentrations of radionuclides in system piping and components is expected to be less in the future than experienced in many past operating cycles. Therefore, EPU will not preclude the ability to meet 10 CFR 20 requirements or ALARA guidelines.

Activation Product Plateout/Deposition

This source constitutes the radiation coming from activation (corrosion) products, such as Co-60, that have deposited on piping and component surfaces over time. This source is responsible for the majority of the occupational dose received in operating a nuclear power plant, and is the primary contributor to dose. While this source was not considered in the pre-startup design shielding analyses, the assumed RCS concentrations in the design analyses were sufficiently conservative compared to RCS concentrations allowed or experienced, that the shielding was sufficiently designed to compensate for this unaccounted for source.

Since this source is the primary contributor to occupational exposure, many of the ALARA practices implemented over the life of the plant have been designed to minimize the dose consequences from this source. Practices such as primary system chemistry controls and cobalt minimization are designed to minimize the production and subsequent deposition of these corrosion products. Remote tooling has been used to replace personnel access into areas where the radiation levels from this source are relatively high, such as a steam generator channel head. The net effect of these ALARA practices is the reduction in the person-rem required for normal operation and maintenance from approximately 500 person-rem per refueling outage year in the 1980s to approximately 150 person-rem per refueling outage year in the last decade. For the past decade, the non-refueling outage year dose has typically been between 5 person-rem and 15 person-rem, resulting in an average value of approximately 80 person-rem per year when outage and non-outage years are combined.

The increased core neutron flux associated with EPU will cause an increase in the activation rate of corrosion products. This would be approximately linear with power level and hence should be about 16%. However, even assuming an increase of 20% to account for secondary effects, this would only result in an increase of approximately 16 person-rem/year. The cost associated with an increased dose of 16 person-rem/yr is insignificant compared to the benefits of operation at the higher power level, and therefore ALARA principles are met. Doses will remain significantly less than the doses experienced in the 1980s and 1990s. Since the plant was able to operate within the criteria of 10 CFR 20 during those higher dose years, there will be no difficulty in meeting 10 CFR 20 criteria subsequent to EPU implementation.

Additionally, new ALARA practices continue to be employed. For example, during the 2009 refueling outage, new steam generators were installed at CR-3. These steam generators were built with several enhancements that will benefit ALARA such as electro-polishing to minimize the plateout of activated corrosion products. Zinc injection is being employed to further minimize the plateout of Co-60. Therefore, it is expected that the dose rates in and around the new steam generators will remain less than the pre-replaced steam generator dose rates, even considering the slight increase in corrosion product activation due to EPU.

Irradiated Components/Spent Fuel

This source involves the dose received while in the presence of or handling components that are radioactive because of their irradiation and activation while in or near the reactor core. Also included in this source is the spent fuel which is highly radioactive due to both irradiation and the fission process. This would include work performed in the reactor cavity during an outage, as a fraction of the radiation levels will be coming from irradiated components such as the reactor head. It also includes the dose received from irradiated components and spent fuel while handling such components, which is primarily done with water shielding above the component. EPU will directly impact the radiation levels from these components due to the increased amount of fissions and neutron flux levels for irradiation. The increase should be approximately equal to the 16% increase in power level, but is assumed to be 20% to account for secondary effects such as core design.

The dose received from this source is part of the average 80 person-rem/yr discussed in the previous section. Therefore, the potential increase of 20% would be a part of the potential increase of the 16 person-rem discussed above. The dose from this source is estimated to be a small fraction of the average 80 person-rem/yr and hence would be a small fraction of the projected increase.

The shielding design for movement of spent fuel and associated irradiated components such as control rod assemblies, is to maintain sufficient water depth above the component to maintain the increased dose rate to personnel less than an additional 2 mrem/hr. Experience has shown that the water depth is sufficient such that the increased dose rate to personnel from spent fuel in transit is less than 1 mrem/hr. A 20% increase would therefore maintain the design dose rate criterion of 2 mrem/hr.

Public Dose

Based on the most recent land-use census, the nearest residence is 2.4 miles away. This large distance, combined with the general design principle of providing sufficient shielding to ensure dose rates at generally accessible areas onsite are ALARA, ensures that the direct dose to the public offsite will be insignificant. This general ALARA principle, to minimize the dose consequences at generally accessible areas onsite, applies to both original structures and to any new structures that are added, such as shielded buildings to store solid radioactive waste or removed components such as steam generators.

One of the most significant contributors to public direct dose from nuclear power plants is from gamma radiation from activation gases such as N-16. However, this contributor is only applicable to BWRs, where these activation gases are transported to a turbine building with limited shielding. For a PWR, such as CR-3, most of the activation gases remain in the containment building or are contained within the waste gas system for decay prior to release. Containment buildings are designed with thick walls to provide adequate shielding for an assumed post-accident source; therefore the dose rates from these activation gases are insignificant outside of containment.

CR-3 maintains thermoluminescence dosimeters (TLDs) at a number of locations onsite and offsite for the measurement of direct radiation as part of the Radiological Environmental Monitoring Program. The offsite TLDs show no significant difference in radiation levels compared to the control location, and hence help confirm that offsite direct dose is insignificant. A 20% increase due to the impact of EPU on sources that could result in direct dose would still result in offsite direct dose that is insignificant compared to the 25 mrem limit of 40 CFR 190.

10 CFR 20 imposes a limit of 100 mrem/year for members of the public who may frequent locations onsite. CR-3 demonstrates compliance with this limit through the use of results from area TLDs located throughout the generally accessible areas of the site, combined with conservatively estimated occupancy factors, which is the number of hours per year that a member of the public may be at a given onsite location.

The results of the evaluation for 2008 of public dose onsite was 1.0 mrem/year for a member of the public in the Restricted Area and 4.8 mrem/year for a member of the public in the Controlled Area. These results are similar to previous years. A potential 20% increase in these doses, as a conservative estimate of the consequences of EPU, would result in doses remaining less than 10% of the 10 CFR 20 limit of 100 mrem/year.

Radiation Monitoring Setpoints

As discussed in the CR-3 FSAR Section 11.4, the Radiation Monitoring System (RMS) has three main subsystems: Area Gamma Monitoring System, Atmospheric Monitoring System, and Liquid Monitoring System. Area monitor alarm setpoints are established to provide early warning of changing radiological conditions. Setpoints for area radiation monitors are based on radiation surveys made during normal plant operation. Alarm setpoints for process/effluent radiation monitors are established to indicate leakage or malfunction of equipment or a potential for a radioactivity release that may exceed a release rate limit. The high alarm setpoint of some effluent radiation monitors will also initiate interlocks that terminate releases to the environment. Effluent radiation monitor setpoints at CR-3 are based on methodology prescribed by the ODCM. The methodology of the ODCM is not power level dependent, but is based on assuring that regulatory limits and guidelines are met.

With all things being equal (e.g., fuel leakage, component leakage) except power level, the EPU is expected to increase the concentration of radioactive isotopes in most streams and components by approximately the percentage of the core power uprate. The relative isotopic compositions in the process and effluent streams are not expected to change significantly due to EPU.

Currently, the radioactivity levels measured by many process monitors and effluents monitors are either indistinguishable from background, or a small fraction of the monitor's measurement capability, and will remain so during post EPU operations for the reasons given above.

The radioactivity levels in process streams and in effluent streams are affected more by fuel integrity and the integrity of plant components than small changes in power level. This is illustrated by examining the early part of the CR-3 operational history (Reference 4) when activity levels in process and effluent streams were much higher, sometimes by orders of magnitude. With the implementation of zinc injection prior to EPU and ongoing efforts to assure that fuel integrity is maintained, it is not expected that EPU will cause a long term increase of most fission and corrosion products routinely observed in plant effluents or plant process streams.

Based on the evaluation outlined above, radiation monitor setpoint bases and methodology will remain valid following EPU.

Post-Accident Vital Area Accessibility

NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980, Section II.B.2, requires demonstration of the ability for post-accident access to "vital areas." Vital areas is defined in the NUREG as, "Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area." The analyses to demonstrate this ability are often referred to as post-accident shielding analyses or mission dose calculations.

NUREG-0737 specified the source term to be used for this demonstration as a percentage of the core inventory released into specified locations. Since the EPU will impact the calculated core inventory, the EPU conditions are evaluated for impact on the ability to ensure access to vital areas.

The CR-3 Control Room and Technical Support Center are areas that require continuous occupancy and the dose analyses for these areas is performed in habitability calculations performed as part of the design basis accident analyses. The impact of EPU on the habitability for these two areas is addressed in Section 2.9.2, Radiological Consequences Analysis. Additionally, the following vital areas are addressed in this section in regard to the impact of EPU on accessibility:

- Emergency Diesel Generator (EDG) Rooms
- RM-A1 and RM-A2 – high range effluent radiation monitors
- Methods for adding water to the Emergency Feedwater Tank

<u>Emergency Diesel Generator (EDG) Rooms</u>

During a Large Break LOCA (LBLOCA) there are no post-accident sources in the general vicinity of the EDG rooms, and time spent in the rooms for activities would be unlimited. Therefore, for a LBLOCA, the access and egress is controlling. For a Small Break LOCA (SBLOCA), there could be sources of post-accident activity near the EDG rooms. Hence, time in the EDG room would be limited. There are no required missions in the EDG room. It was included in the NUREG-0737 response as a desired area for access given the importance of the EDGs and the potential desire to access the rooms to take immediate actions to restore operability (e.g., manually operate a component that is designed to operate automatically, but didn't for some reason). The intent of the SBLOCA dose analysis was to demonstrate that there was sufficient time available (minutes) to take such manual actions. It was not intended to demonstrate that time was available to make major repairs to the EDG should they fail. The existing calculations resulted in a time limit of 25 minutes in the EDG room for a SBLOCA to maintain total dose (i.e., including access and egress) within the 5 rem guidelines.

Based on the total dose from the EPU analysis for the LBLOCA, the calculated access/egress dose increased from 0.389 rem to 0.419 rem and hence this mission can still be accomplished with a dose much less than the 5 rem guidelines, with unlimited time in the EDG room. Based on the SBLOCA results, the time available for actions in the EDG rooms to maintain a dose less than 5 rem is reduced from 25 minutes to approximately 10 minutes. This still provides sufficient time to perform short compensatory actions in the EDG rooms.

<u>RM-A1 and RM-A2 – high range effluent radiation monitors</u>

The Grab Sample Station, which contains the RM-A1 and RM-A2 high range monitors particulate and iodine filters, is located in the Auxiliary Building in Zone 32. Based on the total dose from the EPU analysis, the calculated dose increased from 0.769 rem to 1.248 rem and hence this mission can still be accomplished with a dose less than the 5 rem guidelines.

<u>Methods for adding water to the Emergency Feedwater Tank</u>

The three potential methods for adding water to the Emergency Feedwater Tank (EFT) include adding water from the Condensate Storage Tank (CDT), a Fire Storage Tank, or from the hotwell. Each of these methods is performed in a location remote from any post-accident sources, as two are performed outside on the berm and one in the turbine building. Therefore, the only source assumed in the existing mission dose analysis is from the assumed radioactive plume from a LOCA release. These actions are required for a SBLOCA, but not for a LBLOCA. The existing calculation conservatively assumed a SBLOCA plume dose rate of 0.01 rem/min. Using this dose rate, and the determined mission times, the following were the calculated mission doses:

- Cross-tie EFT/CDT via CDV-103 – 12.5 minutes Dose = 0.125 rem
- Cross-tie EFT with Fire Storage Tank - 12 minutes Dose = 0.12 rem
- Fill EFT-2 from hotwell via CDV-259/260 – 5.5 minutes Dose = 0.055 rem

Conservatively, assuming that the SBLOCA plume dose rate increases by 100% as a result of EPU, the projected total mission dose would be approximately 0.25 rem and therefore continue to be well below the 5 rem guideline.

Normal Operation Radwaste Effluent and Annual Dose to the Public

CR-3 is committed to the requirements of the ODCM, which provides effluent specifications based on 10 CFR 20, as well as 40 CFR 190, and 10 CFR 50, Appendix I. The ODCM will continue to maintain the same limits and operational requirements after implementation of EPU as these limits are not a function of power.

As EPU does not change existing radioactive waste systems or plant operating procedures related to processing liquid or gaseous wastes, the increase in the amount of radioactive material released to the environment, in liquid and gaseous effluents or shipped offsite in the form of solid waste, will be approximately proportional to the increase in reactor coolant activity. In turn, reactor coolant activity increases are expected to be directly proportional to the increase in core inventory for fission products and power level or activation products.

To conservatively estimate the impact of EPU on effluents and to verify that public dose limits will continue to meet Appendix I guidelines, as implemented by the ODCM, a scaling factor, based on relative core inventory (pre- vs. post EPU), is applied to current effluent doses.

The worst case annual effluent doses (Reference 5), from 2000 through 2008, are presented in Table 2.10-1. These results are based on actual, not hypothetical, releases and were calculated based on the methods prescribed by the ODCM. Doses are far below the Appendix I based limits with the worst case values being approximately 0.1% of the limit.

Scaling factors can be developed in several ways – comparing pre-and post EPU core inventory for select radionuclides and comparing post EPU reactor coolant concentrations with pre-EPU concentrations are two methods.

A comparison of end of cycle core activities was chosen to derive a conservative scaling factor. Select radionuclides (chosen from those which are identified in plant effluents) were compared for pre-and post EPU cores. The increase in core activity in the post EPU core was approximately equal to the increase in power for most of nuclides of interest (e.g., I-131, I-133, Sr-90, Cs-137). A scaling factor of 2 was chosen as it is conservative and bounds the percent increase for all radionuclides of interest. When applied to pre-EPU doses, it can be seen in Table 2.10-1 that a factor of 2 increase demonstrates that EPU effluent doses would remain a small fraction (< 1%) of the ODCM limits (based on 10 CFR 50, Appendix I), all other things equal (e.g., fuel and component leakage).

As discussed above, the public dose at offsite locations from direct radiation will remain insignificant. With an effluent dose consequence that is less than the ALARA guideline of 10 CFR 50, Appendix I, the combined (direct and effluent) dose consequence will remain well below the 25 mrem/year fuel cycle standard of 40 CFR 190.

The EPU does not change the types of solid radwastes which are generated, or add a new type of solid radwaste as there are no new inputs being added to the radwaste system, and the radwaste system will not be modified as part of EPU. Most solid waste that is in the form of spent resin depletes on non-radiological chemical parameters and not on radionuclide loading. And as the purity of makeup water and chemical control of the plant is not being relaxed under EPU, the amount of resin waste which accounts for most of the low level radioactivity (in terms of Curies), is not expected to change significantly. The increase in RCS activity, which may occur due to the EPU, will result in a small change in the amount of radioactivity collected on solid radwaste media.

It is concluded that following the EPU, the liquid and gaseous radwaste effluent treatment systems, in conjunction with the controls provided by the ODCM, will remain capable of maintaining normal offsite doses within the regulatory requirements.

Ensuring Occupational & Public Radiation Exposures are ALARA

The Radiation Protection Program at CR-3 ensures the internal and external radiation exposures to station personnel and the general population will be within applicable limits and will be ALARA, as described in Chapter 11 of the CR-3 FSAR, and enforced by station procedures.

Implementation of the overall requirements of 10 CFR 50, Appendix I, relative to the utilization of radwaste treatment equipment to insure that radioactive discharges and public exposure are ALARA, are formalized in the ITS and ODCM requirements and implemented by plant procedures.

The ALARA policy governs all work in restricted areas at CR-3. Management commitment to the policy is reflected in the design of the plant, and the plant operation and maintenance procedures. Design features credited to support the CR-3 commitment to ALARA personnel exposures such as shielding, ventilation systems, and radiation monitoring will remain unaffected by the EPU.

The discussion above demonstrates that dose limits imposed by regulatory requirements will be met under EPU conditions. The EPU does not impact plant design features needed to keep occupational or public dose ALARA nor the procedures used to implement the ALARA program for plant personnel and the general public.

Therefore, it can be concluded that additional steps are not necessary to ensure dose increases which may occur under EPU conditions will be maintained ALARA.

2.10.1.3 Conclusion

CR-3 has reviewed the effects of the proposed EPU on radiation source terms and plant radiation levels. The evaluation concludes that CR-3 has taken the necessary steps to ensure that any increases in radiation doses will be maintained as low as reasonably achievable. CR-3 further concludes that the proposed EPU meets the requirements of FSAR Sections 1.4.11, 1.4.17, 1.4.68, and 1.4.70. Therefore, CR-3 finds the proposed EPU acceptable with respect to radiation protection and ensuring that Occupational and Public Radiation Doses will be maintained as low as reasonably achievable.

2.10.1.4 References

1. ANSI/ANS-18.1-1999, “Radioactive Source Term for Normal Operation of Light Water Reactors,” and ANS/ANSI-18.1, ERRATA, December 1, 2005.
2. USNRC, NUREG-0017, Rev. 0, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, PWR-Gale Code,” April 1976.
3. NUREG-0737, “Clarification of TMI Action Plan Requirements,” November 1980.
4. CR-3 Semiannual Radioactive Effluent Release Reports, 1977 – 1983.
5. CR-3 Radioactive Effluent Release Reports, 2000 – 2008.

Table 2.10-1
Historic and Projected EPU Offsite Doses Compared to 10 CFR 50, Appendix I ALARA Guidelines

	Historic CR-3 Doses (2000 – 2008)	Projected Post-EPU Doses (x 2 scaling)	Appendix I ALARA Guideline	Units
Liquid				
Total Body	9.39×10^{-5}	1.88×10^{-4}	3	mrem/yr
Maximum Organ	3.65×10^{-3}	7.30×10^{-3}	10	mrem/yr
Gaseous				
Gamma Air Dose	2.69×10^{-3}	5.38×10^{-3}	10	mrad/yr
Beta Air Dose	1.95×10^{-2}	3.90×10^{-2}	20	mrad/yr
Total Body	5.61×10^{-3}	1.1×10^{-2}	15	mrem/yr
Maximum Organ	1.68×10^{-2}	3.36×10^{-2}	15	mrem/yr

2.11 Human Performance

2.11.1 Human Factors

2.11.1.1 Regulatory Evaluation

The area of Human Factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. CR-3 conducted a human factors evaluation to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The CR-3 review covered changes to operator actions, human-system interfaces, procedures and training needed for the proposed EPU.

The NRCs acceptance criteria for Human Factors are based on:

- GDC-19,
- 10 CFR 50.120, and
- 10 CFR 55.

Additionally, guidance is provided in GL 82-33.

CR-3 Current Licensing Basis

As noted in FSAR Section 1.4, the design criteria used during the licensing of CR-3 predates the GDC provided in 10 CFR 50, Appendix A. The origin of the CR-3 specific criteria relative to the Atomic Energy Commission proposed GDC is discussed in FSAR Section 1.4. The criteria presented in FSAR Section 1.4 were found by the NRC to be acceptable for the design, construction, and operation of CR-3.

The following are the applicable CR-3 specific criteria:

- FSAR Section 1.4.11, Control Room [GDC-19]

Additionally, FSAR Chapter 12, Conduct of Operations, provides guidance related to the acceptance criteria for implementation of the requirements of 10 CFR 50.120. Training of selected personnel who provide onsite support to CR-3 is based on a systematic approach to training. 10 CFR 55 is described in FSAR Section 12.2, Training, and addresses the qualifications for licensed operators. The training program is accredited by the National Nuclear Accrediting Board and operates under the auspices of the Institute of Nuclear Power Operations (INPO). [10 CFR 50.120 and 10 CFR 55]

The CR-3 implementation of Supplement 1 to NUREG-0737 (GL 82-33) is discussed in FSAR Sections 12.2.3.8, 14.2.2.5 and the Improved Technical Specifications (ITS) Bases 3.3.17.

2.11.1.2 Technical Evaluation

Introduction

Human Factors Engineering and Human Performance initiatives are foundational characteristics that help ensure that plant operators can effectively and safely operate the facility under normal, abnormal, and emergency conditions. When initiating a plant change, the engineering change (EC) process requires the completion of a Human Factors review for changes that may impact the Control Room layout (alarms, indication, appearance or performance). In addition, plant operations personnel participate in the EC process by contributing to the conceptual design studies, human factors reviews, and procedure impact assessments.

Approved EC packages are reviewed by qualified training department personnel to determine (1) the impact on accredited training programs and program materials, and (2) the impact on simulator control room hardware, software models and instructor facility features and graphics. The schedule for the completion of training program revisions and simulator modifications will support the procedure validation process and the startup training programs.

Description of Analyses and Evaluations

To ensure changes associated with the EPU do not introduce unanticipated consequences, a review of the effects of those changes on human performance interfaces, procedures and training was performed using a standard set of questions developed by the NRC staff and published in RS-001, Review Standard for Extended Power Uprates, Insert 11. The following are the NRC staffs questions and CR-3 responses.

Question 1: Changes in Emergency and Abnormal Operating Procedures

Describe how the proposed EPU will change the plant emergency and abnormal operating procedures.

Question 1 CR-3 Response:

The existing CR-3 Emergency and Abnormal operating procedure set will continue to provide the guidance necessary to respond to the full spectrum of anticipated events using existing mitigation strategies. The following procedure changes result from changes to the physical plant and plant response resulting from the EPU. Several of these changes significantly reduce operator burden. The B&W EOP Technical Basis Document (TBD) is in the process of being revised for plant operation at EPU power levels. Necessary conforming EOP changes will be made as a result of that revision.

1. The Loss of Subcooling Margin (LOSCM) procedure will be revised to reflect the automatic actuation of the new Inadequate Core Cooling Mitigation System (ICCMS). The ICCMS will automatically trip RCPs and reset the EFIC ISCM level setpoint during a LOSCM event. This will facilitate completion of existing time critical actions to (1) trip all RCPs within one minute of LOSCM, and (2) to select EFIC ISCM level setpoint within ten minutes of LOSCM. For events involving a LOSCM and inadequate High Pressure Injection (HPI) flow the existing EOP requires the operator to manually depressurize the SGs to enhance primary-to-secondary heat transfer. This guidance will be revised to reflect verification of the automatic actuation of the new Fast Cooldown System (FCS). At EPU conditions FCS actuation is required within ten minutes of a LOSCM event with inadequate HPI flow. The revised procedure will include specific guidance for ensuring that each new automatic actuation function performs properly. Verification steps will include appropriate contingency actions to be followed if any of the automatic functions described above fail. Contingency actions will be validated to ensure all technical and timing requirements are satisfied. In addition to the above, guidance will be provided for resetting the Inadequate Core Cooling Mitigation System following an automatic actuation when subcooling margin has been restored. The Inadequate Core Cooling Mitigation System (ICCMS) and Fast Cooldown System (FCS) are described in more detail in Appendix E.

2. The Loss of Subcooling Margin, LOCA Cooldown, HPI Cooldown, Inadequate Core Cooling, Loss of Decay Heat Removal (DHR) procedures and several Emergency Operating Procedure rules and enclosures will be revised to incorporate new Low Pressure Injection (LPI) minimum flow rate requirements for LPI flow, Core Flood Tank (CFT) isolation, ECCS Suction Transfer, and procedure transition. New LPI flow values are required because of the installation of the new LPI crosstie lines which allow one LPI pump to provide flow to both LPI nozzles. The changes to these procedures are limited to specifying new administrative requirements for performing existing actions. The actions themselves have not been modified.

3. The Loss of Subcooling Margin (LOSCM) procedure will be revised to include a requirement to terminate any fluid additions to the Makeup and Purification System from the Boric Acid Storage Tanks, the RC Bleed Tanks, or Demineralized Water System.

4. The Inadequate Core Cooling procedure will be revised to include guidance to verify actuation of the Fast Cooldown System for events involving a LOSCM with inadequate HPI flow.

5. The LOCA Cooldown procedure and EOP enclosures will include revised High Pressure Injection (HPI) termination criteria. The current HPI termination guidance will be revised to state HPI pumps can be terminated during transition from the BWST to the RB Sump if all of the following conditions are true: (1) both trains of LPI are in operation, (2) LPI flow in both trains is greater than 2192 gpm (uncorrected), and (3) LPI line A-to-B differential pressure is less than 80 psid (uncorrected).

6. The RCP Recovery procedure will be revised to address RCP recovery following a LOSCM and automatic actuation of the Inadequate Core Cooling Mitigation System (ICCMS). The procedure will require the ICCMS actuation logic to be reset prior to RCP restart.

7. The Loss of Subcooling Margin, LOCA Cooldown, HPI Cooldown, Inadequate Core Cooling, SGTR, Natural Circulation Cooldown, Loss of DHR, RCS Boration, Loss of RCS pressure, Shutdown from Outside the Control Room procedures and several EOP Rules and enclosures will be revised to eliminate references to DHV-210 and DHV-211. Decay heat valves DHV-210 and DHV-211 are being removed as part of the LPI crosstie modification. DHV-210 and DHV-211 were normally open valves in the LPI/DHR injection flow path that had to be manually verified open for system operability. Removal of these valves improves LPI flow by reducing injection line flow resistance and reduces operator burden by eliminating the requirement to verify their position during EOP response.

8. The Station Blackout procedure will be revised to (1) require the operator to “Bypass” the FCS if subcooling margin is lost, and (2) eliminate the local operator action to align a back up air supply to the ADVs. FCS bypass will allow existing required procedure actions to be completed by manual operator action when LOSCM occurs. The addition of a new safety related ADV air system eliminates the need to manually align air to the ADVs.

9. The Post Accident Boron Concentration Management EOP enclosure will be revised to require that boron precipitation mitigation be placed into service during the transition from the BWST to the RB Sump using the new hot leg injection system. This action will no longer depend on calculations based on results of a boronometer indication or time but rather the condition of sump swapover.

10. The RCS Boration procedure will be revised to provide the option to utilize the new makeup tank (MUT) bypass valve MUV-661 for establishing RCS boration.
11. The Plant Runback procedure will be revised to eliminate actions associated with the automatic asymmetric control rod runback, which is being deleted. Procedural guidance for asymmetric control rod response will be removed from the Plant Runback procedure and relocated to a new abnormal procedure or procedure section, (2) incorporate revised integrated control system (ICS) runback rates and limits for FW and RCP trip runbacks, and (3) correct various references to thermal power limits (MWt).
12. The Turbine Building Flooding procedure will be revised to remove condensate pump "uncoupling" from the entry conditions. Variable speed condensate pumps (CDPs) are being replaced with constant speed pumps for the EPU, therefore magnetic coupling failure is no longer a valid indicator of turbine building flooding.
13. The Fire Protection procedure will be revised to require the operators to select the new ADV Override switches to the override position upon confirmation of a fire in the control complex.
14. The Shutdown From Outside the Control Room procedure will be revised to require the operators to (1) ensure the new ADV override switches are in the override position prior to evacuating the main control room, and (2) to reset the associated ADV override lock out devices in order to establish ADV control from the Remote Shutdown Panel (RSP).

Question 1 Conclusion:

The anticipated changes to the emergency and abnormal operating procedures do not alter basic mitigation strategies, and do not involve significant changes in equipment usage or required operator actions. In addition, no new procedures are required as a result of the EPU. Automating the manual operator actions for LOSCM and inadequate HPI flow events, and simplifying the actions for post accident boron precipitation control will result in a significant reduction in operator burden. The procedure changes outlined above, and any others that become necessary as a result of new Technical Bases Document guidance, will be implemented in accordance with a rigorous change control process that includes compliance with a writer's guide, and a thorough verification and validation process which includes verification of all time critical operator actions and dose assumptions. The procedures will be simulator validated by multiple crews prior to approval. Active operating personnel will be thoroughly trained and evaluated on the approved procedures prior to implementation. Adherence to this process ensures the CR-3 emergency and abnormal operating procedures continue to provide appropriate guidance to cover the full spectrum of anticipated events.

Question 2: Changes to Operator Actions Sensitive to Power Uprate

Describe any new operator actions needed as a result of the proposed EPU. Describe changes to any current operator actions related to emergency or abnormal operating procedures that will occur as a result of the proposed EPU.

Identify and describe operator actions that will involve additional response time or will have reduced time available. The response should address any operator workarounds that might affect these response times. Identify any operator actions that are being automated or being changed from automatic to manual as a result of the power uprate. Provide justification for the

acceptability of these changes.

Question 2 CR-3 Response:

As a result of the EPU several EOP/AP directed operator actions will change. The required changes are not complicated and do not require new or enhanced diagnostic skills. Several changes result in an overall reduction in operator burden. Each required change to operator actions will be implemented by procedure, and each procedure change will be appropriately validated. Operators will receive formal classroom and simulator training on each change prior to implementation.

1. **During a LOSCM event the existing time critical manual operator action to trip RCP motors within one minute of losing subcooling margin (SCM) will be automated.**

   To support the EPU a new, safety related Inadequate Core Cooling Mitigation System (ICCMS) will be installed. The ICCMS will monitor RCS pressure, incore thermocouple temperatures, and reactor trip status. The ICCMS compares RCS pressure/temperature to a minimum SCM limit and HPI flow to a minimum HPI flow limit based on RCS pressure (HPI flow monitoring function is discussed in response 3 below). If RCS SCM is lost, concurrent with a reactor trip, the system will automatically trip the RCPs within one minute. This action is credited in the LOCA analysis for operation at MUR power and is currently approved by the NRC as a 1 minute manual operator action.

   The ICCMS is fully automatic and requires no operator interaction. New SCM and HPI flow margin indicators and associated controls, a new ICCMS monitoring panel, and ICCMS related annunciator alarms are being added to the MCB to allow the operator to properly evaluate ICCMS status. The ICCMS will be included in mandatory pre-startup training for active operations personnel. Classroom training will cover ICCMS design features, the purpose of the system, and the expected operator actions associated with ICCMS alarms, actuations and failures. The ICCMS will be installed on the CR-3 simulator prior to pre-startup simulator training. Simulator training will include a demonstration exercise to familiarize operators with ICCMS design features and alarms. Unannounced casualty scenarios will verify correct operator response to ICCMS alarm functions and actuations, and the ability to perform the required actions following a ICCMS failure.

   The Loss of Subcooling Margin procedure will be revised to include specific guidance for ensuring each of the automatic actuation functions of the ICCMS occur within their allotted time. Each verification step will include appropriate contingency actions to be followed if the automatic actuation system fails. Contingency actions will be validated to ensure applicable technical and timing requirements are satisfied.

   The Inadequate Core Cooling Mitigation System (ICCMS) is described in more detail in Appendix E.

2. **During a LOSCM event the existing time critical manual operator action to select the EFIC ISCM level setpoint within 20 minutes will be automated. Timing for this action will be reduced from 20 minutes to 10 minutes.**

   During a LOSCM event current EOP guidance requires the operator to manually transfer the EFIC control setpoint for SG level control from the natural circulation setpoint to the Inadequate Subcooling Margin (ISCM) setpoint within 20 minutes of losing SCM. This action is credited in the LOCA analysis for operation at MUR power and is currently approved by the NRC as a manual operator action. The LOCA analysis for EPU credits the ISCM setpoint change at 10 minutes following LOSCM. In practice, the ISCM setpoint transfer has been performed manually immediately following the completion of the manual RCP trip and is generally completed within the first minute following a LOSCM. With the installation of the ICCMS this time critical action will be performed automatically. The ICCMS will monitor RCS pressure, incore thermocouple temperatures and reactor trip status. If RCS SCM is lost, concurrent with a reactor trip, the system will automatically select the EFIC ISCM setpoint within ten minutes of LOSCM.

As discussed in response 1 above, the automatic actuation logic of the ICCMS, including the ISCM setpoint transfer function, will be covered in the pre-startup training program. The Loss of Subcooling Margin procedure will be revised to include specific guidance for ensuring the automatic ISCM setpoint transfer occurs within the allotted time. Each verification step will include appropriate contingency actions to be followed if the automatic setpoint transfer fails. Contingency actions will be validated to ensure applicable technical and timing requirements are satisfied.

3. **During a LOSCM event the existing EOP requires the operator to manually verify adequate HPI flow. This action will be automated with new ICCMS and SPDS HPI Flow monitoring functions.**

   During a LOSCM event, the existing CR-3 EOP directs the operator to verify adequate total HPI flow by (1) manually adding individual injection line flows, and (2) comparing the total HPI flow rate to a minimum required flow curve based on RCS pressure. To support EPU the ICCMS and Safety Parameter Display System (SDPS) will independently perform the HPI flow monitoring function automatically. The ICCMS will display HPI flow margin on two new safety related MCB displays. The SPDS will display total HPI flow on a graphic display that shows the minimum required HPI flow based on RCS pressure. If HPI total flow falls below the minimum required flow, SPDS will provide distinctive audible and visual alarms. The existing manual method will be retained in the procedure as contingency guidance for a multiple monitoring system failure. This change results in a reduction in overall operator burden and improves operator awareness by providing continuous real-time flow monitoring capability.

4. **During a LOSCM event the Loss of Subcooling Margin (LOSCM) procedure will require the operator to terminate any fluid additions to the Makeup and Purification System from the Boric Acid Storage Tanks, the RC Bleed Tanks or Demineralized Water System.**

   During a LOSCM event control room operators are required to immediately transition to the applicable Emergency Operating Procedure. If the LOSCM event were to occur while a fluid addition to the Makeup and Purification System (MU&P) was in progress, that addition could result in non-conservative RCS boron precipitation impact. To ensure RCS boron concentrations remain with the analysis assumptions, EOP guidance will direct the operator to terminate any ongoing fluid addition to the MU&P system during the LOSCM event.

5. **During a LOSCM event concurrent with inadequate HPI flow, the existing EOP directs the operator to manually perform a rapid primary system cooldown via SG pressure reduction. This action is being automated with the installation of the ICCMS and FCS. For EPU FCS actuation is required within 10 minutes of losing SCM if HPI flow is inadequate.**

   During a LOSCM event, if HPI flow is determined to be below the minimum required flow limit see discussion in Item 3 above), the existing CR-3 EOP directs the operator to manually perform a rapid primary system cooldown by depressurizing both steam generators. This action is being automated to support EPU.

The analyses supporting operation at EPU conditions credits the use of primary to secondary heat transfer via steam generator blowdown to augment HPI flow for certain design basis SBLOCA events. The initial steam generator blowdown for events involving a LOSCM and inadequate HPI flow must be initiated within ten minutes of losing SCM. A new Fast Cooldown System (FCS) will be actuated automatically by a new Inadequate Core Cooling Mitigation System (ICCMS) to perform the steam generator blowdown. When actuated by the ICCMS, the FCS system will establish an alternate Atmospheric Dump Valve (ADV) control setpoint ($\leq$ 350 psig). Steam generator pressure will be reduced and maintained at this control point.

The Loss of Subcooling Margin procedure will be revised to include specific guidance for verifying (ensuring) ICCMS/FCS automatic actuation by monitoring SG pressure response. The verification steps will include appropriate contingency actions to be followed for automatic actuation system failures.

Because the FCS actuation relies on a safety grade automatic actuation system, timely actuation is assured and a significant reduction in operator burden is realized. The revised procedure, including contingency actions, will be simulator validated prior to use using diverse scenarios and different licensed control room personnel to verify technical response and timing requirements are satisfied.

6. **The criteria for terminating High Pressure Injection (HPI) flow following a LOSCM event is being revised due to the addition of LPI crosstie piping.**

   HPI termination criteria are being revised due to the addition of new low pressure injection crosstie piping located downstream of the existing LPI flow instrumentation. The existing EOP guidance requires the operator to verify $\geq$ 1400 gpm LPI flow in each LPI train prior to terminating HPI flow. Because the existing LPI lines are not cross tied this criterion assures adequate ECCS flow even if one LPI line is completely severed.

   The LPI lines are being cross tied for EPU, therefore, the HPI termination criteria during transition from BWST to RB Sump in the EOP will be revised. The operators will be allowed in the EOP to terminate HPI flow following a LOSCM event provided (1) both trains of LPI are in operation, (2) LPI flow in both trains is greater than 2192 gpm (uncorrected), and (3) LPI line A to B differential pressure is less than 80 psid (uncorrected). With these criteria satisfied, core cooling is assured even for events involving one severed injection line. Injection line differential pressure instrumentation will be installed on the main control panel to provide the information necessary to properly evaluate the HPI termination criteria. For all other conditions, HPI termination will be directed from the Technical Support Center (TSC). Specific TSC guidance will be developed for HPI termination during conditions other than those covered by the EOP.

7. **RCP Recovery guidance will be revised to address RCP recovery following a LOSCM and automatic actuation of the Inadequate Core Cooling Mitigation System (ICCMS). The revised guidance will require the ICCMS actuation logic to be reset prior to RCP restart.**

   The operator actions for RCP recovery following a LOSCM (which are not time critical) will be revised to require ICCMS actuation logic to be reset prior to RCP restart. This is necessary because the ICCMS train “A” and “B” actuation signals seal in following actuation. When subcooling margin is restored these signals must be manually reset to restore control to associated components. The ICCMS reset control switches will be located on the main control board front panel.

8. **During a Station Blackout event operator actions will be revised to (1) require the operator to "Bypass" the FCS if RCS subcooling margin is lost, and (2) eliminate the local operator action to align a back up air supply to the ADVs.**

   The approved EOP mitigation strategy for a LOSCM during a SBO event is to immediately establish a maximum possible primary system cooldown using two ADVs and full Emergency Feedwater flow. The FCS will actuate when SCM is lost and after a specified time delay will rapidly reduce OTSG pressure to $\leq$ 350 psig and maintain it at that pressure. To comply with the requirement to immediately perform the rapid plant cooldown by performing a complete SG depressurization the operator will be required to select the two FCS control switches to the "BYPASS" position. This action will prevent the FCS from controlling the ADVs at the 350 psig setpoint. With FCS bypassed, normal ADV control using the MCB EFIC control stations will be maintained and the operators can perform the required cooldown in accordance with existing procedure guidance.

   Concerning the change to eliminate alignment of backup air to the ADVs, the addition of a new safety related ADV backup air system eliminates the need to manually align a backup source of air to the ADVs.

9. **The operator actions for Boron Precipitation Control will be revised to direct the operator to use the new hot leg injection system for boron precipitation control.**

   This change will proceduralize the use of the new hot leg injection flow path for boron precipitation mitigation. Boron precipitation control is established during the transition from BWST to RB sump recirculation phase of LOCA response. Previous methods required the use of a lengthy enclosure involving both control room and local operator actions. The revised method will only require the opening of either of the two new boron precipitation motor operated valves (MOVs). MOV control switches and position indicating lights will be located on the main control board. This change simplifies a previously complex task and results in a significant reduction in operator burden. This action will no longer depend on calculations based on results of a boronometer indication or time but rather the condition of sump swapover. For certain smaller LOCAs, where RCS pressure remains above the shutoff head of the LPI pumps following transition to the RB sump, operators will limit RCS depressurization and cooldown that could cause boron precipitation until adequate HLI flow has flushed the core region.

10. **The RCS Boration procedure will be revised to provide the operator with the option to utilize the new MUT Bypass valve MUV-661 for establishing RCS boration**.

    The MUT bypass valve modification improves RCS boration response time by diverting Letdown flow from the MUT inlet to the MUT outlet piping. With the valve selected to the bypass position, concentrated boric acid is injected into the letdown flow path and enters directly into the makeup pump suction header rather than the MUT. The control switch for MUT bypass valve will be located on the same main control board panel as existing boric acid injection valves, and will require no new diagnostics, skills or abilities for use.

11. **The Fire Protection procedure will require the operators to select the new ADV override switches to the override position upon confirmation of a fire in the control complex.**

A fire in portions of the CR-3 control complex could result in an inadvertent opening of the Atmospheric Dump Valves (ADVs). The impact of the increased capacity of the ADVs for EPU on an inadvertent opening would reduce OTSG pressures resulting in an unacceptable RCS cooldown. A new ADV override feature is being installed to prevent an inadvertent opening of the ADVs during a control complex fire. This feature will activate remote lockout devices that block all control signals to the ADV positioners when selected to the override position.

The ADV override feature will be actuated by manual control switch(s) located just inside the main control room door. This location was selected because it is in the direct egress path for control room evacuation at a location where emergency lighting is already installed. The operator action required to actuate the override is to simply select the override control switch(s) to the override position. With the ADV override actuated, the ADVs cannot spuriously open and will remain closed. Post trip SG pressure will be controlled by Turbine Bypass Valves (if available) or by the MSSVs. Automatic and manual control capability for the ADVs can be restored by (1) ensuring the control room override switch(s) are in the "Normal" position and manually resetting the lockout devices, or (2) transferring plant control to the Remote Shutdown Panel and manually resetting the lockout devices.

The ADV override function is actuated upon entry into the Fire Protection procedure. Resetting the ADV override function is not time critical.

12. **The Shutdown From Outside the Control Room procedure will require the operators to (1) ensure the new ADV override switches are selected to the override position prior to evacuating the main control room, and (2) restore the ADV control capability following transfer to the Remote Shutdown Panel (RSP).**

In the normal sequence of events the ADV override feature will be actuated as part of the Fire Protection procedure implementation for fires within the control complex. In the unlikely event that immediate control room evacuation is required, the Shutdown From Outside the Control Room procedure will require the operator to "Ensure" the ADV override feature is actuated during the control room evacuation. The ADV override control switch(s) will be located immediately adjacent to the main control room egress point. Actuating the ADV override function will have an insignificant impact on timely control room evacuation.

The procedure will direct the operator to reestablish the ADV control function following the transfer of control to the RSP. This will be accomplished by selecting the ADV override lockout devices to the RESET position. Restoration of ADV control is not time critical, however it is required to initiate a RCS cooldown from the RSP. The Appendix R cooldown analysis assumes a nine hour hold period before commencing an RCS cooldown. If ADV control from the RSP is not available, the cooldown can performed by controlling the ADVs with local hand wheels.

13. **Operator actions associated with a Plant Runback will be revised to (1) eliminate actions associated with an automatic asymmetric control rod runback, and (2) establish revised Integrated Control System (ICS) automatic runback rates and limits.**

Concerning item (1), the automatic ICS asymmetric rod runback is being eliminated. A rapid power reduction following a dropped control rod is not required and often results in a more severe transient than the dropped rod itself. Operator guidance for asymmetric control rod response will be proceduralized to initiate a controlled power reduction. Other procedure actions associated with a misaligned rod will remain unchanged.

Concerning item (2), ICS runback rates and limits are stated in terms of a percentage of the plants “maximum continuous rating” (maximum electrical output) at 100% power. Because the EPU results in a significant increase in electrical output, ICS runback rates and limits will be rescaled. The impact on operator response will be to ensure the automatic runback reduces power to the newly defined post runback target values which will be clearly stated in the Plant Runback procedure.

Question 3: Changes to Control Room Controls, Displays and Alarms

Describe any changes the proposed EPU will have on the operator interfaces for control room controls, displays, and alarms. For example, what zone markings (e.g. normal, marginal, and out-of-tolerance ranges) on meters will change? What set points will change? How will the operators know of the change? Describe any controls, displays, alarms that will be upgraded from analog to digital instruments as a result of the proposed EPU and how operators will be tested to determine they could use the instruments reliably.

Question 3 CR-3 Response:

To familiarize operators with EPU related changes a comprehensive training program will be provided to active operations personnel on the entire scope of EPU modifications. A preliminary training scope assessment is provided in the response to Question 5 below. EPU related training will involve both classroom presentations and simulator training exercises and will include appropriate evaluation. Wherever possible, the CR-3 simulator will be used to demonstrate the operation of new control room controls, alarms and setpoints and to train the operators on anticipated changes in overall plant response.

1. Changes to Control Room Controls

    a) Two new three position (Bypass, Auto, Actuate) FCS actuation selector switches will be added to the main control board. In the “Bypass” position the FCS automatic actuation will be blocked. When selected to the “Auto” position the FCS will be armed for automatic actuation. The “Actuate” position will manually actuate the FCS logic. When actuated, the FCS will reduce the ADV control set point to $\leq$ 350 psig to facilitate enhanced primary to secondary heat transfer. Indicator lights on each FCS control switch will illuminate when the FCS circuitry is actuated. Existing EFIC SG Pressure indicators will provide positive indication that the FCS is performing its design function.

    b) To support installation of the FCS actuation selector switches, the main control board switches for two feedwater block valves normally used to provide main feedwater to the SG high nozzles are being removed. These two valves are normally de-energized and are only used to support activities during plant outages. The valves will remain operable from their local control switches. Position indication on the main control board will be relocated near other feedwater isolation valve indications.

    c) Two new two position selector switches will be added to the main control board to reset Inadequate Core Cooling Mitigation System channel “A” and “B” actuation logic following an automatic actuation and recovery of adequate SCM.

    d) Two new main control board switches will added to allow the operator to select ICCMS

RCS subcooling margin displays to SCM based on either T_{HOT} or T_{INCORE}. ICCMS automatic actuation features are based only on the T_{INCORE} input.

e) A new ICCMS monitoring panel “lamp check” pushbutton will be added to the main control board.

f) The function of two valve control switches on the engineered safety feature (ESF) section of the main control board will change. MOVs DHV-210 and DHV-211 are being removed from the LPI flow path. The DHV-210 and DHV-211 control switches are being reassigned to operate the new hot leg injection valves DHV-514 and DHV-614. Valve escutcheon plates and engineered safeguards (ES) status board mimics will be altered to reflect the correct flow path.

g) The new Makeup Tank Bypass valve (MUV-661) will be installed on the main control board in the location formerly occupied by the MUT vent valve. The MUT vent valve will be relocated to the location formerly occupied by the MUT nitrogen supply valve, and the MUT nitrogen supply valve control switch will be eliminated. MUT nitrogen is only required during outages and will be operated locally.

h) Open/Closed valve position indication from the condensate pump recirculation valves will be added to the main control board.

i) Two new two position (Auto/Open) selector switches, with position indication, will be added to the main control board to provide control of EFWP recirculation valves. Valve control switches will be located near the existing EFP control switches.

j) Two new ADV Override switches will be installed in the main control room. When selected to the override position, the switches will actuate remote lockout relays which, when actuated, will remove all control signals to the ADVs and prevent inadvertent opening due to fire. ADV control from the Remote Shutdown Panel (RSP) will be restored as part of the RSP transfer process which includes manual reset of lockout relays..

2. Changes to Control Room Displays

a) New instruments

- The FCS modification includes the addition of dedicated SG pressure transmitters ranged 300 to 1100 psig for SG pressure control during a FCS actuation. The output of these new transmitters will be fed to new analog controllers and to the plant computer and SPDS for monitoring.
- New subcooling margin monitoring capability, independent of the SPDS, will be installed on the main control board. The two existing control room RCS subcooling monitors currently display the SPDS calculated SCM. These monitors will be replaced with Regulatory Guide 1.97 Category 1 displays driven by Inadequate Core Cooling Mitigation System instrumentation. The SPDS SCM indication and alarm features will remain available on the SPDS displays.
- New HPI total flow margin displays will be installed on the main control board. Two new HPI flow monitors will monitor HPI total flow and will provide indication of HPI

flow margin relative to the RCS pressure based minimum required HPI flow. A negative flow margin indicates inadequate HPI flow. These monitors will be Regulatory Guide 1.97 Category 1 displays driven by Inadequate Core Cooling Mitigation System (ICCMS) instrumentation.

- Low Pressure Injection line A to Line B differential pressure indicators (2) will be installed on the main control board and will be Regulatory Guide1.97 Category 2, Type D instruments. These indicators will be used in conjunction with LPI flow indication to evaluate HPI termination criteria in the EOP.
- A Low Pressure Injection line pressure instrument will be installed on each injection line. These instruments will input LPI line differential pressure to RB pressure. This information is sent to the SPDS alpha page displays and will be used by Technical Support Center staff, in conjunction with LPI flow rates, ECCS component status, and other tools to evaluate HPI termination criteria for conditions other than those addressed by the EOP. LPI Line pressure instrumentation will be qualified as Regulatory Guide 1.97 Category 2, Type D.
- A new Inadequate Core Cooling Mitigation System (ICCMS) monitoring panel will be installed on the main control board. This panel will provide important status and supervisory information on each of the ICCMS initiate/actuation channels.
- Two new RCS transmitters, one wide range (0 – 2500 psig) and one low range (0 – 600 psig) are being added to feed ICCMS System Channel 3 initiation logic. This instrumentation will also be fed to the plant computer for monitoring.
- Four new HPI low range flow transmitters (0 – 200 gpm) are being added to feed ICCMS channel 3 initiation logic. This instrumentation will also be fed to the plant computer for monitoring.
- Eight existing incore detectors and circuits will be upgraded to same range and qualifications as the incore detectors feeding ICCMS channels 1 and 2. These incore detectors are being upgraded to feed ICCMS channel 3 initiation logic. These instruments will also be fed to the plant computer for monitoring.
- Three existing two pen recorders for RCS void trending, reactor vessel level and hot leg level will be replaced with two new three pen recorders. The recorders will be relocated slightly to support installation of ICCMS controls and indicators.

b) Rescaled instruments

- ICS unit load demand (ULD)
- ICS automated unit load demand (AULD)
- Main Feedwater flow instrumentation
- Condensate flow instrumentation

c) Revised banding - None

d) Deleted Instruments

- Existing Decay Heat crosstie flow indicator (DH-38-FI) will be deleted from the main control board. The crosstie flow indication will be available in the control room as a plant computer variable. The existing DH crosstie flowpath is normally isolated during power operation.
- EFP-2 redundant flow indicator (EF-62-FI) is being deleted from the main control board redundant instrument panel.

3. Changes to Annunciator Alarms

   a) New Alarms

- "Condensate Valve A/B Trouble" alarms will indicate low instrument air supply pressure to the new CD control valve actuators.
- "Condensate Pump A/B Trouble" alarms will be added to indicate a low CD loop flow condition and alert the operator to a CDP, CD pump recirculation valve, or CD flow control valve problem.
- "High Deaerator Flow" alarm will be added to indicate excessive CD flow to the deaerator (inadequate deaerator bypass flow).
- A "RCP Trip/EFIC ISCM Level Initiation" alarm will be added to indicate that the RCP Trip and/or EFIC ISCM Level Initiation function has actuated on any of the ICCMS channels. The alarm will consist of a single Annunciator window with individual event points from each of the three ICCMS channels.
- A "Fast Cooldown System Initiation" alarm will be added to indicate that the Fast Cooldown System Initiation function has actuated on any of the ICCMS channels. The alarm will consist of a single Annunciator window with individual event points from each of the three ICCMS channels.
- A "Fast Cooldown System Bypass" alarm will be added to indicate that the Fast Cooldown System automatic actuation has been inhibited by the control board selector switch selected to the "bypass" position.
- A "Loss of Subcooling Margin" alarm will be added to indicate that Loss of Subcooling Margin has been sensed by any of the ICCMS channels. The alarm will consist of a single Annunciator window with individual event points from each of the three ICCMS channels.
- A "HPI Flow Margin Low" alarm will be added to indicate that a low HPI flow margin has been sensed by any of the ICCMS channels. The alarm will consist of a single Annunciator window with individual event points from each of the three ICCMS channels.
- A "ICCMS Channel Bypassed" alarm will be added to indicate that any of the ICCMS initiate and/or actuate channel functions have been placed in "Bypass". The alarm will consist of a single Annunciator window with individual event point inputs from each of the ICCMS functions that have bypass capability.
- A "ICCMS Trouble" alarm will be added to indicate that supervisory functions within

the ICCMS online monitor sense a problem. The alarm will consist of a single Annunciator window with individual event point inputs from the ICCMS online monitoring system functions.

- A "RCP Trip/EFIC ISCM Level Actuated" alarm will be added to indicate that the RCP Trip/EFIC ISCM Level Initiation function has actuated on two or more ICCMS channels and that the RCP Trip/EFIC ISCM Level selection functions have actuated.
- A "Fast Cooldown System Actuated" alarm will be added to indicate that the Fast Cooldown System Initiation function has actuated on two or more ICCMS channels and that the Fast Cooldown System has actuated.
- "EFP Recirculation Valve Out of Position" alarm event points will be added to the existing "Out of Service" alarm windows for EFP-2 and EFP-3.
- A "ADV Trouble" alarm will be added to indicate abnormal conditions in the ADV DC power system, valve controllers or override lockout devices. The alarm will consist of a single Annunciator window with multiple event point inputs.

b) Deleted Alarms

- "CDP decoupled" alarm and event points will be deleted. New CDPs do not utilize magnetic couplings; flow is controlled with the new AOVs.
- The "Asymmetric CRD Runback" alarm window and event point will be deleted. The ICS Asymmetric Rod Runback is being eliminated.

c) Revised alarm setpoints:

- Main turbine supervisory alarm setpoints will be revised based on vendor recommendations and post modification test results.

4. Changes to plant computer alarms and functions

a) New Alarm Functions: None

b) New Plant Process Computer (PPC) functions/capabilities:

- Open/Closed position indication from the condensate pump recirculation valves will be added to the PPC
- Individual CDP loop flows will be added to the PPC
- A and B SG pressure feeding the FCS will be added to the PPC
- HPI flow from four new narrow range flow instruments will be added to the PPC
- RCS Wide range pressure instrumentation feeding ICCMS channel 3 will be added to the PPC.
- RCS low range pressure instrumentation feeding ICCMS channel 3 will be added to the PPC.
- LPI line differential pressure instrumentation (2) will be added to the PPC

- LPI line A pressure instrumentation will be added to the PPC
- LPI line B pressure instrumentation will be added to the PPC

5. Changes to significant actuation or control setpoints

   a) New setpoints:

   - The new Inadequate Core Cooling Mitigation System (ICCMS) will initiate (1) an automatic trip of all Reactor Coolant Pumps, and (2) automatic selection of the EFIC inadequate subcooling margin control setpoint upon sensing a reactor trip and LOSCM for a specified time delay.
   - The new Inadequate Core Cooling Mitigation System (ICCMS) will initiate a Fast Cooldown System (FCS) actuation upon sensing a reactor trip, LOSCM and inadequate HPI flow (Setpoint variable based on RCS pressure) for a specified time delay.
   - When actuated, the FCS will insert a new SG Pressure control setpoint of $\leq$ 350 psig for the ADVs.
   - Feedwater booster pump recirculation control valve setpoints will be change to assure a minimum flow of 6000 gpm through each running pump.
   - Main feedwater pump turbines will be modified to include a new high FW discharge pressure automatic trip at 1600 psig (uncorrected). The purpose of the trip is to protect the FW system from over pressure.

   b) Revised setpoints:

   - ICS Loss of main feedwater pump runback endpoint will be changed to 40% of maximum continuous rating (MCR).
   - ICS Loss of main feedwater booster pump runback endpoint will be changed to 40% of MCR.
   - ICS asymmetric rod runback will be eliminated.
   - ICS loss of RCP runback endpoint will be changed to 70% of maximum continuous rating (MCR).
   - ICS main steam header pressure setpoint will be changed to 915 psig.
   - ICS main steam header pressure bias applied following a Rx trip will be changed to 95 psig.
   - The ICS Tave setpoint will be changed to 582^{o}F.

6. Digital Instruments

   a) Main control board indication for (1) Subcooling Margin, (2) inadequate HPI flow, and (3) LPI line pressure and differential pressure will employ digital (LED) displays. Similar displays are currently used on the main control board for several parameters including RCS Tave and the existing RCS subcooling monitor displays. Use of these instruments will require no special training.

b) New data acquisition for indication or controls will be accomplished using digital technology.

Question 4: Changes on the Safety Parameter Display System

Describe any changes to the Safety Parameter Display System resulting from the proposed EPU. How will the operators know of the changes?

Question 4 CR-3 Response:

The SPDS will be revised to include a display showing actual total HPI flow and a "Minimum Required HPI Flow Limit" based on RCS pressure. If HPI flow falls below the minimum flow limit during a loss of SCM event, the SPDS display will alert the operator by actuating a distinctive audible and visual alarm which includes a visual timer showing total accumulated time since actuation of the inadequate HPI flow alarm. The HPI flow alarm function will actuate anytime the flow limit is violated coincident with a reactor trip and LOSCM, even if the HPI flow display is not selected. To provide redundancy, four additional HPI flow instruments will input into the SPDS.

The following new information will be displayed on the SPDS alphanumeric page display:

- HPI low range flow from four new ICCMS HPI flow transmitters
- Steam generator pressure from two new FCS SG pressure transmitters
- LPI line differential pressure from two new LPI line A to Line B differential pressure transmitters
- LPI line pressure from new LPI line pressure transmitters on each injection line
- RCS wide range pressure from new ICCMS channel 3 RCS wide pressure transmitter
- RCS low range pressure from new ICCMS channel 3 RCS low range pressure transmitter.

New SPDS features will be included in mandatory pre-startup training for active operations personnel. Classroom training will cover the physical changes to the SPDS, the reasons/purpose of the changes, and the expected operator actions associated with the new SPDS inadequate HPI flow alarms function. The new SPDS monitoring and alarm functions will be installed on the CR-3 simulator prior to pre-startup simulator training. Simulator training will include a demonstration exercise to familiarize Operators with the new SPDS features, and unannounced casualty scenarios to verify correct operator response to SPDS alarm functions.

In addition to the SPDS modifications described above, CR-3 is installing independent SCM and HPI flow margin monitoring capability on the main control board. The new monitors, which are part of the ICCMS modification, will be Regulatory Guide 1.97, Category 1 qualified and will provide primary main control board indication of RCS SCM and HPI flow margin. The SPDS will provide backup indication to be used in conjunction with the ICCMS and will be available if the ICCMS based instrumentation was lost.

Question 5: Changes to the Operator Training Program and the Control Room Simulator

Describe any changes to the operator training program and the plant referenced control room simulator resulting from the proposed EPU, and provide the implementation schedule for making

the changes.

Question 5 CR-3 Response:

The Operator training programs (Licensed, Non-Licensed, and Shift Technical Advisor (STA) at CR-3 employ the Systematic Approach to Training (SAT) process. The SAT process provides a structured approach for assessing and properly scoping training requirements, selecting appropriate training settings, preparing properly sequenced training materials, and implementing the required training.

Specific changes to the operator training program content will be identified during the analysis phase of the SAT process currently scheduled to begin in mid-2011. In general, the training will focus on Improved Technical Specification changes, procedure changes, EPU modifications, and anticipate changes to integrated plant response resulting from those modifications.

A preliminary training review has identified the following potential EPU training requirements:

1. Training on the startup and power ascension test plan and associated procedures.
2. Changes to plant technical specifications and FSAR
3. Training on changes plant operating procedures:
    a. Condensate System operation
    b. Feedwater System operation
    c. Plant Startup (including turbine startup)
    d. Plant Shutdown (including turbine shutdown)
    e. Power Operations (including deaerator bypass operation)
4. Training on new Surveillance Procedures:
    a. ICCMS surveillance requirements
    b. FCS surveillance requirements (including ADVs, ADV dedicated power, and ADV dedicated air systems)
5. Training on EPU reactor core design and physics testing requirements.
6. Training on EPU plant modifications with special emphasis on:
    a. Plant instrument, controls, and setpoint changes
    b. OTSG requalification for EPU (orifice plate adjustment)
    c. Emergency Feedwater system modifications (recirculation isolation valves) including control logic, MCB modifications, ANN alarms, technical specifications and surveillance requirements.
    d. Condensate System modifications including CDP, CDP recirculation valve operation, CD flow control valves, deaerator bypass, and CD reject (hotwell level control) modifications.
    e. Feedwater System modifications including new main feedwater and feedwater booster pumps, modifications to the pump recirculation valve operation, new MFWP high discharge pressure trip, FWV-14/15 modification, and new IP and HP feedwater heaters.
    f. New high-pressure and low-pressure turbines
    g. Fast Cooldown System (FCS) including new instrumentation, FCS logic and control, ADV modifications, ADV override controls, dedicated air and power systems, normal and emergency operation, technical specifications, and surveillance requirements
    h. MUT Bypass system
    i. SPDS modifications with emphasis on inadequate HPI flow monitoring and alarm functions

j. ICS modifications (new MCR, revised post trip header pressure bias, elimination of asymmetric rod runback, revised runback limits and endpoints, AULD modification)
k. Inadequate Core Cooling Mitigation System (ICCMS) including initiate and actuate channel functions, new instrumentation, ICCMS cabinet details (location, contents, indications and controls, power supplies), qualified main control board SCM and HPI Flow monitors, ICCMS monitoring panel, ICCMS bypass and reset capability, technical specifications, and surveillance requirements.
l. The new LPI crosstie/hot leg injection system and associated pressure monitoring instrumentation, control switches, technical specifications, and surveillance requirements.
m. The HPI throttle valve modification (basis and impact)

7. EOP changes associated with:
    a. Small Break LOCA response (Including ICCMS, Fast Cooldown System and EFW flow requirements)
    b. Core flood line break mitigation (demonstrate influence of LPI cross-tie)
    c. Large Break LOCA response including sump swap over
    d. Boron precipitation mitigation (use of new Hotleg Injection system)
    e. HPI termination criteria including the function, basis and use of the new injection line differential pressure instrumentation.
    f. RCP restart following a LOSCM event (requires ICCMS reset)
    g. CFT isolation criteria
    h. Station Blackout (control of FCS during LOSCM)
8. Abnormal Operating Procedure (AOP) changes associated with:
    a. Plant runbacks including revised ICS runback rates and endpoints and deletion of the Asymmetric Rod automatic runback.
    b. Revised AP guidance for asymmetric control rod response
    c. RCS boration (Option to use MUT Bypass)
    d. Fire Protection procedure guidance for ADV override operation during control complex fire.
    e. Shutdown From Outside the Control Room procedure guidance for control of the ADV override control feature (actuate and reset)

EPU training will be conducted in phases, and will be carefully scheduled to assure that all required training activities are complete before startup. Initially, licensed and non-licensed operator training programs will focus on a general overview of the EPU related modifications and associated procedure changes. Comprehensive training on the entire EPU modification scope will follow and will include both classroom and simulator training, with appropriate evaluation. Wherever possible, the CR-3 simulator will be used to demonstrate anticipated changes in overall plant response and to provide opportunities for hands on control during normal evolutions, selected transients and accident scenarios.

In addition to the training described above, Just In Time (JIT) startup training will be provided to the Operators prior to the EPU plant initial startup. JIT training will focus primarily on the startup and power ascension testing plan.

Simulator Changes:

As the EC design process progresses, plant modifications will be evaluated, in detail, to determine their full impact on simulator software models, control room panels, instructor facility schematics, and associated simulator control functions.

Changes to the simulator software models will be made using a separate software configuration so as not to interfere with ongoing non-EPU related operator training. Simulator modifications will be completed on a schedule established to support the operator training schedule for EPU related training. Changes to the simulator software configuration will be controlled in accordance with the established simulator change control process.

A preliminary review has identified the following potential EPU related simulator changes:

1. Update the Reactor Core model
2. Adjust OTSG orifice plate settings
3. LPI/DH System model changes:
   a. Delete DHV-210/211 from the DH Model
   b. Add Check Valves DHV-510/511 to the DH model
   c. Revise flow mimic on ES panel A and B for injection flowpath
   d. Add LPI crosstie and hotleg injection piping and valves to DH system model
   e. Tune LPI system to achieve proper LPI flow response
   f. Reassign DHV-210/211 control switches on ESF panel to operate injection valves DHV-514/614
   g. Add Hot Leg Injection position indication
   h. Install LPI line (A to B) differential pressure instrumentation (two strings) including main control board indication.
   i. Install LPI line pressure instrumentation (one per injection line, downstream DHV-500/600) with inputs to plant computer and SPDS Alpha pages
   j. Modify MCB mimics, I/F schematics and stylized control panel pages as required
3. Inadequate Core Cooling Monitor System (ICCMS):
   a. Simulate ICCMS functions for inadequate SCM and HPI flow margin determination
   b. Add four additional HPI low range four instruments
   c. Add new RCS wide and low range pressure instruments
   d. Revise incores that feed ICCMS channel 3
   e. Fabricate and install ICCMS monitoring panel on MCB
   f. Provide three channel ICCMS initiate logic and 2 channel ICCMS actuate logic for RCP trip function and EFIC ISCM setpoint transfer based on LOSCM
   g. Provide actuation logic for FCS actuation based on LOSCM and inadequate HPI flow
   h. Provide ICCMS based Annunciator alarm inputs
   i. Replace existing RCS subcooling monitors with new ICCMS SCM monitors
   j. Add control switches (1 per monitor) to select subcooling margin based on T_{INCORE} or T_{HOT}
   k. Add indicating lights (3 per monitor) to indicate subcooling margin display is showing (1) SCM based on $T_{INCORE,}$ $T_{HOT,}$ or (2) the display is showing degrees superheat.
   l. Add two new ICCMS HPI flow margin monitors
   m. Add ICCS actuation reset switches (2)
   n. Add ICCMS monitoring panel lamp check switch
   o. Modify RCS Void and HL/Vessel level recorders
   p. Modify MCB mimics, I/F schematics and stylized control panel pages as required
4. ADV/Fast Cooldown System model changes:
   a. Revise the ADV flow characteristics and controller response

   b. Revise the ADV air supply system (with appropriate local operator actions)
   c. Revise ADV logic and control circuitry to incorporate the FCS (including FCS dedicated power dependency)
   d. Add ADV override control switches, annunciator alarm points and control logic including a remote function to reset ADV override lockout relays.
   e. Remove feedwater valve FWV-34/35 C/S from the PSA control panel
   f. Add FWV-34/35 indicating lights to the ICS control panel
   g. Add two ADV rapid cooldown control switches with indicating lights to the Main Control Board (MCB) in the former FWV-34/35 locations
   h. Add dedicated FCS SG pressure instrumentation (2) (to FCS, PPC and SPDS)
   i. Provide FCS based Annunciator alarm inputs
   j. Modify MCB mimics, I/F schematics and stylized control panel pages as required
5. Make Up and Purification System model changes:
   a. Provide eight additional HPI low range flow transmitters
   b. Provide HPI low range inputs to ICCMS, SPDS and ICCMS
   c. Modify HPI throttle valve positions and tune system to achieve proper flow balance
   d. Add new MUT bypass system
   e. Modify MCB mimics, I/F schematics and stylized control panel pages as required
6. Condensate System model changes:
   a. Replace variable speed condensate pumps with two new fixed speed condensate pumps
   b. Provide air operated flow control valve and recirculation valve on each CD train
   c. Condensate reject system modification
   d. Modify control logic for condensate system
   e. Provide Deaerator bypass system and instructor controls
   f. Add MCB modifications for CDP replacement (Annunciator alarms, rescale flow instrumentation, recirculation valve indication, etc.)
   g. MCB mimics, I/F schematics and stylized control panel pages as required
7. Feedwater System model changes:
   a. Provide two new Main Feedwater Pumps
   b. Add new MFWP high discharge pressure trip to each MFWP
   c. Provide two new Feedwater Booster Pumps and lube oil system
   d. Modify MFWP/FWBP recirculation lines, valves, control settings, and/or orifices as required.
   e. Tune MFWP turbine response (if required)
   f. Provide new Feedwater Heaters FWHE-5A/B
   g. Provide new Feedwater Heaters FWHE-6A/B
   h. Provide oxygen sampling system downstream of FWHE 6A/B
   i. Rescale MFW flow instrumentation
   j. Modify Main Feedwater Pump suction valve (stroke time)
   k. Tune extraction steam flow to deaerator and FWHE-5A/B and 6A/B (as required)
   l. Modify MCB mimics, I/F schematics and stylized control panel pages as required
8. EFW System model changes:
   a. :Add recirculation isolation valves and control logic to EFP-2 and EFP-3, including new valve control switches and applicable annunciator alarm event points.
   b. Provide EFIC LOSCM setpoint transfer based on ICCMS initiate logic
   c. Delete redundant EFW flow instrument (EF-62-FI) from MCB
   d. Tune EFW flow response per design/test data

   e. Modify MCB mimics and I/F schematics and stylized control panel pages
9. Add new high pressure and low pressure turbines
   a) Revise high pressure turbine control valve program for full arc emission
10. Safety Parameters Display System (SPDS) modifications
    a. Upgrade Simulator SPDS with revised plant software package (SPDS computers on the CR-3 simulator uses the same software package as the actual plant)
11. Rescale the Integrated Control System
    a. Rescale entire ICS based on new maximum continuous rating (MCR) and new MFWP performance curves.
    b. Modify AULD for operation at EPU power
    c. Revise ICS runback rates and endpoints for Loss of RCP and FWPs
    d. Delete the Asymmetric CRD runback
    e. Revise turbine header pressure bias for post RX trip conditions

Summary

The EPU will result in changes to the CR-3 training programs and to the CR-3 plant referenced simulator. The SAT process will be used to properly evaluate training needs and implement required training on an appropriate schedule. The CR-3 simulator will be modified, as necessary, and tested sufficiently early to support procedure validation and implementation of the operator training schedule.

Results

The results of the EPU Human Factors review conclude that the identified changes will not alter the basic mitigation strategies with which the operators are familiar. The most significant change, the addition of the new ICCMS/FCS, automates time critical actions that were previously performed manually. This results in a significant reduction in operator burden. Compensatory actions for ICCMS/FCS failures involve simple, and familiar diagnostic steps (response to an SPDS alarm function) and are accomplished by the manipulation of manual control switches located in a common location on the MCB. Changes associated with instrument scaling and setpoints revisions are minor and do not involve a level of complexity that could lead to misunderstanding the parameter. Operator training programs will be revised in accordance with the SAT process and appropriate training will be provided on all plant modifications, administrative/technical requirement changes, and procedure revisions. The CR-3 simulator will be updated and tested in sufficient time to provide effective reinforcement of procedure and plant physical changes as well as build proficiency with the required operator action changes.

2.11.1.3 Conclusion

CR-3 has reviewed the changes to operator actions, human-system interfaces, and procedures required for the proposed EPU and concludes that the evaluation (1) has appropriately accounted for the effects of the proposed EPU on operator actions and training, (2) provides reasonable assurance that operator performance will not be adversely affected by the changes made to implement the proposed EPU, and (3) has properly identified training program and simulator modifications resulting from the EPU. Further, CR-3 will continue to meet the requirements of FSAR Section 1.4.11, 10 CFR 50.120, and 10 CFR 55 following implementation of the proposed EPU. Therefore, CR-3 finds the proposed EPU acceptable with respect to the Human Factors aspects of the required system changes.

2.11.1.4 References

None

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

2.12.1.1 Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The CR-3 review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed EPU thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations.

The NRC's acceptance criteria for the Approach to EPU Power Level and Test Plan are based on:

- 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service.

CR-3 Current Licensing Basis

The initial startup test program at CR-3 is described in FSAR Chapter 13. The test program was developed to ensure the safe and efficient operation of the plant up to its initial rating of 2452 MWt. All activities established in the initial startup test program were structured in accordance with NRC Regulatory Guide 1.68, and complied with the provisions of 10 CFR 50, Appendix B, Criterion XI, Test Control.

2.12.1.2 Technical Evaluation

Introduction

Since initial startup testing, CR-3 has increased the licensed rated thermal power (RTP) limit several times. Table 2.12.1-1 lists the initial licensed RTP limit, the license amendments allowing the previous power uprates, the resulting change in RTP, and the type of uprate. At each of these new power levels, plant operating parameters were verified to be in accordance with predicted analyses and design documentation. This verification relied on using standard test procedures and plant surveillance testing procedures.

CR-3 is proposing an EPU to increase core thermal power from 2609 MWt (referred to as Measurement Uncertainty Recapture [MUR] throughout) to 3014 MWt (throughout the balance of this section "RTP" refers to 3014 MWt unless otherwise noted). To accomplish this uprate CR-3 will install several modifications (refer to Appendix E) to support operation at a higher reactor thermal power limit with proportionally higher main steam and feedwater flows. As a result of these changes, a rigorous test program is required to ensure plant SSCs operate safely during plant heatup, startup, and power ascension to the EPU power limit.

2.12.1.2.1 Comparison of Proposed EPU Test Program to the Initial Startup Test Program

CR-3 has reviewed the initial plant startup test plan described in FSAR Chapter 13, Initial Tests and Operation, and the initial "Startup Report." The initial startup test program is summarized in the following FSAR Tables:

- Table 13-1, Prefueling Test Summary
- Table 13-2, Post Fueling – Precriticality Test Summary
- Table 13-3, Zero and Low Power Test Summary
- Table 13-4, Power Test Summary

FSAR Table 13-1, Prefueling Test Summary, describes the basic objectives of the preoperational test program that was performed prior to initial fuel loading. The objective of the Prefueling test program was to confirm that plant SSCs were capable of performing their designed function. The test objectives described on FSAR Table 13-1 were reviewed and it was determined that only where modifications are required for EPU operation could some of the results obtained during initial testing potentially be invalidated. Post modification and power ascension test programs will ensure the affected preoperational test objectives will be satisfied. Testing will be performed during post modification testing or integrated into the startup and power ascension test program in a logical sequence, and may not be completed prior to fuel loading. The potentially affected tests, test objectives, and planned EPU testing are shown in Table 2.12.1-.2.

As shown in Table 2.12.1-3, CR-3 is proposing the elimination of several tests performed during the initial startup test program. Justifications for the elimination of non-transient tests are included in Table 2.12.1-3. Justifications for the elimination of selected transient tests are provided in this section. The elimination of several transient tests is based, in part, on the results of analytical testing performed by AREVA NP. The analytical methodology used for this testing is also described in this section.

2.12.1.2.2 Post Modification Testing Requirements

In accordance with the CR-3 Engineering Change (EC) process, post modification functional testing may be implemented by several different methods including:

- Existing approved plant procedures
- Specially written EC Functional Test procedures
- Work order instructions
- Specific step-by step instructions in the EC package

The method selected will be based on the scope, complexity, and safety significance of the specific modification. Procedural controls covering preparation, approval, and performance are provided for each method.

Post modification testing is an integral part of the EC process. Table 2.12.1-4, R17 Plant Modification Testing, provides a list and brief description of significant plant modifications scheduled for installation during R17, and a preliminary description of planned post modification and startup testing. Additional post maintenance testing and startup requirements will be identified as a function of the design process and final test procedures will be prepared and approved prior to EC implementation.

A number of EPU related modifications were installed during previous plant outages and will have been in service for one or more fuel cycles before being operated at EPU rated power. Post modification testing for these components was completed following initial installation. However, additional monitoring will be imposed to ensure these modifications perform as designed as the plant approaches EPU rated power. Table 2.12.1-5, Additional Testing for Previously Installed EPU Modification, provides a brief description of each applicable pre-R17 modification and a list of planned activities to ensure satisfactory performance.

2.12.1.2.3 Justification for Eliminations of Transient Testing

CR-3 has reviewed the scope of testing performed during the initial plant startup (as described in FSAR Chapter 13, and the initial “Startup Report”). In accordance with NUREG-0800, Section 14.2.1-III.C, CR-3 is proposing elimination of certain transient tests performed during initial plant startup. This request is based on reviewing: (1) the experience gained during the conduct of the initial startup test program, (2) over thirty years of commercial operational experience which has included several actual plant transients similar in scope to those included in the initial test program, and (3) the results of an integrated plant transient performance analyses performed specifically to evaluate post EPU transient response during selected operational transients. Based on this review, CR-3 concludes that eight transient tests included in the initial startup test program should be excluded from the scope of EPU startup and power ascension testing. This section provides the justification for not performing the specified transient tests.

<u>Justification for Elimination</u>

CR-3 is proposing elimination of eight transient tests performed during the initial plant startup test program.

CR-3 has reviewed the scope and purpose of each of these tests and has determined that the test objectives for each have either been satisfied by means of operating experience, evaluation and other analytical methods, or are no longer applicable due to plant design changes. The results of these evaluations and analyses provide reasonable assurance that the fundamental operating characteristics of the uprated plant remain consistent with the operating characteristics prior to the uprate, and justify the proposed elimination of these tests from the EPU startup test program.

The transient analytical methodology used for these analyses are described later in this section. Specific justification for eliminating each test is provided below.

1. **Reactor Trip Test -** Initial startup testing included a reactor trip test from 40% power. The purpose of the test was to measure the plant response during and after a deliberate reactor trip from power. Major plant control system functions (Reactor Protection System (RPS), Integrated Control System (ICS), Control Rod Drive (CRD), Electro-hydraulic Control (EHC), Pressurizer (PZR) control) are not being modified to support EPU operation. TBV and ADV capacities have increased to support EPU but the increased capacity was not required to obtain a satisfactory post trip response. The applicable normal operational control algorithms for these components remain unchanged. CR-3 has experienced a number of reactor trips from full power and has confirmed that plant SSCs perform as designed. An assessment of reactor trip response from 100% EPU power (3014 MWt) has been performed by AREVA NP using the Digital Power Train (DPT) analytical computer code. The results of this test confirm no unexpected differences post reactor trip plant response following a reactor trip from EPU power level. Therefore, CR-3 concludes that the power uprate has not introduced any new

thermal-hydraulic phenomena or system interactions and does not invalidate the reactor trip test as originally performed.

2. **Reactor Coolant Pump Trip Test -** Integrated Control System testing during initial startup included a RCP trip from 40% power and a subsequent RCP restart at 20% power. The purpose of the test was to demonstrate proper Integrated Control System (ICS) response to asymmetric RCS loop flow rates. The ICS subsystem that controls FW flow ratioing has not been modified for EPU operation. The original startup test was conducted below the ICS runback limit for loss of a single RCP and therefore did not involve a verification of ICS runback capability. Subsequent to initial startup testing, CR-3 has successfully performed several RCP shutdown/startup operations at power without incident. To verify post EPU response to a RCP trip an analytical test was run from a power level of approximately 78% EPU power (approximately 2350 MWt) using the DPT analytical computer code. The 2350 MWt initial power was selected because it is approximately the highest thermal power level from which a RCP trip can occur and not result in a reactor trip at MUR. Because a RCP trip can challenge the OTSG high level limit setpoint due to FW flow reratioing, the DPT model was modified for this test to include a dynamic calculation of Operate Range level instrumentation. During ICS rescaling for EPU, the runback limit for the RCP trip will be set at 70% EPU power (2010 MWt), therefore the DPT test required a runback to 70% EPU power and FW flow reratioing to match FW loop flows to primary heat input. The results of this test identified no significant differences in plant response following a single RCP trip. Therefore, CR-3 concludes that the power uprate has not introduced any new thermal-hydraulic phenomena or system interactions and does not invalidate the RCP trip test as originally performed. (NOTE – this test was described under the *Unit Load Transient Test* in the initial “Startup Report.”)

3. **Main Feedwater Pump Trip Test** - ICS Testing during initial startup included a MFWP trip test from 75% power. The purpose of the test was to demonstrate proper ICS response and RCS pressure/pressurizer level control during a plant runback to 56% power. To support operation at EPU power level the MFWPs have been modified to produce a higher full power flow rate but the pump control system remains unchanged. During ICS rescaling for EPU, the runback limit for the MFWP trip will be reduced based on total FW flow capability during single pump operation (the revised runback limit will result in the same thermal power endpoint (approximately 1300 MWt) as the MUR limit). To verify post EPU response to a MFWP trip an analytical test was run from approximately 78% power (2348 MWt) using the DPT analytical computer code. The 2348 MWt power was selected because it is approximately the highest power level from which an MFWP trip can occur and not result in a Reactor trip. The results of this test confirm no significant differences in plant response following a MFWP trip from approximately 78% power. Therefore, CR-3 concludes that the power uprate has not introduced any new thermal-hydraulic phenomena or system interactions and does not invalidate the MFWP trip test as originally performed. (NOTE – this test was described under the *Unit Load Transient Test* in the initial “Startup Report.”)

4. **Loss of Offsite Power** – Initial Startup testing included a loss of offsite power (LOOP) test from 15% power. This test was designed to verify operation of station diesel generators, turbine driven emergency feedwater pump and Engineered Safeguard (ES) buses under LOOP conditions, and to confirm the ability to establish and maintain plant control on natural circulation. EDG and Emergency Feedwater (EFW) pump operability will be verified using existing plant surveillance procedures. The plant EFW Initiation and Control System (EFIC) functions have not been modified for EPU operation. Atmospheric dump valve (ADV) capacity has been increased for EPU, but the control logic used

during a LOOP event, without loss of subcooling margin, is unchanged. CR-3 has experienced several actual LOOP events from full power conditions which confirm the adequacy of the existing plant design. Sections 2.8.5.2.2 Loss of Nonemergency AC Power to the Station Auxiliaries, 2.8.5.2.3, Loss of Normal Feedwater Flow, and 2.8.5.3.1, Loss of Forced Reactor Coolant Flow, describe the design basis events associated with a LOOP (loss of non-safety AC, loss of MFW and Loss of RCS forced flow). Those analyses determined that all applicable acceptance criteria were met and that operation at EPU power did not adversely affect event results, impose any restrictions or require any plant modifications. To further verify post EPU response to a LOOP, an additional test was run from 100% EPU power level using the RELAP5/MOD2-B&W analytical computer code configured for “best estimate” analysis. The results of this test confirm no unexpected differences in post trip plant response. Therefore, the CR-3 concludes that the power uprate has not introduced any new thermal-hydraulic phenomena or system interaction and does not invalidate the LOOP trip test as originally performed.

5. **Dropped Control Rod Test** - Initial startup testing included a dropped control rod test from 75% power. The CR-3 “Startup Report” (Section 4.14) lists the following objectives for this test: (1) verify LHR and departure from nucleate boiling ratio (DNBR) within limits, (2) verify the ability of the Control Rod Drive (CRD) System to detect and properly indicate asymmetric rod and asymmetric fault conditions, (3) verify the ability of the ICS to perform an automatic runback to < 60% power, and (4) verify control rod out motion is inhibited above 60% power with an asymmetric fault present.

   Concerning objective 1, Section 2.8.5.4.3, Control Rod Mis-operation, describes a design basis dropped rod event analysis (dropped rod without ICS runback) performed from 100% EPU power. The analysis determined that all applicable acceptance criteria were met and that EPU did not adversely affect event results, impose any restrictions or require any plant modifications (i.e., did not introduce any new thermo-hydraulic phenomena or system interactions). To supplement the analytical results obtained above, two additional dropped control rod tests were performed using the DPT analytical computer code. The first test was run from 100% EPU power and included an automatic ICS runback to < 60% EPU power. The results of this test confirm no significant differences in plant response during or after the plant runback. A second test was run from 100% power with the asymmetric rod runback feature removed. This test indicated no unexpected or adverse operating effects resulting from the elimination of the asymmetric rod runback feature.

   Concerning objectives 2, and 4 there have been no changes made to the CRD System that would affect the ability of the systems to: (1) detect asymmetric rod and asymmetric fault conditions, or (2) inhibit control rod out motion.

   Concerning objective 3, the automatic ICS asymmetric rod runback feature is being deleted. Plant power reductions following an asymmetric rod event will be performed by control room personnel in accordance with approved procedures. Analytical testing with the DPT analytical computer code identified no unexpected or adverse operating effects resulting from the elimination of the automatic asymmetric rod runback feature. Therefore, CR-3 concludes that the power uprate has not introduced any new thermal-hydraulic phenomena or system interaction and does not invalidate the Dropped Rod Test as originally performed.

6. **Turbine Trip Tests -** Initial startup testing included turbine trip tests from 75% and 100% power. The purpose of these tests was to verify overall plant response during the post trip ICS runback to 15%

reactor power. Both tests were invalidated by the addition of a Reactor Protection System (RPS) anticipatory reactor trip based on a turbine trip at > 45% power. Section 2.8.5.2.1, Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure, describes a design basis turbine trip transient analysis from 100% EPU power. The analysis determined that all applicable acceptance criteria were met and that EPU did not adversely affect event results, impose any restrictions or require any plant modifications. In addition to the design basis event, an assessment of turbine trip response from 35% EPU power was performed using the DPT analytical computer code. The results of this test confirm no significant differences in plant response following the turbine trip. To provide additional assurance of acceptable plant transient response, CR-3 will perform a < 40% turbine trip test on its plant specific simulator following integration and testing of R17 component modifications and cycle 18 reactor fuel data. Finally, the power ascension test program will include a main turbine trip test from < 40% EPU power to confirm the results of the analysis and verify overall plant response during the ICS runback following a turbine trip.

7. **Unit Loss of Electrical Load** - Initial startup testing included a Unit Loss of Electrical load test (load rejection) from 100% power. The purpose of this test was: (1) to demonstrate the ability of the plants ICS to run the plant back to 15% reactor power while carrying house loads on the main generator via plant auxiliary transformer, and (2) to verify the loss of electrical load transient does not result in nuclear fuel damage or leakage. As described in FSAR section 14.1.2.8.5.2 the pressurizer PORV setpoint has been raised above the RPS high pressure reactor trip setpoint, therefore a load rejection from full power will result in a reactor trip on high Reactor Coolant System (RCS) pressure rather than a controlled runback to 15%. Concerning the aspect of fuel failure, the load rejection event from full power was analyzed for EPU power conditions and it was determined to be bounded by the turbine trip analysis. Turbine trip transients from 54% and 100% EPU power were analyzed and it was determined that all applicable acceptance criteria were met. Section 2.8.5.2.1, Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam Pressure Regulatory Failure, includes the results of this analysis and a description of the analytical methodology employed.

8. **Shutdown from Outside the Control Room** - Initial startup testing included a Shutdown from Outside the Control Room Test from 15% power. The test described in FSAR Table 13.4 and Section 4.12 of the Startup Report is no longer valid. In response to Appendix R, CR-3 has installed a Remote Shutdown System (RSS) which is described in FSAR Section 7.4.6. The purpose of the test is to demonstrate the capability of bringing the plant to a safe cold shutdown from outside of the main control room (MCR). Several components controlled from the remote shutdown panel were modified to provide greater steam dump and EFW flow capability; however, the actual component controls and indicators, provided on the Remote Shutdown Panel (RSP) have not been changed. As a result of larger ADV capacity a new ADV override feature has been added to prevent an uncontrolled cooldown resulting from a fire induced spurious ADV opening. Operators will ensure the ADV override is established as part of the control room evacuation procedure and will restore ADV control following the transfer of control to the RSP. The operator action to establish the ADV override during the control room evacuation is a simple feasible action, reflecting an insignificant impact on timely evacuation. Restoration of ADV control is not time critical (i.e., not required within the initial 8 hours). The ADV override is described in Appendix E and the operator actions are discussed in Section 2.11. Because the controls and indicators on the RSP have not been modified and the operator action to establish the ADV override reflects an insignificant impact on timely control room evacuation CR-3 concludes that the power uprate does not invalidate the testing performed following installation of the RSS. To provide additional assurance of acceptable plant response CR-3 will perform a remote

shutdown transient test on its plant specific simulator following integration and testing of EPU component modifications and reactor fuel cycle data. The purpose of the test will be to: (1) demonstrate the ability to establish plant control, and (2) confirm procedure sequence and timing is acceptable at EPU conditions. It will not include a cooldown to cold shutdown conditions.

Transient Analytical Methodology

As discussed above, CR-3 has reviewed the scope of the initial startup test program and is proposing elimination of several of the transient tests performed during the initial plant startup test program. Justification for the elimination of five of the transient tests described above is based, in part, on the results of a comprehensive transient analysis performed by AREVA NP. Those five tests are:

- Reactor Trip from 40% power
- Reactor Coolant Pump trip from 40% power
- Main Feedwater pump trip from 75% power
- Loss of Offsite Power (LOOP) from 15% power
- Dropped Control Rod from 75% power

To better assess overall plant response, three of the transient test scenarios listed above (reactor trip, LOOP and dropped control rod) were analyzed from 100% EPU power. The RCP and FWP trip tests were analytically run from the highest power level that did not result in a reactor trip.

Each of these transient tests have been run by AREVA NP using either RELAP5/MOD2-B&W or the Digital Power Train (DPT) simulation code, properly configured for the CR-3 EPU plant configuration and benchmarked against CR-3 response data.

RELAP5/MOD2-B&W

The thermal-hydraulic analysis of the Loss of Offsite Power (LOOP) transient at both MUR and EPU power conditions was performed with the RELAP5/MOD2-B&W (R5/M2) computer code. RELAP5/MOD2 is an advanced system analysis code developed by Idaho National Engineering Laboratory (NEL) to be used as a best-estimate analysis tool for light water reactor systems, and subsequently modified by AREVA. RELAP5/MOD2-B&W has been approved by the NRC for use in both LOCA and non-LOCA analysis. The code allows modeling of both the primary and secondary systems, as well as fill-systems such as Emergency Feedwater (EF) System. It has been extensively used in PWR safety analysis and other thermal-hydraulic analyses.

Digital Power Train

The remaining transient tests were run with AREVA NP's Digital Power Train (DPT) computer code. The DPT computer simulation tool was initially developed as “Power Train”, a real time hybrid computer simulation of the B&W lower loop nuclear power plant (reference B&W topical report BAW-10149 Revision 1, November 1981). DPT retains the same capability of Power Train, but has been migrated to a digital computer platform and renamed.

DPT is used to predict the performance and behavior of the major components in the B&W nuclear steam supply system for a wide range of plant conditions and operation. DPT was designed to model as much of the power plant as is feasible, including those components whose behavior influences integrated plant transient response. DPT is a system-level program simulation code that models the overall NSSS, including detailed modeling of the Integrated Control System (ICS), Reactor Protection System (RPS), makeup/pressurizer level control, RCS pressure controls, and over pressure protection (relief valves).

The DPT primary system is comprised of the reactor vessel, two Once Through Steam Generators (OTSGs), hot and cold leg piping, primary coolant pumps, the control and safety rod systems, and the pressurizer. Point reactor kinetics is used to model the dynamics of core power generation. The pressurizer is modeled as a non-homogeneous thermodynamic system including the effects of heaters, sprays, wall condensation and relief valves. Primary flow is modeled as a constant proportional to the reactor coolant pump status. Transient behavior of primary flow (RCP trips) is determined by flow versus time coastdown curves. All models can account for both forward and reverse flows. Thermal transport in the RCS piping is modeled with variable transport delay.

The original DPT secondary configuration was based upon the Rancho Seco nuclear plant (B&W 177 fuel assembly (FA) NSSS). Included in the secondary model are the steam lines connecting the steam generator to the turbine, turbine throttle valves, turbine bypass valves (TBVs), atmospheric dump valves (ADVs), and the turbine generator. The Feedwater and Condensate System includes the condenser, hotwell, condensate, and feedwater pumps, three low pressure heaters with a drain tank, three high pressure heaters with a drain tank, and feedwater control valves for each steam generator. The turbine extraction and feedwater heater drain mass flows are functions of throttle flow based on steady-state plant heat balance information.

DPT was benchmarked against turbine/reactor trip and MFWP trip events at the Oconee nuclear power plant, turbine trip with runback and loss of MFW flow events at the Three Mile Island nuclear power plant (both plants B&W 177 FA NSSS) and against a LOFW transient conducted at the Alliance Research Center OTSG test facility. The DPT model demonstrated good fidelity during benchmark testing.

To support the CR-3 transient test program, the DPT secondary plant configuration and associated control systems have been modified to better replicate the CR-3 plant design in both MUR and EPU configurations. These modifications include the following:

1. The Main Feedwater System was reconfigured to reflect the split Feedwater System used at CR-3. Modifications include the addition of a second high pressure feedwater heater train, a second main feedwater pump (MFWP) and associated control valves, main block valves, cross connection piping and an automatic FW crosstie valve.

2. The ICS control logic was modified to include (1) the CR-3 FW control logic, (2) FW temperature compensation to the FW demand signal, and (3) CR-3 ICS runback limits and rates.

3. The main feedwater pump models were revised to include the effects of pump cavitation.

4. Dynamic modeling of SG Operate Range level instrumentation (for RCP trip test only)

Limitations imposed by the DPT model were evaluated to ensure they would not adversely impact the analytical results obtained during the test runs. The following DPT limitations were identified:

- DPT primary coolant loops will not simulate void formation (except in the pressurizer volume)

  Impact – None. The transient test scenarios are not expected to void the primary coolant loops.

- Each primary loop is modeled with one cold leg containing two parallel RCPs

  Impact – None. The resolution provided by a single cold leg on each SG is sufficient to evaluate overall system response to any of the transient scenarios analyzed with DPT. For the RCP trip test, the simulator benchmark data had the T_{COLD} instrument selected to the cold leg with the running RCP to ensure a correct comparison with DPT. The LOOP transient response could be affected by the single cold leg configuration, therefore the LOOP transient was evaluated with RELAP5/MOD2-B&W.

- DPT models a single low pressure feedwater train to represent two cross coupled parallel trains of heaters and pumps (one main feedwater booster pump, one hot well pump, and one condensate pump, no deaerator)

  Impact – None. The simplified low pressure feedwater arrangement does not impact plant transient response during the test scenarios. To simulate a single Feedwater Booster Pump (FWBP) trip, the flow from the DPT booster pump was quickly reduced to match the maximum output of a single CR-3 booster pump. The DPT feedwater heaters have been tuned to provide an appropriate feedwater temperature profile without the need for deaerator heating.

- The DPT ICS model does not include the Automatic Unit Load Demand (AULD) feature (steady state control of MWt)

  Impact – Minimal. During large transients the AULD automatically disengages and transfers control to the Unit Load Demand (ULD). During the loss of FW Heater test the transient was gradual and the AULD would remain engaged. The DPT data shows a small reduction in reactor power that would likely be corrected by AULD action in the plant. This limitation had no impact on the overall test results.

- The DPT model does not support manual operator action to open PZR spray valve

  Impact – Minimal. Pressurizer spray was manually opened in the simulator benchmark runs for MFWP and RCP trip tests to get full spray flow (in automatic the CR-3 pressurizer spray valve only opens 40%) In the DPT model pressurizer spray valve opens fully in automatic.

- The DPT model does not include dynamic modeling of the OTSG level instrumentation. SG level inputs to the ICS are based on the calculated collapsed level in the SG downcomers.

  Impact – minimal. SG level is not a normal controlling input to the ICS during power operation above approximately 25% power, however it does provide a limiting function to limit FW demand if Operate Range level exceeds the high level limit setpoint . For the transients being evaluated only the RCP trip test potentially challenges the SG high level limit. Therefore, for the RCP trip test, the DPT model was updated to include a dynamic calculation of Operate Range level instrumentation.

The limitations of the DPT code identified above do not adversely affect the models fidelity for the transients analyzed. Therefore, the DPT code, as modified for the CR-3 transient testing program, is suitable to evaluate plant response to the five transients included in the transient test program.

To verify the fidelity of the reconfigured DPT model, it was formally benchmarked against the CR-3 plant referenced simulator at MUR conditions. The CR-3 simulator is verified to meet the fidelity requirements of ANSI-ANS-3.5 annually and it has been benchmarked against actual plant transients similar to those included in the test program. It has a proven track record of accurately replicating actual plant response during both normal operation and transient conditions. The reactor coolant system and steam generator models on the simulator were upgraded in 2009 with the THOR advanced thermal hydraulic models. The reactor neutronics model was updated with the S3R real-time engineering neutronics model based upon the Studsvik-Scandpower CASMO-SIMULATE engineering code. These models, when coupled to the mature balance of plant and control system models already on the simulator, provide an excellent tool for assessing plant transient response.

Each of the five transient test scenarios was run from MUR conditions on both DPT and the CR-3 simulator. Key parameters were identified and recorded during each run. DPT and CR-3 simulator test results were then compared. The benchmarking effort demonstrated that the reconfigured DPT model effectively replicates CR-3 response in MUR configuration.

Following benchmarking, the DPT model was reconfigured to EPU configuration. Plant modifications required for EPU operation were reviewed, and those that could impact transient response during the test scenarios were incorporated into the DPT model. Refer to Tables 2.12.1-4 and 2.12.1-5 for modifications modeled in the Transient Analyses.

Finally, each of the five transients were run with the plant in the EPU configuration and the results were compared and evaluated to the MUR test results.

<u>Supplemental Analytical Testing</u>

In addition to the five analytical transient tests described above, CR-3 performed four additional analytical test runs using the DPT code for transients not included in the scope of the original startup test program. These tests are summarized below.

- **Main Turbine Trip from 35% power –** The purpose of this test was to verify plant response to a main turbine trip below the Anticipatory Reactor Trip System (ARTS) setpoint. The objectives of the test were to:
  - Confirm the ability of the Integrated Control System to “track” back to approximately 25% power without a reactor trip
  - Verify proper control of key primary and secondary plant parameters during power reduction to 25% power
  - Verify header pressure control is transferred to TBVs and TBVs control properly
  - Verify the ICS runback is terminated before T_{AVE} ramps
  - Verify SGs control at low level limits

  The DPT test results indicate no unexpected system interactions or thermo-hydraulic phenomena. CR-3 will perform a main turbine trip test from < 40% power as part of the power ascension test program. Analytical results will be considered when developing the startup test procedure. DPT results will be benchmarked against actual plant data following plant startup testing.

- **Feedwater Booster Pump trip from highest survivable EPU power** – The objective of this test was to confirm the ability of the plant to survive a booster pump trip and runback from approximately 70% EPU power (approximately 2118 MWt). The 70% power value was selected because it is approximately the highest power level from which a FW booster pump trip can occur and not result in a reactor trip at MUR configuration. The runback endpoint was approximately the same thermal power as the MUR runback (approximately 1300 MWt). Specific objectives of the test are:

  - Verify a single booster pump continues to support the operation of two MFWPs

  - Verify automatic ICS runback capability to the MUR thermal power setpoint without a reactor trip

  The DPT test results suggest plant response to this event is improved at EPU conditions due to larger capacity of the remaining FW booster pump. The test revealed no unexpected system interactions or thermal-hydraulic phenomena.

- **Loss of one final stage feedwater heater (FWHE-3A/3B) from 100% power** – The purpose of this test is to verify plant response to asymmetric main feedwater loop temperatures at 100% EPU power. The test is initiated by bypassing main feedwater flow around one of the final stage FW heaters. This results in a rapid drop in MFW temperature to approximately the outlet temperature of FWHE-5. The objectives of the test were to:

  - Verify the ICS maintains the primary to secondary heat balance

  - Verify total feedwater flow demand is corrected (reduced) due to colder average FW temperature

  - Verify FW loop flows properly reratio to control delta T_{COLD}

  The DPT results indicate total FW flow control and flow reratioing function properly at EPU conditions, and primary to secondary heat balance is controlled effectively. Testing revealed no new or unexpected system interactions or thermo-hydraulic phenomena.

- **Dropped Control Rod Test – ICS No Runback** – The purpose of this test is to verify plant response to a dropped control rod with the Integrated Control System Asymmetric Rod Runback disabled. The test is initiated by dropping a single control rod from 100% EPU power. The objectives of the test are:
  - To ensure removal of the Integrated Control System Asymmetric Rod Runback does not introduce any unexpected adverse response following a dropped control rod event with no initial operator intervention
  - Evaluate RCS temperature response to ensure RCS T_{AVE} control is not adversely impacted by the presence of a CRD "out inhibit"
  - Evaluate FW response to ensure P/S heat balance is maintained

  The DPT results indicate that reactor power stabilized at about 95% following the dropped control rod. T_{AVE} stabilized about 2 degrees below setpoint. Control rod out motion was inhibited by the CRD out inhibit. Feedwater flows were adjusted to match reactor power and primary to secondary heat balance was controlled effectively. Testing revealed no new or unexpected system interactions or thermo-hydraulic phenomena.

<u>CR-3 Simulator Testing</u>

To provide additional assurance of acceptable plant transient response from EPU conditions CR-3 will perform each of the nine transient tests listed above on its plant referenced simulator following integration and testing of R17 component modifications and cycle 18 reactor fuel data. To support this activity the nine transient tests listed above have been run on the CR-3 simulator using an automated “script” to eliminate timing errors. The test sequences were run with the simulator in the MUR configuration. Following integration and testing of R17 component modifications and cycle 18 reactor fuel data each transient test scenario will be re-run with the simulator in the EPU configuration. Test results will be compared to the MUR benchmark data. Unanticipated differences in plant response will be investigated.

2.12.1.2.4 Adequacy of Proposed Testing Plan

CR-3 has many years of plant operating experience and understands well the plants operating characteristics and system interactions. The EPU test program will draw on this experience, as well as the results of the original startup test program and applicable industry experience as a means of ensuring safe operation at the new core thermal power level.

A comprehensive power ascension test program will monitor key component / system parameters and verify SSCs are performing within established design and operational limits. The program is based, in part, on the initial startup test program described in FSAR Chapter 13 and the associated CR-3 initial “Startup Report.” The test program establishes continuous monitoring of key plant operating parameters during power ascension and will specify “hold points” to allow for a thorough evaluation of plant performance. In addition, post modification functional test procedures, surveillance procedures, work order tasks, and maintenance procedures will be performed, as required, to verify that modified and existing SSCs are operating within applicable performance criteria.

During the power ascension, several transient and load change tests will be performed to ensure that the Plants dynamic behavior is satisfactory. This testing will demonstrate that integrated plant response during expected operational transients is consistent with design predictions, and that no new thermal hydraulic phenomena or adverse system interactions have been introduced by the uprate. The proposed transient and load change tests are described later in this section and summarized in Table 2.12.1-6.

The engineering change (EC) process at CR-3 requires design engineers to evaluate industry operating experience (OE) relevant to the design change or modification they are working on and to incorporate lessons learned in their design and functional test programs (as appropriate). The results of OE reviews are documented in the EC packages. In addition, CR-3 has evaluated various sources of operating experience associated with the EPU startup test process itself. This effort has identified a number of “lessons learned” applicable to startup and power ascension testing, post modification testing, main turbine testing, and the vibration monitoring program. Table 2.12.1-7 provides a summary of the OE items reviewed, and how the lessons learned from these events will be factored into the Power Ascension Test Program.

CR-3 has reviewed the planned EPU test program, including post modification testing, plans for the initial approach to the proposed maximum licensed thermal power level, transient and dynamic testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and the test program’s conformance with applicable regulations. The following discussions describe the proposed CR-3 Power Ascension Testing Program in detail and clearly demonstrate that the proposed testing program contains the necessary elements to assure safe operation at the EPU power level.

Proposed Power Ascension Test Plan

The Power Ascension Test Plan for EPU will be comprised of the following elements:

- MUR baseline testing – during R16 startup and fuel cycle 17, power operation baseline data will be collected on SSCs important to EPU power operations
  - Replacement steam generators
  - Main electrical generator (including LO, hydrogen and bus duct cooling systems)
  - Third stage Condensate Heaters and associated heater drain system
  - Moisture Separator Reheaters and associated heater drains (including belly drain heat exchangers)
  - Piping vibration monitoring (secondary piping including MS/FW piping in containment)
  - General area radiological and environmental conditions
- Applicable portions of the initial startup testing described in FSAR Tables 13-1, 13-2, 13-3, 13-4 and the CR-3 initial “Startup Report”
- Post modification functional testing (including vendor prescribed testing for major replacement components)
- Performance tests (PT), Surveillance tests (SP) and Operational tests (OP) normally performed during a post refueling startup.
- Special test requirements
  - Monitoring key plant / component parameters
  - Condensate System functional testing
  - Main turbine testing
  - Verification of Integrated Control System functionality
  - Transient testing
  - Vibration monitoring

Prior to the commencement of startup and power ascension testing, the test program will require the completion of the following activities:

- A complete “Station Readiness Review” in accordance with plant procedures. The station readiness review is accomplished by obtaining formal reviews and approvals from the Operations, Engineering, Maintenance, Licensing and Outage Scheduling departments confirming the readiness of the plant and plant organizations to commence heatup/startup activities including mode changes. This process provides the Plant General Manager (PGM) with the information necessary to verify the stations readiness to commence restart activities.

- Verification that required plant instrumentation setpoint changes and instrument recalibration is complete

- Verification that all required operator and maintenance training activities are complete

- Verification that all pre-startup testing activities are complete with satisfactory results

- Verification that all required plant procedure revisions, drawings, calculations, etc. are complete and that the procedures have been issued

- Verification that all commitments related to power ascension testing, the EPU License Amendment (LAR), the NRC EPU Safety Evaluation Report (SER), or other CR-3 EPU implementation requirements are satisfied

Performance Expectations and Test Plateaus

As part of the EPU Power Ascension Test Program, plant, systems, and component performance expectations will be established based on analyses, operating experience from similar equipment, consultation with plant engineering personnel, industry information sources (OE, vendors, consultants, etc.), and other design inputs. From these performance expectations, a master “monitored parameter list” will be developed, key parameters will be identified, and appropriate acceptance criteria will be established for each key parameter.

Some areas warranting special consideration during the post EPU startup and initial power ascension include the following:

- Reactor core performance and power distribution
- Steam Generator performance
- Main turbine and turbine supervisory response during rollup, loading and power ascension.
- Condensate pump performance and condensate system flow control response during heatup, startup and power ascension (Including deaerator and hotwell level control functions)
- Hydraulic interactions between the new feedwater booster pumps and main feedwater pumps, including monitoring and tuning the pump recirculation valves
- ICS control functions including TBV control, SG level control, T_{AVG} control, SUCV/LLCV overlap, MFWP control, main block valve (MBV) operation, and tracking/runback capability
- Operation of the Deaerator Bypass system
- Secondary component cooling system performance (including LO coolers, hydrogen coolers, and bus duct coolers)
- Feedwater heater and moisture separator reheater (MSR) drain control systems (including drain system optimization for higher drain flow rates)
- Vibration monitoring for condensate, feedwater, heater drain, and steam piping, valves and components

During the EPU startup, RCS T_{ave} and steam generator header pressure setpoints will be maintained at their MUR values until the plant reaches 2609 MWt. Using this approach key plant parameters will closely follow baseline plant response data collected during cycle 17 operations and performance anomalies will be easily identified.

Power will be increased in a slow and deliberate manner, stopping at predetermined hold points to accommodate steady-state data gathering, and key parameter evaluation. These hold points, or “test Plateaus” will be established at specific points during the power ascension to 2609 MWt (the MUR full power condition) to verify acceptable plant performance. In addition, observations of dynamic plant response between test plateaus will be evaluated by comparison with pre-determined acceptance criteria to verify SSCs and the overall plant is performing as expected.

At the 2609 MWt test plateau, when all applicable acceptance criteria have been satisfied, the RCS T_{AVE} and main steam header pressure setpoints will be reset to their EPU required values. The plant will be stabilized and key plant parameters will be reevaluated to ensure compliance with the applicable acceptance criteria at the elevated setpoints. Following this verification, the Plant General Manager and the Plant Nuclear Safety Committee (PNSC) will review the startup and power ascension test results, and any performance anomalies identified during the startup, to determine if power may be increased above 2609 MWt. Plant General Manager authorization, and PNSC concurrence, is required to proceed with the power ascension above 2609 MWt.

As power is increased above 2609 MWt, the intervals between test plateaus will be reduced. Reactor power will be increased through six additional test plateaus each differing by approximately 3% of EPU rated power. Plant response during the power maneuvers between plateaus and steady state performance at each test plateau hold point will be monitored, documented, and evaluated against pre-determined acceptance criteria. If key plant parameters satisfy acceptance criteria and no unexpected behavior is observed, system and component performance will be considered to comply with their design criteria and the plant will be maneuvered to the next test plateau.

A number of transient and dynamic tests will be performed during the power ascension. Transient test data will be compared to predictions obtained from analytical models and other engineering inputs to verify correct transient response. Any significant deviation from predicted response will be evaluated and reconciled before proceeding with the power ascension.

Transient tests are listed and described in this section and summarized in Table 2.12.1-6. A summary of the overall test plan is provided in Table 2.12.1-8.

Acceptance Criteria

Power ascension test procedures will clearly identify the criteria against which the success or failure of the test will be judged. Where applicable, acceptance criteria will account for measurement errors and uncertainties. Quantitative criteria will include appropriate tolerances. Qualitative criteria will be clearly defined.

CR-3 will utilize a two tiered approach for establishing test procedure acceptance criteria:

- Level 1 acceptance criteria will be established for parameters that are:
    - Safety significant
    - Critical to the acceptable operation of a plant SSC

  If a Level 1 criterion is not satisfied, the power ascension will be stopped and the plant will be placed in a condition that is judged to be safe based upon prior testing. The power ascension test procedure and Technical Specifications will provide direction for actions to be taken to assure the plant is safe and stable. Resolution of the issue that resulted in exceeding the Level 1 criterion must be complete before resuming testing. Following resolution, the applicable portion of the test procedure must be repeated to verify that the Level 1 requirement is satisfied. A description of the problem and resolution must be included in the report documenting successful completion of the test.
- Level 2 acceptance criteria are values that relate to plant functions or parameters that are not safety significant. If Level 2 criteria are not met, the Power Ascension Test Plan may continue.

Investigation of the issue that resulted in exceeding the Level 2 criterion may continue in parallel with the power ascension. These investigations will be handled in accordance with existing plant processes and procedures.

Transient and Dynamic Tests

The following transient testing will be performed in support of the CR-3 power uprate. The purpose of these tests is to ensure SSCs are performing as expected and to verify that no adverse system interactions or unexpected thermal hydraulic phenomena are introduced as a result of the EPU or the associated modifications.

1. **Integrated Control System Functional Testing**

During power ascension to EPU conditions, ICS control functions will be tested using normal testing procedures. Testing will be conducted to evaluate the response of various ICS control functions at approximately 10, 40, 75, and 92% EPU rated thermal power. The tests are accomplished by manually introducing small step changes in control signals and monitoring resulting ICS response. The testing will evaluate the following ICS control functions:

- SG level control
- Turbine bypass valve operation
- Main feedwater control in both the d/p and flow control modes
- Delta Tc controller response
- Turbine header pressure control
- T_{AVG} control (by both the reactor and feedwater subsystems).

The power ascension test procedure will include steps to verify and document important ICS control functions such as:

- Turbine Bypass Valve position/demand vs. power
- Start-Up/Low Load Control Valve overlap
- Low Load Block Valve and Main Block Valve operation (including overlap)
- Main Feedwater Pump response during the transition to and from the flow control mode
- Correlation between condensate flow and FW flow over the power range.

ICS functional test results will provide a new operational baseline, and will be used to benchmark and tune CR-3 simulator response.

2. **Turbine/Generator Operation Test**

Prior to rolling the main turbine with steam, CR-3 will complete all applicable post modification testing requirements specified by the turbine vendor and the engineering/functional test group. Operability testing will commence with the turbine at zero speed and will continue until full. The test program will include:

- Verification of proper operation of turbine support systems (EHC, lube oil, seal oil, gland steam, component cooling, hydrogen cooling, etc.)
- Placing the Turbine generator on turning gear and monitoring for interference (rubs)

- Verification that Turbine Supervisory Instrumentation is operable and shaft eccentricity and rotor position are within allowable limits
- Turbine latching and functional testing verification of main turbine trip block trip functions (low vacuum, low oil pressure, overspeed, and thrust bearing).
- During the turbine roll up to synchronous speed, key parameters such as bearing vibration, casing expansion, differential expansion, bearing metal and lube oil return temperatures, and EHC control functions will be carefully monitored
- Predicted turbine critical speeds will be verified during the rollup to synchronous speed
- With the turbine at synchronous speed the electronic trip device, the turbine trip solenoid valves, the overspeed protection controller (OPC), and the mechanical overspeed trip mechanism will be tested (see Test 3 below). During these tests turbine steam isolation valves will be verified operable
- During initial loading and power ascension additional parameters will be monitored including voltage regulator operation, generator stator and rotor temperatures, hot and cold gas temperatures, bus duct temperatures, etc.
- At < 40% power a turbine trip test will be performed (see Test 4 below).
- Throughout the turbine rollup, startup and power ascension, the throttle valve (TV) and governor valve (GV) response will be verified to follow the expected profile (Note – the HP turbine modifications convert it from partial arc to full arc admission).
- A complete set of turbine generator baseline data will be collected during the startup and power ascension.

3. **Main Turbine Overspeed Trip Test**

At a power level between 10% and 15% RTP and the main turbine at synchronous speed, the OPC function will be defeated and turbine speed will be increased to the mechanical overspeed trip setpoint to verify the operability of the mechanical overspeed trip mechanism. The purpose of this test is to:

- Verify the operability of the overspeed trip mechanism (mechanical)
- Verify operability of the main turbine trip valves
- Verify operability of Turbine Bypass Valves (TBVs will initially be partially open and controlling header pressure).

4. **Main Turbine Trip from < 40% EPU Power**

At a power level of < 40% RTP, the main turbine will be manually tripped from the control room and the following will be verified:

- The turbine electronic trip mechanism and all turbine trip valves operate properly
- When the turbine generator trips the ICS header pressure control transfers to the turbine bypass valves (TBVs will initially be closed)
- The ICS initiates a runback to approximately 25% of EPU rated power
- Turbine Bypass Valves properly control turbine header pressure
- Major plant control system functions such as condensate flow, deaerator level control, RCS pressure/pressurizer level control, and T_{AVG} control operate properly during the transient

The CR-3 simulator will be benchmarked against a comprehensive set of plant data obtained during the main turbine trip test in accordance with the Event Comparison Testing procedure. Any discrepancies in simulator response will be entered into the simulator trouble report system for resolution. Event comparison test results will be retained in the simulator files for future reference. Analytical test results obtained during the DPT testing will also be benchmarked against key plant parameters recorded during the main turbine trip test. Significant deviations from actual plant response will be documented and resolved in accordance with the plant's corrective action program.

5. **10% Load Change at 35% RTP Power**

At approximately 35% RTP the ICS unit load demand (ULD) will be decreased from 35% to 25% RTP at a rapid rate (to be determined based on allowable maneuvering rates). The purpose of this test is to evaluate the transient response of the ICS and ICS controlled components during a rapid power maneuver with the ICS in the d/p control mode (MFWPs controlling d/p, FW control valves controlling flow). The dynamic behavior of the various plant components will be observed and evaluated against predefined acceptance criteria to ensure that EPU modifications and ICS recalibration have not adversely affected plant transient response. When acceptable dynamic performance has been verified, power will be ramped back to 35%.

6. **Load Change 35% to Main Block Valves OPEN**

From approximately 35% RTP the ICS unit load demand will be increased until the Main Feedwater Block Valves open on both FW loops. During this evolution the ICS FW subsystem will transition from the d/p control mode (Low Load Control Valves controlling flow, MFWPs controlling d/p across valves) to the flow control mode (Main Block Valves open, MFWP Speed controlling flow). The purpose of this test is to evaluate the dynamic response of the ICS and ICS controlled components during the transition to the flow control mode. The dynamic behavior of the various plant components will be observed and evaluated against predefined acceptance criteria to ensure that EPU modifications and ICS scaling/recalibration have not adversely affected plant transient response.

7. **Load Change from Main Block Valves Open to Main Block Valves Closed**

Following verification of Main Block Valve operability and flow control mode verification (Test 6 above), power will be reduced at normal maneuvering rates until the Main Block Valves close. During this evolution the ICS FW subsystem will transition from the flow control mode (Main Block Valves open, MFWP speed controlling flow) to the d/p control mode (Low Load Control Valves controlling flow, MFWPs controlling d/p across valves). The purpose of this test is to evaluate the dynamic response of the ICS and ICS controlled components during the transition to the d/p control mode. The dynamic behavior of the various plant components will be observed and evaluated against predefined acceptance criteria to ensure that EPU modifications and ICS recalibration have not adversely affected plant transient response.

8. **10% Load Change at 100% RTP Power**

From approximately 100% RTP the ICS unit load demand will be decreased from 100% to 90% EPU power at the maximum allowable rate (to be determined based on allowable maneuvering rates). The purpose of this test is to evaluate the transient response of the ICS and ICS controlled components during a rapid power maneuver with the ICS in the flow control mode (Main Block Valves open, MFWPs controlling FW flow). The dynamic behavior of the various plant control systems will be observed and evaluated against predefined acceptance criteria to ensure that the combination of increased power and changes to the plant configuration (EPU modifications) have not adversely affected plant response. For the power reduction test the maximum allowable rate was selected because plant procedures allow the use of that rate during rapid power reductions. Once acceptable dynamic performance has been verified at 90% power will be returned to 100% power at a normal maneuvering rate.

Table 2.12.1-6 provides a summary of the plant transient tests that will be incorporated in the CR-3 EPU Power Ascension Test Plan.

Vibration Monitoring

A Piping and Equipment Vibration Monitoring Program has been established to ensure that flow induced piping vibrations, following EPU implementation, are not detrimental to the plant SSCs. To support this effort, CR-3 will install state of the art monitoring equipment at selected locations. Recognized experts in the field of vibration analysis have been contracted to assist the CR-3 in the identification of locations to be monitored, the establishment of acceptance criteria, the evaluation of electronically acquired vibration data, and the acquisition and evaluation of hand logged data on piping and components that are not amenable to installed electronic instrumentation.

The vibration monitoring program will be performed in two phases. Phase I began during the Fall 2009 outage (16R) and will continue during the subsequent fuel cycle. Phase II will begin with the installation of monitoring equipment during R17 and will continue through heatup, startup and power ascension to EPU conditions.

Phase I program activities concentrate on Balance of Plant (BOP) data point selection, instrumentation installation, baseline data acquisition, data analysis, and detail planning for work in the Reactor Building during R17. BOP systems addressed in this phase include accessible lines that will experience a significant increase in their process flow rates or piping and components that have historically exhibited visible vibration displacements during plant operation. Branch lines attached to these lines (experiencing increased process flows) will also be manually monitored (screened) to detect anomalous behavior that would indicate that a more in-depth evaluation should be undertaken since experience has shown that branch lines are susceptible to vibration-induced damage. The scope of the Piping and Equipment Vibration Monitoring Program includes the following systems:

- Main Steam, including Reheater Inlet
- Feedwater
- Condensate
- Extraction Steam
- Heater Drains
- Moisture Separator Drains
- Turbine Gland Steam and Drains

The program scope will also include any lines or equipment within the monitored systems that have been modified or otherwise identified through the CR-3 Corrective Action Program as having already experienced vibration issues. MSR belly drains, secondary cooling recirculation lines, and CDHE drain lines were chosen since they are expected to exhibit higher vibration levels. The main turbine governor valves (MSV-1, 2, 3, and 4) are also included.

Comprehensive BOP thermal studies using the PEPSE Code are an integral part of the overall EPU Program. Flows, temperatures, and pressures calculated for EPU conditions have been compared with values at the same points from benchmarked PEPSE calculations available at MUR rated power. Average flow velocities were calculated for each piping run under consideration from mass flow values and thermodynamic parameters output from the PEPSE Code. Candidate monitoring points were ranked according to the % increase in flow velocity. Flow changes resulting in fluid velocity changes below 1% were eliminated from consideration for electronic data acquisition. Component internal flow paths were not considered in this comparative study.

ASME OM-S/G–2007 is an established standard for the reliable operation and maintenance of nuclear power plant equipment, particularly as they relate to start-up and periodic performance and functional testing and monitoring of systems and components. This standard includes the establishment of test objectives, test intervals, test methods, test data requirements, as well as the analysis and acceptability of test results and the course of action to be pursued when test results are unacceptable. This standard is the basis for the establishment of the piping vibration acceptance criteria for the EPU piping project.

CR-3 will also perform a visual survey of the secondary piping systems during transient and steady state operational modes and select locations in accordance with the guidance provided in ASME OM Code Part 3, VMG 3, Section 4, for hand logging of vibration measurements. These measurements will be evaluated in accordance with VMG 3 acceptance criteria as called out in Section 3.2.3 of the ASME Code.

Due to the small change in temperature and the associated small change in thermal displacement in the secondary piping systems due to EPU, specific thermal expansion testing as outlined in OM Code, Part 7 is not required. However, during the hold points at the power level plateaus personnel responsible for the vibration monitoring walk downs will also be observant of any thermal expansion problems such as crushed insulation or piping in contact with adjacent equipment.

During Phase II, the piping and equipment within the scope of the vibration monitoring program will be observed at several different plant operating conditions. The first observations will be conducted at turbine roll-off. These initial observations and observations recorded during the power ascension at 30%, 50%, and 85% EPU rated power will establish the baseline piping vibration level for further comparison.

During the power ascension, visual observations and instrumented data recording will be performed at the 85%, 88%, 91%, 94%, 97%, and 100% EPU power levels. At each power level plateau there will be a hold for sufficient time to perform visual observations and data recording and, if required, to assess the vibration response in the piping systems and at the locations identified to be of potential vibration concern. The observations and data obtained will be assessed to determine if:

- The vibration response meets the acceptance criteria of OM Code, Part 3
- The vibration readings between the various power plateaus reveal unexpected or unacceptable trends.

In addition to the Piping and Equipment Vibration Monitoring Program described above, CR-3 will obtain a full set of steam generator baseline vibration data using installed Loose Parts Monitoring System (LPMS) accelerometers. Data will be obtained during the startup and power ascension following R16 and again during the startup and power ascension following R17. Monitoring will begin prior to RCP startup and will be repeated incrementally during the power ascension to full EPU power.

Anomalous LPMS indications will be evaluated in accordance with existing plant procedures. Any instances where vibrations that are determined to be unacceptable during the power ascension will be addressed by making a thorough engineering evaluation, and if necessary, a plant modification.

2.12.1.3 Conclusion

CR-3 has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed EPU thermal power level, and the test program conformance with applicable regulations. CR-3 concludes that the proposed EPU test program provides adequate assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Further, CR-3 finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, CR-3 finds the proposed Approach to EPU Power Level and Test Plan acceptable and will perform testing, as described in this section, during the EPU startup and power ascension.

2.12.1.4 References

None

Table 2.12.1-1
Rated Thermal Power History

License Amendment	Date	RTP Limit/change	Type of Uprate
Initial License	12/03/1976	2452 MWt	Initial power level
Amendment 41	09/30/1994	2452 MWt to 2544 MWt	Stretch
Amendment 205	12/04/2002	2544 MWt to 2568 MWt	Stretch
Amendment 228	12/26/2007	2568 MWt to 2609 MWt	Measurement Uncertainty Recapture (MUR)

Table 2.12.1-2
Potentially Affected FSAR Table 13-1, Pre-Fueling Tests

FSAR Table 13-1 Reference	System Test Description	Potentially Affected Test Objectives	Planned EPU Testing
System Test 1	Electrical Systems	(f) Exciter check for proper voltage build up (g) Insulation tests as required.	1. Key main generator and exciter parameters will be carefully monitored during startup and power ascension. Hold points will be established to compare key test parameters with applicable acceptance criteria. Enhanced monitoring will begin as power ascension above MUR RTP commences. 2. In addition to the main generator and exciter the following other electrical systems will be monitored: • Step up Transformers and bushings • Output breakers • Bus duct coolers
System Test 4	Condensate and Feedwater System	• Flushing and hydrostatic testing as applicable. • Complete inspection for system integrity • Verify valve and control operability and setpoints. • Functional test when main steam is available.	1. Post modification testing will include requirements for system flushing. A complete walkdown and inspection for system integrity will be performed prior to system turnover. 2. The startup and power ascension test procedure will include guidance to verify FW and CD system valves, pumps and control systems are operable and performing as designed (including verification of key operational setpoints).
System Test 9	Secondary Services Closed Cycle Cooling	(a) Adequate flow is provided for heat removal. (b) Pumps function as designed (c) All temperature, flow and pressure instrumentation performs satisfactorily, including applicable alarms.	1. Key system parameters and component temperatures will be monitored during startup and power ascension. Hold points will be established to compare test parameters with applicable acceptance criteria. 2. Enhanced monitoring will begin as power ascension above MUR RTP commences. Particular emphasis will be placed on verifying overall system performance as the H2 cooler flow increases.

Table 2.12.1-2
Potentially Affected FSAR Table 13-1, Pre-Fueling Tests

FSAR Table 13-1 Reference	System Test Description	Potentially Affected Test Objectives	Planned EPU Testing
System Test 15	Decay Heat Removal System	(a) Pumps function as designed (b) Adequate flow is provided for heat removal. (c) System meets requirements when operated in the safeguards mode. (d) All temperature, flow and pressure instrumentation performs satisfactorily and provides alarms that function satisfactorily at required locations. (e) All manual and remotely operated valves operate satisfactorily. (f) Flushing and hydrostatic tests will be performed where applicable	1. Post modification test procedures will include requirements to perform system flushing and a complete walkdown to verify system integrity. 2. Post modification and system functional testing will verify DHPs meet minimum design criteria with sufficient margin. System flow requirements will be verified and documented in both the DHR and LPI modes of operation. 3. Post modification testing on new components and surveillance testing all safety related components, controls and alarms in the system will be used to verify operability.
System Test 23	Chemical Addition and Boron Recovery System.	(a) All manual and remotely operated valves function properly. (c) All controls and alarms function as specified. (i) Also flushing and hydrostatic tests will be performed where applicable	1. Post modification test procedures will include requirements to perform flushing and a complete walkdown of the MU Tank Bypass flow path to verify system integrity. 2. Post modification and system functional testing will verify the MU Tank Bypass flow path and isolation valve MUV-661 are functional.
System Test 33	Low Pressure Injection System	(a) All pumps function as designed (b) All manual and remotely operated valves operate satisfactorily. (f) All instrumentation performs satisfactorily and provides alarms that function satisfactorily at the required locations.	1. Post modification and system functional testing will verify DHPs meet minimum design criteria with sufficient margin. System flow requirements will be verified and documented in both the DHR and LPI modes of operation. 2. Post modification test procedure and valve surveillance procedure will include requirements to verify operability of new boron precipitation hot leg injection valves. 3. Post modification system functional testing and normal surveillance testing will verify operability of new instrumentation and alarm functions.

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
13.4.6 Table 13.2 (1)	**Control Rod Trip Test** Verify proper CRD and RPS functioning and that rod drop times meet safety requirements and that APSRs do not trip.	Post Fueling Precriticality	Yes	1. Existing plant surveillance procedures verify each RPS channel and associated CRD breaker are demonstrated operable. 2. Existing plant surveillance procedures verify the RPS CRD breakers respond properly to a manual reactor trip. 3. Existing plant surveillance procedures verify safety and regulating rod drop times meet applicable surveillance requirements.
13.4.6 Table 13.2 (2)	**RC Flow and Flow Coastdown Test** Verify RC flow and coastdown characteristics for selected pump combinations and verify ability of associated instrumentation to properly measure.	Post Fueling Precriticality	No	**CR-3 is proposing elimination of this test.** Reactor coolant pumps, seals, and motors have not been modified to support plant operation at EPU conditions and therefore have no impact on RCP coastdown time. Therefore, the power uprate does not invalidate the testing performed during initial startup. Reactor coolant system flow instrumentation is not being modified for EPU. Analysis performed for Section 1.1, NSSS Parameters, and referenced in Section 2.2.2.6, Reactor Coolant Pumps and Supports demonstrates that RCS total flow rate is only slightly reduced at EPU conditions due to density. Plant procedures measure RCS total and loop flow rates at HZP to ensure they are within technical specification limits.
13.4.6 Table 13.2 (3)	**Chemical and Radiochemical Tests** FSAR Table 13-1 (38) describes a chemical and radiochemical test program for the RCS, SGs, SF pools, SW and DC systems. Testing is specified from RCS fill and vent through power	Post Fueling Precriticality	Yes	Chemical and Radiochemical testing during startup and power ascension is defined by existing plant procedures and includes the scope of testing described in FSAR Table 13.1. 1. Nuclear Chemistry Master Scheduling Program defines specific parameters to be monitored for all modes of plant operations. 2. Primary Water Chemistry Program provides the programmatic guidance to control primary system water chemistry in order to maximize fuel and

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
	operation.			material integrity while minimizing plant radiation fields during all modes of operation. 3. Secondary Water Chemistry Program provides the programmatic guidance to minimize corrosion throughout the secondary plant and minimize fouling of the steam generators and BOP heat exchangers during all modes of plant operation.
13.4.6 Table 13.2 (4)	**Pressurizer Effectiveness** Set spray flow to meet design requirements.	Post Fueling Precriticality	No	**CR-3 is proposing elimination of this test.** On high RCS pressure the pressurizer spray valve is designed to open to a preset position (40%) which has been verified to provide the required design flow rate of $\geq$ 190 gpm. The RCS pressure and pressurizer level operating setpoints are unchanged as a result of the EPU. The components comprising the pressurizer pressure and level control systems have not been modified and the pressurizer spray flow design requirements are unchanged for the EPU. Therefore, additional testing is not required.
13.4.6 Table 13.2 (5)	**In-Service Loose Parts and Vibration Monitoring System** Take baseline data under steady state and transient RC flow conditions.	Post Fueling Precriticality	Yes	CR-3 will obtain a full set of OTSG baseline data, using installed LPMS accelerometers, during the startup and power ascension following R16 and again during the startup and power ascension following R17. Monitoring will begin prior to RCP startup and will be repeated incrementally during the power ascension to full EPU power. Anomalous LPMS indications will be evaluated in accordance with existing plant procedures. Concerning vibration monitoring, a comprehensive vibration monitoring program will be implemented for EPU and will cover secondary system piping and components during heatup, startup and power operation. The purpose of the program is to identify vibration anomalies and establish a baseline database. The vibration monitoring program is described in Section 2.12.1.2.4.

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
13.4.6 Table 13.2 (6)	**Calibration and Neutron Response of Source Range Monitoring** Verify that source range channels indicate ≥ 2 cps.	Post Fueling Precriticality	Yes	1. Prior to reactor/plant startup source range nuclear instrumentation will be calibrated using existing surveillance procedures. 2. During the reactor startup, source range instrumentation is required to be verified operable in accordance with existing operating procedures. Note – current requirement for NI-1 and NI-2 operability is > 0.4 cps <u>OR</u> NI14 and NI15 operable with > 0.1 cps.
13.4.8.1 Table 13.3 (1)	**Ejected and Stuck Control Rod Assembly Worth Measurements** To establish worth of Control Rod Assemblies (CRAs) calculated to be the most reactive.	Post Criticality Zero/Low Power	No	**CR-3 is proposing elimination of this test.** Controlling Procedure For Zero Power Physics Testing provides the sequence for performing zero power physics tests, including comparison of rod worth measurements to predictions. **Control Rod Assembly Discussion** The original CR-3 Technical Specifications 3/4.10 Special Test Exceptions, 3/4.10.4 Shutdown Margin allowed for a minimal amount of control rod worth to be immediately available for reactivity control when tests were performed for rod worth measurement. The LCO 3.10.4 allowed for SHUTDOWN MARGIN requirement to be suspended provided that a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s). PER ITS 3.1.9 PHYSICS TESTS Exceptions – MODE 2, SDM must be maintained above the minimum required (changing to 1.3% $\Delta k/k$). Since operation with all but the most reactive control assembly out is no longer permitted and SDM must be maintained, measurement of the most reactive control assembly is not required. **Control Group Discussion** Control rod Groups 7 and 6 and sometimes Group 5 are measured every

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				cycle as part of the normal startup physics testing with a historical (Cycles 10-16) total measured rod worth varying between approximately 1660 and 3200 pcm. The historical total control rod group worth at beginning of cycle (BOC), including all trippable rod groups (Groups 1-7) is between 7000 pcm and 7400 pcm for Cycles 14 through 17. Based on the historical total rod worths for Groups 1 through 7 and the measured rod group worths of approximately 7000 pcm, it is not expected that the most reactive control rod group would exceed the sum of the currently measured groups. A most reactive rod worth prediction and measurement would therefore be no more difficult than that currently performed on a cycle basis. A +6%/-9% (P-M) total measured rod worth acceptance criterion has been justified for CR-3 as an appropriate value for zero power physics testing. This acceptance criterion is smaller than the typical value of ±10% for the sum of groups provided in ANSI/ANS 19.6.1-2005 Table A.1. It also provides a criterion for individual rod worth differences of ±15% which is equal to that provided in the ANSI standard. Statistics have been reviewed for the comparison of total rod worth measurements to predictions for Cycles 10 through 16 (7 data points). These provide a mean difference (p-m) of -2.65% with a standard deviation of 3.56% for the total measured rod worth. Therefore, the current procedure for control rod worth measurement is sufficient and there is no need to identify and measure the worth of the most reactive rod group.
13.4.8.1 Table 13.3 (2)	**Control Rod Group Calibration** To determine differential and integral reactivity worth.	Post Criticality Zero/Low Power	Yes	Controlling Procedure For Zero Power Physics Testing provides the sequence for performing zero power physics tests. This procedure is used to validate individual and total regulating rod group predicted worth. Differential control rod worth is calculated and recorded as part of the process for

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				determining integral control rod worth.
13.4.8.1 Table 13.3 (3)	**Boron Worth Calibration** To determine the boron reactivity worth.	Post Criticality Zero/Low Power	Yes	Controlling Procedure For Zero Power Physics Testing provides the sequence for performing zero power physics tests. This procedure is used to determine differential Boron worth and all rods out critical boron.
13.4.8.1 Table 13.3 (4)	**Moderator Temperature Coefficient Measurement** To verify reactivity effects associated with reactor coolant temperature change.	Post Criticality Zero/Low Power	Yes	Controlling Procedure For Zero Power Physics Testing provides the sequence for performing zero power physics tests. This procedure directs that moderator temperature coefficient be determined in accordance with plant procedure, Moderator Temperature Coefficient Determination At Startup Following Refueling.
13.4.8.1 Table 13.3 (5)	**Excess Reactivity Measurement** To measure the excess reactivity of the core to confirm the predicted shutdown margin.	Post Criticality Zero/Low Power	Yes	1. Controlling Procedure For Zero Power Physics Testing provides the sequence for performing zero power physics tests. Tests verify the measured reactivity associated with control rods, boron, and changes in temperature (moderator and fuel) agree with predicted values. 2. The procedure for Reactivity Balance Calculations, is used to determine actual shutdown margin and to perform other reactivity balance calculations.
13.4.8.1 Table 13.3 (6)	**Nuclear Instrumentation Check** To verify the overlap of the source and intermediate range nuclear instrumentation at zero power.	Post Criticality Zero/Low Power	Yes	1. Prior to reactor/plant startup source range nuclear instrumentation will be calibrated. 2. Intermediate range instrumentation will be calibrated. 3. During the reactor startup, source and intermediate range instrument overlap are verified.
13.4.8.1 Table 13.3 (7)	**Biological Shield Survey** To confirm adequacy of radiation shielding to zero and	Post Criticality Zero/Low Power	Yes	A “Biological Shield Survey” will be included in the power ascension test plan to augment routine radiation surveys performed during normal operation. The intent of the survey will be to assure that plant shielding is adequate and

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
	low power levels.			areas with elevated radiation levels are properly identified and posted. 1. Prior to R17 appropriate survey locations will be identified and benchmark data at each location will be recorded. 2. During startup and power ascension to 85% EPU power selected locations will be spot checked and compared to baseline readings. 3. Between 85 and 100% EPU power all locations will be surveyed and a comparison between pre and post EPU radiological conditions will be made. 4. The results of the shielding surveys will be documented, including the comparison between pre and post EPU radiological conditions and an assessment of the adequacy of plant biological shielding (primary shield and secondary shield walls, containment, and Auxiliary Building walls) will be completed.
13.4.8.1 Table 13.3 (8)	**Effluent and Effluent Monitoring** Effluents will be analyzed and documented during zero and low power testing • Monitors will be calibrated and verified for proper operation. • Setpoints will be incorporated into all monitors and valve operation will be checked. • All alarms and indicators will be tested for proper operation at predetermined setpoints and ranges.	Post Criticality Zero/Low Power	Yes	Plant effluents will be analyzed and releases monitored as required throughout the startup and power ascension in accordance with normal plant procedures. The Offsite Dose Calculation Manual (ODCM) describes the effluent monitoring requirements currently in place at CR-3: 1. The CR-3 ODCM requires that plant effluent monitors are calibrated and functionally tested on a frequency consistent with NRC recommendations. These requirements are implemented in plant procedures. Chemistry Procedures are used for calibration of effluent monitors. Surveillance procedures are used for functional testing of these monitors. 2. The CR-3 ODCM requires that set-points be employed on all effluent release pathways. Surveillance procedures implement this requirement by calculating a set-point for each approved release. The set-point is provided to Operations by way of a release permit. 3. Valve operations are verified by Operations procedures for each type of release.

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
13.4.8.1 Table 13.3 (9)	**Chemical and Radiochemical Tests** FSAR Table 13.1 (38) describes a chemical and radiochemical test program for the RCS, SGs, SF pools, SW and DC systems. Testing is specified from RCS fill and vent through power operation.	Post Criticality Zero/Low Power	Yes	Chemical and Radiochemical testing during startup and power ascension is defined by existing plant procedures and includes the scope of testing described in FSAR Table 13.1. 1. Nuclear Chemistry Master Scheduling Program defines specific parameters to be monitored for all modes of plant operations. 2. Primary Water Chemistry Program provides the programmatic guidance to control primary system water chemistry in order to maximize fuel and material integrity while minimizing plant radiation fields during all modes of operation. 3. Secondary Water Chemistry Program provides the programmatic guidance to minimize corrosion throughout the secondary plant and minimize fouling of the steam generators and BOP heat exchangers during all modes of plant operation.
13.4.8.2 Table 13.4 (1)	**Turbine/Reactor Trip Test** To confirm proper unit response during turbine/reactor trip (at 40% and 100% power or near as practical).	Turbine Trip Tests (75 and 100%) Reactor Trip Test (40%)	No	1. Initial startup testing included turbine trip tests from 75% and 100% power. The purpose of these tests was to verify overall plant response during the ICS runback to 15% reactor power. **CR-3 is proposing elimination of these tests**. Justification for the elimination of this test is provided in section 2.12.1.2.3. 2. Initial startup testing included a reactor trip test from 40% power. **CR-3 is proposing elimination of this test**. Justification for the elimination of this test is provided in section 2.12.1.2.3.
13.4.8.2 Table 13.4 (2)	**Integrated Control System Test** To determine unit ability to maintain stability during steady state and load transient conditions.	0 – 100% (Normal Transients) 40% RCP Trip 75%	Yes (except as noted)	1. The startup and power ascension test procedure will include numerous hold points to verify key parameters and the ability of the ICS to maintain steady state control. 2. The startup and power ascension test procedure will include a requirement for verifying key ICS control functions during the power ascension 3. The startup and power ascension test procedure will include a

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
		MFWP Trip		requirement for performing three ICS transient (load change) tests during the power ascension to demonstrate normal ICS control functions 4. The startup and power ascension test procedure will include a requirement for performing a turbine trip transient < 40% power to verify ICS control during a runback 5. The “Startup Report” described tests to verify the ability of the ICS to properly control the plant in the “Turbine following” and the “Reactor following” modes of operation. **CR-3 is proposing elimination of these tests.** ICS operation in integrated, turbine following, and reactor following modes are routine B&W plant evolutions exercised during every plant startup and shutdown. There have been no changes made to ICS hardware or control algorithms that would affect control in turbine or reactor following modes. Therefore, CR-3 concludes that the power uprate has not introduced any new thermal-hydraulic phenomena or system interactions and does not invalidate the testing performed during initial startup. 6. The “Startup Report” described tests to verify the ability of the Integrated Control System to properly control the plant following (1) a RCP trip from 40% power and (2) a MFWP trip from 75% power. **CR-3 is proposing elimination of these tests.** Justification for the elimination of this test is provided in section 2.12.1.2.3.
13.4.8.2 Table 13.4 (3)	**Unit Loss of Electrical Load** To verify proper response of the reactor and auxiliary systems on separation of the generator from the transmission system (at 100% power or as near as practical).	100%	No	**CR-3 is proposing elimination of this test**. Justification for the elimination of this test is provided in section 2.12.1.2.3.

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
13.4.8.2 Table 13.4 (4)	**Unit Load Steady State and Transient Test** To determine unit behavior at steady state and response to load changes.	0 – 100% (Normal Transients) 40% RCP Trip 75% MFWP Trip	Yes (except as noted)	1. The startup and power ascension test procedure will include numerous hold points to verify plant stability and key system parameters. 2. The startup and power ascension test procedure will include a requirement for verifying key plant control functions during the power ascension 3. The startup and power ascension test procedure will include a requirement for performing controlled plant transient testing during the power ascension 4. The startup and power ascension test procedure will include a requirement for performing a turbine trip transient test (with ICS runback) from 40% power 5. The initial "Startup Report" described tests to verify plant response during the following operational transient s; (1) an RCP trip from < 40% power and (2) a MFWP trip from 75% power. **CR-3 is proposing elimination of these tests.** Justifications for the elimination of these tests are provided in section 2.12.1.2.3.
13.4.8.2 Table 13.4 (5)	**Reactivity Coefficient at Power** To determine reactivity coefficient as related to boron concentration, moderator temperature and control rod position at various power levels.	40%, 75% 100%	Yes	Controlling Procedure For Power Ascension Testing provides the sequence for performing physics testing during the power ascension. The procedure provides guidance for determining reactivity coefficients at power. Reactor Engineering will identify the specific monitoring requirements for EPU in the power ascension test procedure.
13.4.8.2 Table 13.4 (6)	**Unit Heat Balance** To verify the adequacy of computer primary and secondary heat balance	15%, 40% 75%, 100%	Yes	1. During the power ascension, the Unit Startup Surveillance Plan requires the performance of Daily Heat balance Power Comparison as power is raised above 15%, 50%, 75% and at 100%. It also verifies (1) agreement between heat balance power and NI power, (2) agreement between AULD and FIDMS heat balances, and (3) agreement between

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
	calculations.			heat balance power and power based on core delta T. 2. During the power ascension, Unit Startup Surveillance Plan requires the performance of a Quarterly Heat Balance Verification as power is increased above 85%. It also verifies accuracy of heat balance inputs and agreement between the Fixed Incore Detector Monitoring System (FIDMS) and Automatic Unit Load Demand (AULD) heat balance calculations.
13.4.8.2 Table 13.4 (7)	**Core Power Distribution** To measure core flux and power distribution during power ascension and to evaluate core performance and Departure from Nucleate Boiling Ratio (DNBR) at approximately 15%, 40%, 75%, and 100% power.	15%, 40% 75%, 100%	Yes (except as noted)	Controlling Procedure For Power Ascension Testing provides the sequence for performing physics testing during the power ascension. It also requires core power distribution testing at two test plateaus during the power ascension (between 40% – 85% and 85% – 100% power). If Reactor Engineering determines more frequent monitoring is required it will be specified in the power ascension test procedure. Concerning the DNBR evaluation, CR-3 now monitors power peaking factors (f_Q and $F^N_{\Delta h}$) to determine that the core is operating within design limits and to verify adequate margin exists during steady state and transient operation. The plant's Fixed Incore Detector Monitoring System (FIDMS) software calculates the margin to f_Q and $F^N_{\Delta h}$ limits using data from the fixed incore detector arrays. Consequently, linear heat rate and DNBR extrapolations have been replaced by the power peaking and margin monitoring provided by the FIDMS software.
13.4.8.2 Table 13.4 (8)	**Biological Shield Survey** To confirm the adequacy of radiation shielding at intermediate and high power levels.	0%, 40%, 100%	Yes	A "Biological Shield Survey" will be included in the power ascension test plan to augment routine radiation surveys performed during normal operation. The intent of the survey will be to assure that plant shielding is adequate and areas with elevated radiation levels are properly identified and posted. 1. Prior to R17 appropriate survey locations will be identified and benchmark data at each location will be recorded. 2. During startup and power ascension to 85% EPU power selected

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				locations will be spot checked and compared to baseline readings. 3. Between 85 and 100% EPU power all locations will be surveyed and a comparison between pre and post EPU radiological conditions will be made. The results of the shielding surveys will be documented, including the comparison between pre and post EPU radiological conditions and an assessment of the adequacy of plant biological shielding (primary shield and secondary shield walls, containment, and Auxiliary Building walls) will be completed.
13.4.8.2 Table 13.4 (9)	**Pseudo Rod Ejection Test** This test was performed to verify the safety analysis relative to accidental ejection of a control rod which is partly inserted in the core under full power conditions. Using a combination of boration and rod control, the reactivity of the greatest worth rod will be checked to ensure that if it is ejected, reactivity insertion will not be greater than Technical Specifications. This test will be performed at 0% and 40% power.	40%, 47%	No	**CR-3 is proposing elimination of this test.** Controlling Procedure For Zero Power Physics Testing provides the sequence for performing zero power physics tests, including comparison of rod worth measurements to predictions. A +6%/-9% (P-M) total measured rod worth acceptance criterion has been justified for CR-3 as an appropriate value for zero power physics testing. This acceptance criterion is smaller than the typical value of ±10% for the sum of groups provided in ANSI/ANS 19.6.1-2005 Table A.1. The procedure also provides a criterion for individual rod worth differences of ±15% which is equal to that provided in the ANSI standard. Statistics have been reviewed for the comparison of total rod worth measurements to predictions for CR-3 Cycles 10 through 16 (7 data points). These provide a mean difference (p-m) of -2.65% with a standard deviation of 3.56% for the total measured rod worth. Since the rod worth measurements continue to meet their acceptance criteria, there is no reason to further validate the rod worth predictions, including that of a single partly inserted control rod assembly. The current procedure for control rod worth measurement is therefore sufficient to

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				validate the analytic core model including control rod reactivity worth.
13.4.8.2 Table 13.4 (10)	**Shutdown from Outside the Control Room** This test was performed during power ascension (greater than 10%) to demonstrate the capability to control the plant and safely bring the reactor to the hot standby condition from other than the control room location.	15%	No	**CR-3 is proposing elimination of this test.** Justification for the elimination of this test is provided in section 2.12.1.2.3.
13.4.8.2 Table 13.4 (11)	**Loss of Offsite Power** This test was performed during power ascension (greater than 10%) to demonstrate operation of station diesel generators, turbine driven emergency feedwater pump and Engineered Safeguard (ES) buses in the event of a total loss of offsite power sources.	15%	No	**CR-3 is proposing elimination of this test.** The purpose of the test was to verify that the emergency diesel generators and turbine driven EFWP would respond properly during a loss of offsite power. There have been no modifications made to the EDGs, EGD control logic, associated UV relaying or related components. During plant heatup/startup all normal EDG related maintenance and Surveillance Test procedures will be performed in accordance with the plants master surveillance plan and EDGs will be verified operable in accordance with the requirements of ITS. Emergency feedwater pumps EFP-2 and EFP-3 recirculation lines have been modified to include automatic isolation valves to increase total SG flow capability, however the EFWPs, control valves, control logic and related components are unchanged. Post modification testing will verify EFW pump auto-start capability and flow control requirements (see Table 2.12.1-4).

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				Concerning the ability to establish and control natural circulation during a LOOP, CR-3 has experience several actual LOOP events from full power during which SSCs performed as expected. In addition, an assessment of plant response to a LOOP from 100% EPU power has been performed using the RELAP5/MOD2-B&W analytical computer code. The results of this test are provided in section 2.12.1.2.3 and confirm no unexpected differences in post trip plant response. Therefore, CR-3 concludes that the power uprate has not introduced any new thermal-hydraulic phenomena or system interactions and does not invalidate the LOOP trip test as originally performed
13.4.8.2 Table 13.4 (12)	**Power Imbalance**, **Detector Correlation Test** This is to determine the relationship between the indicated out of core power distribution and the core power distribution.	40%, 75%	Yes	Controlling Procedure For Power Ascension Testing will provide the sequence for performing physics testing during the power ascension. It will also include various checks to ensure proper correlation between out of core NI and symmetrical incore detectors beginning at 15% FP and continuing until full power, equilibrium xenon conditions are achieved. If Reactor Engineering determines additional, or more frequent, monitoring is required it will be specified in the power ascension test procedure.
13.4.8.2 Table 13.4 (13)	**Nuclear Instrumentation Calibration At Power Test** To verify the ability to calibrate power range nuclear instrumentation to measured core conditions.	15%, 40% 75%, 100%	Yes	1. Performed at 30, 50, 75, and 100% power. 2. Compares power range NIs to heat balance power, and perform calibration if required. 3. Calibrate the Power Range Nuclear Instrumentation to heat balance. 4. Calibrate the power range instrumentation channels, including bistable trip setpoints.
13.4.8.2 Table 13.4 (14)	**Feedwater System Operation** To verify the ability of the Feedwater System to provide OTSG with adequate water	0 – 100% (Normal Transients)	Yes	The CR-3 “Startup Report” does not describe a standalone Feedwater System Operation test. Rather, the feedwater system performance criteria were included in the following other tests:

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
	volume under steady state and transient conditions.	40% RCP Trip 75% MFWP Trip		• Unit Load Steady State Test verified FW system parameters are within expected ranges at seven hold points between 0 and 100% power. • Turbine / Reactor Trip Test verified response following a reactor or turbine trip. • Unit load Transient Test verified (1) ICS control of FW during normal power maneuvers, (2) the delta Tc controllers ability to reratio FW flow following a RCP trip and RCP startup, and (3) The ability of the FW system to feed both steam generators following a MFWP trip. The startup and power ascension test procedure will include numerous hold points to verify plant stability and key feedwater system parameters. 1. The startup and power ascension test procedure will include a requirement for verifying key feedwater system control functions during the power ascension 2. The startup and power ascension test procedure will include a requirement for performing controlled plant transient testing during the power ascension. A primary focus of these tests is FW system response. 3. The startup and power ascension test procedure will include a requirement for performing a turbine trip transient test (with ICS runback) from 40% power
13.4.8.2 Table 13.4 (15)	**Turbine/Generator Operation** To verify proper operation of turbine/generator.	0 – 100%	Yes	Using a combination of existing plant procedures and PMTs CR-3 will verify all aspects of TG operation during initial SU and power ascension (including vibration, critical speeds, casing/differential expansion, bearing temps, oil return temperatures, EHC control functions, voltage regulator operation, generator stator and rotor temp, hot and cold gas temperature, governor valve (GV) lift sequence following expected profile, etc. Specific tests will verify operability of the following turbine protection functions: • Turbine mechanical overspeed trip mechanism

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				• Turbine electronic trip function • Overspeed protection controller (OPC) During initial startup a complete set of turbine generator baseline data will be collected. Key parameters will be monitored continuously and evaluated against applicable acceptance criteria at each test plateau.
13.4.8.2 Table 13.4 (16)	**Dropped Control Rod Test** Verify ability of unit to detect and properly respond to dropped (asymmetric) rod condition and to verify that core conditions with worst case rod dropped are within safety limits.	75%	No	**CR-3 is proposing elimination of this test.** Justification for the elimination of this test is provided in section 2.12.1.2.3.
13.4.8.2 Table 13.4 (17)	**Incore Detector Testing** Verify adequacy of the Incore Detector system to provide a description of core conditions.	40%, 75%	Yes	Controlling Procedure For Power Ascension Testing provides the sequence for performing physics testing during the power ascension. It also requires incore detector testing, Incore Neutron Detectors Channel Check at three points during the power ascension.
13.4.8.2 Table 13.4 (18)	**RCS Hot Leakage Test** To verify the ability to evaluate Reactor Coolant System (RCS) leakage and maintain within Technical Specification limits.	Unknown	Yes	A complete walkdown of RCS piping and components will be performed in accordance with Class 1 System, System Leakage Test For Inservice Inspection. Upon entering Mode 3 reactor coolant system leak rate is verified to be within Technical Specification limits in accordance with plant procedure RC System Water Inventory Balance. The leak rate calculation is repeated in Mode 1.
13.4.8.2 Table 13.4 (19)	**Pipe and Component Hanger Hot Inspection at Power** To verify the adequacy of	Various	Yes	This test/inspection was originally performed “to verify the adequacy of hangers and restraints at various power levels”. The primary reason for performing these original inspections was to identify any adverse effects resulting from overloading, flow induced vibration, thermal expansion, or

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
	hangers and restraints at various power levels.			dynamic transients (e.g., water hammer), if they were to occur. For EPU, several BOP systems include accessible lines that will experience a significant increase in their process flow rates. As part of the vibration monitoring program accessible portions of these systems will be walked down prior to EPU implementation to establish a baseline vibration state. The assessment will establish a list of target locations that warrant continued observation during the power ascension to EPU conditions. This list will include any identified historical areas of vibration concern and any location which exhibits vibration displacements of approximately 1/8 inch or greater as noted by visual observation aided by the use of simple tools such as rulers, optical wedge, spring hanger scale, etc. At locations requiring more precise displacement instrumentation such as, but not limited to, piezoelectric accelerometers will be used. Any locations selected to have data recorded with the use of instruments as part of the baseline vibration walk down will continue to be monitored with instruments throughout power ascension. During power ascension from 2609 MWt to full EPU power visual observations and instrumented data recording will be performed at the 85%, 88%, 91%, 94%, 97% and 100% EPU power levels. At each power level plateau there will be a hold for sufficient time to perform visual observations and data recording, if required, to assess the vibration response in the piping systems and at the locations identified to be of potential vibration concern. The observations and data obtained will be assessed to determine if the vibration response meets the acceptance criteria of OM Code, Part 3. Any instances of vibrations that are determined to be unacceptable will be entered into the CAP program and appropriately addressed (e.g., possible plant modifications) to assure that applicable codes and standards are met. For instances of unacceptable vibration response, qualitative evaluations (inspections) of pipe and component supports as described in ASME-OM-

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
				S/G-2007 Appendix H-3.2.3, Piping Supports, will be performed, as applicable, at the various EPU power levels. Due to the small change in temperature and the associated small change in thermal displacement in these BOP piping systems due to EPU, specific thermal expansion testing as outlined in OM Code, Part 7 is not required. Consequently, inspection of pipe and component supports for thermal expansion induced adverse effects will not be performed.
13.4.8.2 Table 13.4 (20)	**Chemical Radiochemical Tests** FSAR Table 13-1 (38) describes a chemical and radiochemical test program for the RCS, SGs, SF pools, SW and DC systems. Testing is specified from RCS fill and vent through power operation.	0 – 100%	Yes	Chemical and Radiochemical testing during startup and power ascension is defined by existing plant procedures and includes the scope of testing described in FSAR Table 13.1. 1. Nuclear Chemistry Master Scheduling Program defines specific parameters to be monitored for all modes of plant operations. 2. Primary Water Chemistry Program provides the programmatic guidance to control primary system water chemistry in order to maximize fuel and material integrity while minimizing plant radiation fields during all modes of operation. 3. Secondary Water Chemistry Program provides the programmatic guidance to minimize corrosion throughout the secondary plant and minimize fouling of the steam generators and BOP heat exchangers during all modes of plant operation.
13.4.8.2 Table 13.4 (21)	**Effluent and Effluent Monitoring** Effluents will be monitored and analyzed during power ascension specifically at 15%, 49%, 75%, and 100% rated power.	15%, 49% 75%, 100%	Yes	The CR-3 Offsite Dose Calculation Manual (ODCM) describes the effluent monitoring requirements at CR-3: 1. The CR-3 ODCM requires that plant effluent monitors are calibrated and functionally tested on a frequency consistent with NRC recommendations. These requirements are implemented in plant procedures. 2. The CR-3 ODCM requires that set-points be employed on all effluent

Table 2.12.1-3
Comparison of Proposed EPU Tests to FSAR Chapter 13 Initial Startup Testing

FSAR Section Reference	Test Description	Original Power Level	Test Plan for EPU	Evaluation / Justification
	• Monitors will be calibrated and verified for proper operation. • Setpoints will be incorporated into all monitors and valve operation will be checked. • All alarms and indicators will be tested for proper operation at predetermined setpoints and ranges.			release pathways. 3. Valve operations are verified by Operations procedures for each type of release.

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
LP Turbine Upgrade	Upgrade LP turbine sections with: 1. Inner casing 2. Guide blade carrier 3. Stationary blade rings 4. Rotor and blades 5. Modifications as necessary to support fit up.	No	No	• Verification that CO^2 fire suppression system is functional • Loop calibration and functional test of bearing temperature instruments • Verification that all turbine supervisory instrumentation (TSI) and alarms are functional • Implement PMT recommendations from turbine vendor and functional test group • Functional test all control circuits that were interrupted (because of interference) during turbine installation	• System integrity – in service leak inspection • Turbine Generator Operation Test (See section 2.12.1.2) • Main Turbine Overspeed Trip Test (See section 2.12.1.2) • Monitor for Interference (rubs) during turning gear operation and turbine roll up. • Hood spray test • Verification of non-return valve operation during turbine trip tests • Baseline data collection (vibration, expansion, bearing temperatures) during rollup and power ascension • Turbine efficiency evaluation

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
HP Turbine Upgrade	Upgrade HP turbine with: 1. Rotor and blades 2. Inner casing and blade rings 3. Modifications as necessary to support fit up. 4. Upgrade of gland steam spillover 5. EHC controller cards for turbine full arc valve control.	No	No	• Verification that CO^2 fire suppression system is functional • Loop calibration and functional test of bearing temperature instruments • Functional verification that all turbine supervisory instrumentation (TSI) and alarms are functional • Functional verification of turbine control/stop valve operability • Function testing of Gland Steam system • Implement PMT recommendations from turbine vendor and functional test group • Functional test all control circuits that were interrupted (because of interference) during turbine installation	See LP Turbine EPU Startup Tests above
Fast Cooldown System and ADV modifications	Modify the Atmospheric Dump Valves as follows: 1. Install two new larger capacity Atmospheric Dump Valves 2. Modify ADV air system to provide train separation and	Yes	Yes	• Weld inspection • Calibration of I/Ps, positioners, filter regulators, and pressure transmitters. • Configuration and setup of	• System integrity verification at full SG pressure. • Fast cooldown system operability testing • Technical specification required surveillance testing.

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
	required capacity. 3. Install dedicated independent power supplies with battery backup for each ADV control circuit. 4. Add new ADV control logic to implement “fast cooldown” function. 5. Install control room switches and indicators for FCS manual actuation. 6. Install ADV Override feature including control room switches, local lockout devices, and interface with the remote shutdown panel transfer switches. 7. Install FCS actuation Ann alarm 8. Integrate FCS with the Inadequate Core Cooling Mitigation System (ICCMS) for automatic actuation			pressure controllers and valve positioners. • FCS transfer relay functional test integrated with pressure control loop testing using variable simulated pressure to pressure transmitter. • FCS battery voltage and battery discharge capacity testing. • FCS power supplies annunciator alarm testing • Functional testing of ADV control circuit from pressure transmitter through ADV valve positioners, including operation of limit switches and their relay interfaces. • ADV air system functional test including leak testing of all tubing and verification of check valve operability. • FCS instrument loop string calibration testing including ADV stroke time verification	

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
				• Full stroke valve testing (manual and pneumatic) of ADVs and block valves • Functional test of FCS manual actuation and manual bypass. • Functional testing of FCS automatic actuation from ICCMS. • Functional test of ADV override system, including RSP transfer function. • Functional test of FCS annunciator alarms. • Setup, calibration and testing of ADV radiation monitors.	
SPDS modification	1. Install HPI flow monitoring capability and associated alarm features to SPDS 2. Add four new HPI low range flow transmitter to SPDS alphanumeric page displays. 3. Add two new FCS SG pressure transmitter inputs to SPDS alphanumeric page displays.	No	No	• Instrument loop string calibration for the four new narrow range HPI flow signals to SPDS/ICCM. • Instrument loop string calibration for the two new FCS SG pressure transmitter inputs to SPDS • Functional testing of SPDS	• Functional verification of SPDS functions and displays

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
				HPI flow monitor/alarm using simulated HPI flow and simulated RCS pressure inputs to SPDS will verify display curve and alarm features functioning	
Inadequate Core Cooling Mitigation System (ICCMS)	1. Install ICCMS cabinets (3) and power supplies, initiation, actuation and monitoring components 2. Install inputs from CRD breaker contacts to ICCMS initiate logic in channels 1 through 3 3. Modify existing T_{HOT}, T_{NCORE}, RCS pressure (Wide and low range) and HPI flow (low range) instrumentation strings for input into ICCMS channels 1 and 2. 4. Install new RCS wide range and low range instrumentation for ICCMS channel 3 5. Install new HPI low range instrumentation for ICCMS channel 3. 6. Qualify eight existing incore T/C instruments for input into ICCMS channel 3. 7. Install two independent SCM displays and two HPI flow margin monitor displays on	No	No	• Functional testing of ICCMS power supply system. • Loop testing and calibration of Incore thermocouple inputs to ICCMS. • Loop testing and calibration of RCS wide and low range pressure instrumentation string inputs to ICCMS. • Loop testing and calibration of HPI flow instrumentation inputs to ICCMS. • Loop testing and calibration of T_{HOT} instrumentation inputs to ICCMS. • Loop testing of ICCMS instrument and Online Monitor inputs to the PPC. • Functional testing of ICCMS initiate and actuation logic including	• Technical specification required surveillance testing.

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
	main control board 8. Install indicating lights showing SCM selected to T_{H}, T_{INCORE}, or Superheat 9. Install selector switches (2) for T_{HOT}/T_{INCORE} Selection 10. Install ICCMS monitoring panel on main control board. 11. Install ICCMS Online monitoring system 12. Install ICCMS annunciator alarms 13. Install two new replacement MCB RCS void trend and vessel/hot leg level recorders. 14. Install ICCMS Actuation A and B reset switches			ICCMS status panel inputs and indication. • Functional testing of HPI flow margin monitors and RCS subcooling margin monitors including SCM monitor status lights. • Functional testing of subcooling monitor selector switch functions • Functional testing of the ICCMS online monitoring system • Functional testing of ICCMS annunciator alarm inputs • Functional test of ICCMS actuation reset switches • Verification that ICCMS installation does not inhibit the ability to manually perform ICCMS trip functions (RCP trip, ISCM setpoint transfer, FCS actuation) • Verify capability to “Bypass” the ICCMS RCP trip function and enable RCP restart with the ICCMS trip	

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
				function actuated.	
LPI System Crosstie and Hot Leg Injection	Modify the LPI system piping as follows to enhance post LOCA response: 1. Remove DHV-210/211 2. Install stop-check valves DHV-510/511 3. Install LPI crosstie piping. 4. Install throttle valves DHV-500/600 5. Install Hot leg injection piping and valves 6. Install control room controls. 7. Install two independent LPI line A to B differential pressure instrumentation strings with main control board indication. 8. Install LPI line pressure instrumentation on each LPI line down stream of LPI throttle valves DHV-500/600. Output to plant computer and SPDS	No	No	• Perform seat leakage testing on drop line check valves • Perform system flow balancing via throttle valves DHV-500/600 using high accuracy instrumentation • Functional testing of MOV logic (MOVATs, limit switches, MCB indication, etc.) and valve operation from MCB • Functional testing and loop calibration of LPI differential pressure instrumentation strings. • Functional testing and loop calibration of LPI line pressure instrumentation strings. • LPI/DHR flow tests to confirm design flow requirements for LOCA and DHR. • Surveillance testing of H/L injection system • Collect data to verify hydraulic. model projection • Baseline data collection on	• System integrity verification at elevated RCS pressure. • Technical specification required surveillance testing.

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
				DHP performance	
Condensate Pump Replacement and Flow Control Modifications	Modify the Condensate System as follows to provide greater flow: 1. Replace existing variable speed condensate pumps with two new constant speed pumps and motors. 2. Install two air operated flow control valves 3. Install air operated recirculation valves. 4. Install recirculation valve position indicating light on main control board (CDP start permissive) 5. Modify condensate reject system. 6. Replace condensate demineralizer resin traps.	No	No	• Weld inspection • Control logic and stroke test all modified valves • Calibration of motor monitoring instrumentation and protective relaying • Motor interface with SC cooling • Motor Performance • Pump H/Q curve validation • Flow control valve setup and calibration • Flow control valve interface with CD flow control system • Functional testing of pump recirculation controls and indication	During the SU and power ascension verify the following: • DFT level control with each pump and both. • CD Flow response to FW flow variations (perform in conjunction with transient testing). • CD reject capability. • Pump and motor vibration within acceptable limits over the full operating range. • Baseline Data (Key pump and system parameters, pump amps vs. flow, AOV pos vs. flow vs. power, pump operating temperatures, CX demineralizer resin trap ΔP). • Motor amps and 4160 unit bus loading. • Pump H/Q curve validation

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
FW Booster Pump Replacement and recirculation line.	Install new main feedwater booster pumps, motors and skid mounted auxiliary equipment to meet EPU flow requirements with improved margin. Install larger booster pump recirculation lines and valves to accommodate increased recirculation flow.	Yes	Yes	• Calibration of monitoring instrumentation and protective relaying • Functional check of LO system and control interface • Functional check seal water system pressure control • Motor Performance • Pump H/Q curve validation • Baseline data (key pump and system parameters, pump amps/flow, pump temperature)	• System integrity – in service leak inspection • Vibration monitoring • Verify recirculation control valve operation • Verification of pump cooling water control operation • Verification of LO system operation • Baseline Data (key pump and system parameters, pump amps/flow, pump temperature) • Motor amps and 4160 unit bus loading.
Main Feedwater Pump Replacement	Install new Main Feedwater Pumps to meet EPU flow requirements with improved margin. Install a new MFWP high discharge pressure trip to provide FW system over pressure protection.	Yes	Yes	• Functional test of over pressure trip function • Calibration of monitoring instrumentation • Functional check of LO system • Functional check seal water system pressure control	• System integrity – in service leak inspection • Vibration monitoring • Overspeed trip testing • Baseline data (flow, pressure, operating temperatures) • Pump H/Q curve validation/Verification of pump control response in D/P and flow control modes • Verify recirculation control valve operation

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
Replace MFWP suction Valves (FWV-14 and FWV-15)	Replace main feedwater pump suction isolation valves	No	No	• MOV control logic and MOVATS setup/test • Stroke from control room verify position indication • Verify interlock with MFWP latch/trip circuit • In-service leak inspection • Stroke time test • Verify operability by testing MS/MF isolation function.	No Additional Testing Required
Deaerator Rerate and Bypass System	Modify Condensate/Feedwater system as follows to provide greater flow: 1. Rerate the deaerator (FWHE-1) and deaerator storage tank (FWT-1) for higher thermal-hydraulic conditions. 2. Install piping and manual flow control valves to bypass a portion of flow around the deaerator feed tank. 3. Install a sample point downstream of last stage FWHEs for O_2 control. 4. Install new ANN alarm for high condensate flow to deaerator.	No	No	• In-service leak inspection • Set Relief valves • Loop check associated instruments and level controls • Verify flow alarm operability	• Monitor DFT temperature, pressure, and level vs. power • Perform initial deaerator bypass "startup" (Bypass valve position adjustment) • Verify procedure adequacy and expected plant response. • Monitor dissolved O_2 concentration and operation of the chemical injection system.

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
EFW System Modifications	Modify EFW system as follows: 1. Add power operated recirculation line automatic isolation valves to EFP-2 and EFP-3 recirculation lines. 2. Add recirculation line isolation valve control switches and indicator lights to main control board. 3. Remove redundant EF flow indication from the MCB. 4. Modify low fuel oil alarm setpoint for EFP-3 5. Add annunciator event points to alert operator of EFP recirculation valve trouble	No	No	• In-service leak inspection • Functional testing and loop calibration of new EFW flow instrumentation and recirculation valve control circuits. • Functional test of EFP-3 low fuel level alarm • Functional test EFW recirculation valve trouble alarms	• Perform EFP- 2 and 3 design flow verification • Functional testing of pump recirculation controls and indication • Perform EFP- 2 and 3 vibration monitoring • Obtain baseline data
High Pressure Injection System modifications	1. Install four new HPI low range flow transmitters 2. Reposition HPI throttle valves to achieve higher HPI flow rate.	No	No	• Functional testing and loop calibration of new HPI Low range flow instrumentation • Perform system flow balancing via throttle valves MUV-590, 591, 592, 593 • Verify HPI flow inputs to ICCMS and SPDS	No Additional Testing Required

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
Makeup Tank Bypass Injection Line	Modify the MU system to divert letdown flow (and boric acid injection flow) to MUP suction.	No	No	• In-service leak inspection • Check air system for leaks with air applied to valve • Loop check valve controls • Stroke check MUT bypass valve (MUV-661) from MCB • Verify failure mode on loss of air	• System integrity • Functional flow test
ICS / AULD Scaling	Modify the Integrated Control System as follows: 1. Document Maximum Continuous Rating (MCR) for EPU conditions. 2. Rescale ICS tuning parameters based on post-EPU MCR. 3. Modify runback rates and endpoints 4. Upgrade AULD for EPU conditions. 5. Eliminate automatic asymmetric rod runback	Yes	Yes	• Calibrate ICS modules to IDS specs • Perform Pre-Operational Check of the Integrated Control System	• Integrated Control System Functional Testing (See section 2.12.1.2) • Verify key ICS control function (LLL, MBV, MFWP d/p control, HDR pressure control) perform as expected. • Main Turbine Trip Test from < 40% EPU Power (See section 2.12.1.2)

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
Replace fifth stage Feedwater Heaters (FWHE-2A/2B)	Replace FWHE-2A/B to accommodate the higher discharge pressure from the new FW booster pumps and increased FW flow to support EPU.	No	No	• Weld Inspection • Functional test of all manual valves • In-service leak inspection • Verify setpoint of tube side RVs • Calibrate and functional test instrumentation and valve controllers	• Monitor key performance parameters during power ascension • Perform level control system tuning (if required) • Verify proper operation of the heater drain system • Monitor FW system piping for abnormal movement or vibration • In-service leak test of all welds and flanges
Replace sixth stage Feedwater Heaters (FWHE-3A/3B)	Replace FWHE-3A/3B to accommodate the higher discharge pressure from the Main Feedwater Pumps and increased FW flow to support EPU.	No	No	• Weld Inspection • Functional test of all manual valves • In-service leak inspection • Calibrate and functional test instrumentation and valve controllers	• Monitor key performance parameters during power ascension • Verify proper operation of the heater drain system • Perform level control system tuning (if required) • Monitor FW system piping for abnormal movement or vibration • In-service leak test of all welds and flanges

Table 2.12.1-4
R17 Plant Modification Testing

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	Post Modification Test	EPU Startup Test
Qualification of OTSG for EPU operation	Provides the Replacement Once Through Steam Generator (OTSGs) qualification for operation at 3030 MWt (3014 MWt core power plus 16 MWt added by the Reactor Coolant Pumps). This change includes the implementation steps required to reset the OTSG flow orifices as required for operation at EPU.	Yes	Yes		• In-service leak inspection of OTSG handhole covers at full temperature and pressure. • Monitor key performance parameters during startup and power ascension • Verify full power FW flow and operating level remains within procedural requirements • Validate heat balance power against recorded flow and core ΔT measurements; determine if SP-312A revision is required • Verify RB temperature remains within procedural requirements

Table 2.12.1-5
Additional Testing for Previously Installed EPU Modifications

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	EPU Startup Test
Modification Performed During Refuel 16				
OTSG Replacement	Replace both steam generators during Refuel 16. Perform 100% eddy current testing of OTSG tubes.	Yes	Yes	• Record baseline vibration data using the installed Loose Parts Monitoring System (LPMS) accelerometers at specified intervals during Cycle 17 startup, power ascension and steady state full power operation. • During Cycle 18 record vibration data using the installed LPMS accelerometers at specified intervals during startup, power ascension and steady state full power operation. Compare SG vibration data with Cycle 17 baseline data and evaluate any anomalies. • Establish new baseline data for operation above 2609 MWt.
Turbine Bypass valve Replacement	Modify the Turbine Bypass Valves as follows to increase bypass capacity: 1. Install four new larger capacity Turbine Bypass Valves. 2. Modify TBV positioner calibration to establish staggered operation.	Yes	Yes	• Monitor TBV response during startup • Verify TBV overlap • Monitor TBV response during Turbine Trip Test • Monitor TBV piping for abnormal movement or vibration
Condensate Heater CDHE-3A/3B Replacement	Replace CDHE-3A and 3B to support operation at EPU conditions.	No	No	• Monitor key performance parameters as power is increased above 2609 MWt • Verify proper operation of the HD system • Monitor CD system piping for abnormal movement or vibration
Secondary Cooling Heat Exchanger Replacement	Replace SCHE-1A and 1B to support operation at EPU conditions.	No	No	• Monitor key performance parameters on SC and CW sides of the SCHEs as power is increased above 2609 MWt

Table 2.12.1-5
Additional Testing for Previously Installed EPU Modifications

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	EPU Startup Test
Secondary Cooling Pump Impeller and Motor Upgrade	Modify the Secondary Services cooling pumps as follows: 1. Replace SCP-1A and 1B (pump and motor) to support operation at EPU conditions. 2. Replace pressure control system and revise operating pressure setpoint 3. Replace relief valves.	No	No	Perform the following as power is increased above 2609 MWt • Monitor key SC system performance parameters • Verify SC system pressure regulating valve properly controls system pressure. • Verify H2 cooler flow control valve properly controls generator cold gas temperature
Moisture Separator Reheater Replacement	Replace four Moisture Separator Reheaters to support operation at EPU conditions.	No	No	Perform the following as power is increased above 2609 MWt • Monitor key RH and HD system performance parameters • Monitor RH and HD system piping for abnormal movement or vibration
Moisture Separator Reheater belly drain heat exchangers	Modify condensate and heater drain systems as follows improve plant efficiency: 1. Install two new condensate heat exchangers to recover MSR belly drain heat. 2. Install CD system piping and valves to control flow. 3. Modify MSR HD system to piping and controls to support operation of the new heat exchangers . 4. Install interlock feature to transfer HD	No	No	• Monitor key performance parameters as power is increased above 2609 MWt • Verify CD system total flow and flow balance meets expectations as power is increased above 2609 MWt. • Verify MSR belly drain level control operates as designed as power is increased above 2609 • Verify revised procedure guidance for placing the CDHEs in service. • Test HD low flow interlock interlock function • Monitor MSR drain system piping for abnormal movement or vibration

Table 2.12.1-5
Additional Testing for Previously Installed EPU Modifications

Modification	Description	Impact Transient Response	Modeled in Transient Analysis	EPU Startup Test
	flow to condenser during a CD system transient.			
Iso-phase Bus Duct Cooler Replacement	Replace Iso-phase Bus Duct Cooler to support operation at EPU conditions.	No	No	Perform the following as power is increased above 2609 MWt • Monitor key system performance parameters
Main Turbine lube oil Cooler Replacement	Turbine Lube Oil Cooler tube bundle was replaced due to increased heat loads generated as a result of power uprate.	No	No	Perform the following as power is increased above 2609 MWt • Monitor key turbine lube oil system performance parameters
Main Generator and Exciter Replacement	Replace Main Generator, including hydrogen coolers to support operation at EPU conditions Replace generator Exciter and exciter cooler to support operation at EPU conditions.	No	No	Perform the following as power is increased above 2609 MWt • Monitor key system performance parameters • Verify voltage regulator performance • Verify H2 cooler performance (gas temperature, Stator / rotor temperatures) • Verify Exciter cooler performance • Seal oil system parameters • Generator condition monitor
Modification Performed During Refuel 15				
Step Up Transformer Replacement	Replace Step Up Transformers (3) to support operation at EPU conditions.	No	No	Perform the following as power is increased above 2609 MWt • Monitor key Step Up transformer performance parameters

Table 2.12.1-6
Plant Transient Tests in CR-3 Power Ascension Test plan

Proposed Test	Description	Purpose
Integrated Control System Functional Testing	In accordance with Response Testing of the ICS at Power, introduce small step changes in ICS control signals, and monitor the resulting ICS control system response using the plant computer. Evaluate and document important ICS control functions such as TBV position/demand vs. power, SUCV/LLCV overlap, LLBV and MBV operation vs. power, MFWP response during MBV operation, condensate flow vs. FW flow correlation and condensate system/deaerator level response during load changes.	Verify proper operation of the following ICS control functions: SG level control, turbine bypass valve operation, main feedwater control (in both d/p and flow control modes), delta Tc controller operation, turbine header pressure control, and T_{AVG} control (by both the reactor and feedwater subsystem). Collect a new set of ICS baseline data.
Turbine/Generator Operation Test	Turbine generator operational testing will include required vendor prescribed testing, various post modification function tests, normal startup / surveillance testing and several Power Ascension Test requirements.	Verify all aspects of TG operation during initial SU and power ascension (including vibration, critical speeds, casing/differential expansion, bearing temps, oil return temperatures, EHC control functions, voltage regulator operation, generator stator and rotor temp, hot and cold gas temperature, etc. Verify operability of the turbine mechanical overspeed trip mechanism and the electronic overspeed protection controller (OPC), and other front pedestal turbine trips. Verify governor valve (GV) lift sequence follows expected profile. Collect new set of turbine generator baseline data.

Table 2.12.1-6
Plant Transient Tests in CR-3 Power Ascension Test plan

Proposed Test	Description	Purpose
Main Turbine Overspeed Trip Test	From a steady state power level of between 10 and 15% EPU rated power with the main turbine at synchronous speed, turbine speed will be increased above the mechanical overspeed trip setpoint to verify the operability of the mechanical overspeed trip mechanism and related components.	Verify the operability of the mechanical overspeed trip mechanism, the main turbine trip valves (GVs, TVs, RHSVs, and RHIVs), and the TBV response (TBVs will be partially open and controlling header pressure at the start of this test).
Main Turbine Trip from < 40% EPU Power	From a steady state power of < 40% EPU rated power the main turbine will be manually tripped from the control room. The main generator will trip and Integrated Control System (ICS) will reduce reactor power to approximately 25% of EPU power and transfer header pressure control to the turbine bypass valves.	Verify proper operation of the ICS, the turbine electronic trip mechanism, the turbine trip valves (GVs, TVs, RHSVs, and RHIVs) and the turbine bypass valves (TBVs will initially be closed). Plant control system functions such as header pressure control, deaerator level control, RCS pressure control, pressurizer level control, and T_{AVG} control will be monitored in response to the transient.

Table 2.12.1-6
Plant Transient Tests in CR-3 Power Ascension Test plan

Proposed Test	Description	Purpose
10% Load Change from 35% EPU Power	From a steady-state power condition at approximately 35% EPU the ICS unit load demand (ULD) will be decreased from 35% to 25% RTP at a rapid rate (to be determined based on allowable maneuvering rates). When acceptable dynamic performance has been verified, power will be ramped back to 35%.	Evaluate the dynamic behavior of the various plant control systems against Level 1 and Level 2 acceptance criteria to ensure that the combination of increased power and changes to the plant configuration (EPU modifications) have not resulted in an unacceptable aggregate impact. This test will evaluate the transient response of the ICS and ICS controlled components during a rapid power maneuver with the ICS in the d/p control mode (MFWPs controlling d/p, FW control valves controlling flow). Transient response in the flow control mode will be performed in a separate test.
Load Change 35% to Main Block Valves OPEN	From ~35% EPU power ICS unit load demand will be slowly increased until the Main Block Valves open. During this evolution the ICS FW subsystem will transition from d/p control mode to flow control mode.	Evaluate the dynamic response of the ICS during the transition from the d/p control mode to the flow control mode. (Test will verify (1) LLCV have adequate control margin when MBV open, (2) MBV opening logic, (3) MFWP smoothly transitions from d/p to flow control as MBVs open, (4) LLCVs freeze following transfer.

Table 2.12.1-6
Plant Transient Tests in CR-3 Power Ascension Test plan

Proposed Test	Description	Purpose
Load Change from Main Block Valve Open to Main Block Valve Closed	With the Main Block Valves open and the plant on ICS flow control ICS unit load demand will be slowly decreased until the Main Block Valves close. During this evolution the ICS FW subsystem will transition from the flow control mode to the d/p control mode.	Evaluate the dynamic response of the ICS during the transitions from the flow control mode to the d/p control mode. Test will verify (1) LLCV have adequate control margin when MBV close, (2) MBV close logic, (3) MFWP smoothly transitions from flow control to d/p control as MBVs close, (4) LLCVs assume MFW flow control.
10% Load Change at 100% EPU Power	From approximately 100% RTP the ICS unit load demand will be decreased from 100% to 90% EPU power at the maximum allowable rate (to be determined based on allowable maneuvering rates). When plant response has been evaluated, power will be ramped back to 100% at a normal maneuvering rate.	Evaluate the dynamic behavior of the various plant control systems against Level 1 and Level 2 acceptance criteria to ensure that the combination of increased power and changes to the plant configuration (EPU modifications) have not resulted in an unacceptable aggregate impact. By conducting this test between 100% and 90% EPU power the ICS will be controlling MFW flow in the flow control mode (LLCV frozen, flow error controlling MFWP speed). For the power reduction test the maximum allowable rate was selected because plant procedures allow the use of that rate during rapid power reductions.

Table 2.12.1-7
Operating Experience

Operating Experience Description	Lessons Learned
NEI 08-10 NEI 08-10 - 6.4.2 OE Recommendations - NEI compiled the following recommendations and best practices for a successful startup and power ascension ramp to full uprated power levels. • Startup Test Matrix - A detailed startup-testing matrix should be developed identifying the testing to be performed at each power level. • Overall Controlling Procedure - One controlling procedure should be written based on the matrix to direct the testing and ensure that all tests are performed in the correct sequence. • Qualifications for Test Director - Based on industry experience it is recommended that the power ascension test director should be a very experienced Senior Reactor Operator and should be experienced in performing control system testing. This is key to successful pressure and feedwater level control testing. • 'Power Moves' on Same Shift, Same Crew - Where possible, use of the same shift, same operating crews and data-takers for testing at each power plateau is recommended. This will ensure consistent results. • Preparation of Test Procedure - It is recommended that the testing procedure writer be a member of the test team to simplify procedural changes when required. • Training - Consider Just In Time training on the simulator for engineers and operators involved with the testing.	CR-3 has reviewed the recommendation contained in NEI 08-01 and will incorporate all of them in the startup and power ascension test program. The EPU power ascension test director will be a licensed (or previously Licensed) SRO with prior testing experience. A comprehensive startup test matrix will be developed to properly sequence the startup test program. Key activities from the matrix will be integrated into the plant startup schedule with appropriate prerequisite activities and milestones attached. A controlling procedure listing all startup and power ascension test requirements will be developed, and if practical the procedure writer will be included on the test team. An appropriate level of training on portions of the power ascension procedure that involve new activities, or activities that have been significantly altered by EPU modifications will be provided to operations and engineering personnel.

Table 2.12.1-7
Operating Experience

Operating Experience Description	Lessons Learned
NEI 08-10 NEI 08-10 – Section 6.4.4, Plant Baseline Data, recommends that a historical MUR plant performance data base be established and used to evaluate the effects of EPU modifications. The plant baseline data should be obtained prior to installing plant modifications needed to support operation at the increased power levels. Post modification comparisons should be made immediately after the plant modifications are installed and prior to operating the plant at EPU conditions.	A comprehensive data base of pre-EPU plant and component performance data will be established. The data base will be comprised of: (1) data collected via during the conduct of normal surveillance testing, (2) steady state and transient operational data collected via plant computer groups and the plants RECALL system, and (3) data collected specifically for the purpose of providing an operational benchmark for comparison during startup and power ascension (FIV data, radiation dose levels at pre-selected monitoring points, running loads on plant electrical busses, etc.). Baseline data will be used in conjunction with other design inputs to establish the normal operating windows and specific acceptance criteria for plant system and component.
INPO OED 2008-07 INPO OED 2008-07 documents several examples where Insufficient maintenance work practices and engineering evaluations on auxiliary feedwater (AFW) system components have resulted in bearing high temperatures and damage. Applicability of this OE is to post modification testing of FWP-1A and 1B, FWP-2A and 2B and CDP-1A and 1A.	The following “lessons learned” will be factored into the functional test procedures and the startup and power ascension test procedure steps associated with the testing of the modified MFWP and replacement FWBPs, and CDPs. • Ensure critical vendor information, including operational requirements, parameter operating ranges, and operational limits are incorporated into functional test procedure and normal plant procedures. • Ensure functional test procedure verifies all aspects of pump operation necessary to demonstrate operability. • Ensure functional test includes all required parameters to verify operability PLUS any additional parameters that would indicate potential performance anomalies within the boundaries of acceptable operation. • Ensure test runs require sufficient flow rates and are of sufficient duration to allow key parameters to stabilize at operational levels.
INPO OED 2007-11 This INPO OE Digest documents several examples where unplanned turbine trips occurred during turbine testing as a result of (1) test personnel being unable to recognize if a turbine trip signal has cleared before coming out of “trip bypass”, and (2) test procedures that provide insufficient detail or fail to provide specific guidance for dealing with unexpected conditions, including how to	The following “lessons learned” will be factored into the functional test procedures associated with main turbine and generator operation. • During initial turbine roll up and startup, Turbine Supervisory Instrumentation (TSI) will be in actual operation for the first time on the new turbine. Functional test procedures will verify all TSI is performing properly during startup. If necessary, alternate monitoring methods will be employed until installed TSI is verified operational.

Table 2.12.1-7
Operating Experience

Operating Experience Description	Lessons Learned
"back out" of the procedure when expected response in not obtained.	• Ensure test personnel are familiar with expected TSI responses and are provided with specific guidance if TSI limits are exceeded. • Ensure each functional test procedure includes actions required to properly verify the status of the main turbine trip block. • Ensure the Functional test procedures provide the appropriate level of detail including contingency guidance for unexpected problems.
INPO JIT-038 This INPO JIT documents several examples where plant transients and inoperable equipment have resulted from poorly planned or executed post modification testing. JIT-038 provides 13 specific screening questions that can assist the procedure developer and reviewing authorities in identifying and avoiding error traps that are common to the post modification test process.	The Lessons learned in JIT-038 are presented in the form of a series of questions to be considered by the functional test procedure developer, technical reviewers and the procedure approval authority. CR-3 will require functional test procedure developers, technical reviewers and the approval authorities to screen each functional test procedure against the listed criteria.
Vibration Monitoring Program Operating Experience (OE) from other nuclear plants implementing power upgrades has and will continue to be factored into the CR-3 Piping Vibration Monitoring Program as each aspect of implementation is executed.	• One prominent OE recommendation obtained from the Brunswick Nuclear Plant and being used at CR-3 is to compare the flow changes in the various piping runs as predicted by the steam cycle performance code to identify and rank flow paths with the highest change in fluid velocity. This provides a systematic way of screening candidate piping runs and prioritizing initial surveillance activities. • Several different sources of OE influenced the selection of data acquisition equipment. CR-3 has installed state of the art equipment that is capable of recording and correlating data streams from multiple channels to be able to efficiently identify and quantify cause and effect interrelationships. This high speed equipment also provides the ability to monitor systems during transients where manual measurements would not be feasible. A unique transducer mounting block designed and tested for CR-3 provides the option of easily re-tasking a data point to another piping location depending upon the findings."

Table 2.12.1-8
CR-3 Extended Power Uprate Power Ascension Test Plan

Test Description	Prior to Startup	RTP in % of EPU Power 3014 MWt (Allowance +0%, -5%)												Allowance +0%, -1%					
		0	5	10	15	20	25	30	40	50	60	75	80	85	88	91	94	97	100
Power Ascension Test Plateaus Hold points to verify key plant parameters, steady state and transient data meets pre-determined acceptance criteria and SSCs are performing as expected.		X		X			X		X		X		X	X	X	X	X	X	X
Control Rod Trip Test Verify proper CRD and RPS functioning and that rod drop times meet safety requirements and that APSRs do not trip.	X																		
Chemical and Radiochemical Tests	X																		
Vibration Monitoring Take baseline data under steady state and transient conditions				X				X		X			X	X	X	X	X	X	X
Calibration and Neutron Response of Source Range Monitoring	X	X																	
Control Rod Group Calibration Determine differential and integral reactivity worth.		X																	
Boron Worth Calibration To determine the boron reactivity worth.		X																	
Moderator Temperature Coefficient Measurement		X																	
Excess Reactivity Measurement To measure the excess reactivity of the core to confirm the predicted shutdown margin.	X	X																	

Table 2.12.1-8
CR-3 Extended Power Uprate Power Ascension Test Plan

Test Description	Prior to Startup	RTP in % of EPU Power 3014 MWt (Allowance +0%, -5%)												Allowance +0%, -1%					
		0	5	10	15	20	25	30	40	50	60	75	80	85	88	91	94	97	100
Nuclear Instrumentation Check To verify the overlap of the source and intermediate range nuclear instrumentation at zero power.	X	X	X																
Biological Shield Survey To confirm adequacy of radiation shielding to zero and low power levels.		X		X			X		X		X		X		X		X		X
Integrated Control System Test To determine unit ability to maintain stability during steady state and load transient conditions.		X		X			X	X	X	X				X	X	X	X	X	X
Unit Load Steady State and Transient Test To determine unit behavior at steady state and response to load changes.							X	X	X	X				X					X
Feedwater System Operation To verify the ability of the Feedwater System to provide OTSG with adequate water volume under steady state and transient conditions.		X					X		X		X			X	X	X	X	X	X
Reactivity Coefficient at Power To determine reactivity coefficient.																			X
Unit Heat Balance To verify the adequacy of computer primary and secondary heat balance calculations.					X		X			X		X		X					X
Core Power Distribution To measure core flux, power distribution and DNBR during power ascension.									Test 1 – between 40% and 80% power					Test 2 – between 85% and 100% power					

Table 2.12.1-8
CR-3 Extended Power Uprate Power Ascension Test Plan

Test Description	Prior to Startup	RTP in % of EPU Power 3014 MWt (Allowance +0%, -5%)												Allowance +0%, -1%					
		0	5	10	15	20	25	30	40	50	60	75	80	85	88	91	94	97	100
Power Imbalance, **Detector Correlation Test** Determine the relationship between the out of core power distribution and the core power distribution.					Multiple tests between 15% and 100% full power														
Nuclear Instrumentation Calibration At Power Test To verify the ability to calibrate power range nuclear instrumentation to core thermal power.								X		X		X		X					X
Turbine/Generator Operation To verify proper operation of the main turbine/generator.	X			Functional tests during rollup/startup - continuous monitoring between 10% and 100% full power															
Incore Detector testing Verify adequacy of the Incore Detector system to provide a description of core conditions.					X				X					X					
RCS Hot Leakage Test To verify the ability to evaluate Reactor Coolant System (RCS) leakage and maintain within ITS limits.	X																		X
Pipe and Component Hanger Hot Inspection at Power To verify the adequacy of hangers and restraints at various power levels.		Continuous between 0% and 100% full power																	
Plant Radiation Surveys Verify expected dose rates		Continuous between 0% and 100% full power																	
Main Turbine Overspeed Trip Test Verify the mechanical overspeed trip mechanism and TSV response.					X														

Table 2.12.1-8
CR-3 Extended Power Uprate Power Ascension Test Plan

Test Description	Prior to Startup	RTP in % of EPU Power 3014 MWt (Allowance +0%, -5%)												Allowance +0%, -1%					
		0	**5**	**10**	**15**	**20**	**25**	**30**	**40**	**50**	**60**	**75**	**80**	**85**	**88**	**91**	**94**	**97**	**100**
Main Turbine Trip from 40% RTP Power Verify turbine electronic trip mechanism and plant control system during runback.									X										
10% Load Change at 35% RTP Power Verify plant transient response with FW system in the d/p control mode.								X	X										
10% Load Change at 100% RTP Power Verify plant transient response with FW system in the flow control mode.															X				X
Load Change 35% to Main Block Valves open Verify proper plant transient response during the transition from d/p control mode to Flow control mode.								X	X										
Load Change From Main Block Valves Open to Main Block Valves closed Verify proper plant transient response during the transition from flow control mode to d/p control mode.								X	X										
Primary System Parameter Data Collection Continuous real time data acquisition and monitoring.	Continuous monitoring from plant heatup through 100% full power																		
Secondary System Data Collection Continuous real time data acquisition and monitoring.	Continuous monitoring from plant heatup through 100% full power																		

Table 2.12.1-8
CR-3 Extended Power Uprate Power Ascension Test Plan

Test Description	Prior to Startup	RTP in % of EPU Power 3014 MWt (Allowance +0%, -5%)												Allowance +0%, -1%					
		0	5	10	15	20	25	30	40	50	60	75	80	85	88	91	94	97	100
Condensate System Functional Testing Various tests to verify proper CD system operation over the complete load range.	Continuous monitoring from plant heatup through 100% full power																		

2.12.2 Transient Performance

The NSSS instrumentation and control systems are required to respond to the initiation of design basis plant operational transients without initiating a reactor trip or engineered safety features signal. An evaluation of the CR-3 NSSS instrumentation and control systems response to design basis and selected operational transients at the EPU conditions was performed to ensure the plant response remains acceptable.

2.12.2.1 Regulatory Evaluation

CR-3 Current Licensing Basis

FSAR Section 4.1 defines the current design basis operational transients that CR-3 must be able to sustain without initiating relief valve or turbine bypass valve (TBV) actuation. These are:

- Step load changes – increasing load steps of 10% of full scale unit load demand (1000 MWe) in the range between 20% and 90% of the maximum continuous rating (MCR) of 914 MWe, and decreasing loads steps of 10% of full scale unit load demand between 100% and 20% of the MCR.
- Ramp load changes – increasing load ramps of 10% of the full scale unit load demand per minute in the range between 20% and 90% of the MCR, or decreasing load ramps of 10% full scale unit load demand per minute from 100% to 15% of the MCR. From 15% to 20% of the MCR and from 90% to 100% of the MCR, increasing ramp load changes of 5% of the MCR per minute.

2.12.2.2 Technical Evaluation

Introduction

Evaluations of the design basis transients were performed using EPU NSSS conditions in order to demonstrate adequate margin exists to relevant setpoints over the entire range of the EPU operating conditions to preclude initiating relief valve or turbine bypass valve actuation. The EPU operating conditions are shown in Table 1.1-1, Nuclear Steam Supply System Parameters.

Description of Analyses and Evaluations

Design basis operational transient evaluations were performed using CR-3 design transients and applying expected plant responses at EPU conditions to those of the MUR. AREVA NP's simulation code Digital PowerTrain (DPT) was used to predict the performance and behavior of the major components of the NSSS for various plant conditions and operation. This computer code is a system-level program code and models the overall NSSS, including the detailed modeling of the ICS, RPS, makeup, relief, and pressure controls. The DPT model was developed for the generic B&W lower loop nuclear plant. A comparison of the CR-3 steam generator design was made to the design included in the DPT model. From an operational perspective, the steam generator designs are functionally the same such that the generic DPT model can be used to assess the CR-3 EPU operational transient response.

Results

Design operational transients were analyzed using the DPT model for both the Pre EPU and the Post EPU core power (plus pump heat) of 2584 MWt and 3031 MWt respectively. The power level of 2584 MWt was used (instead of the MUR power level of 2609, plus pump heat) because these simulations were readily available. This has no impact on the trend of the transient response, nor the conclusions, presented within this section.

A comparison of the transient curves for the plant at the two power levels showed that the differences generally reflected the higher T_{HOT}, lower T_{COLD}, higher feed and higher steam flow rates that accompany operation at the higher rated power. The DPT model was used to determine whether there are any significant, unanticipated performance differences between the plant at current conditions, and those projected at the EPU conditions. The plant response differences between the current and projected EPU power levels confirmed acceptable results.

10% Step Power Change Transients

10% Step Load Increase

The 10% step load increase is permitted in the power range between 20% and 90% power (i.e. the last step load increase is from 80% to 90% power). The DPT simulation evaluations assessed the step increase in load from 80% to 90% power at nominal and EPU power conditions. See Figures 2.12.2-1, 2.12.2-2, and 2.12.2-3.

The results of the DPT model simulation for the 10% step load increase transient indicate the following:

- There is no appreciable increase in RCS pressure response, and therefore, no challenge to the power operated relief valve (PORV) or code safety valves due to the transient initiation;

- The steam generator outlet pressure, biased higher at the EPU conditions, essentially maintains the initial pressure difference during the transient, and the TBV are not actuated.

- The plant thermal-hydraulic responses are consistent with those expected (as described above) and predictable at the EPU conditions.

The acceptance criteria defined in FSAR Section 4.1 states that neither the PORV (i.e., “relief valve”) or turbine bypass valves be actuated in the performance of this operational transient. Therefore, the plant response for the 10% step load increase is acceptable for the EPU conditions.

10% Step Load Decrease

The 10% step load decrease transient is permitted in the power range between 100% and 20% power (i.e., the last step load decrease is from 30% to 20% power). The DPT simulation evaluations assessed the step decrease in load at the most limiting power range of 100% to 90% power. See Figures 2.12.2-4, 2.12.2-5, and 2.12.2-6.

The results of the DPT simulation for the 10% step load decrease transient indicate the following:

- There is no appreciable difference in RCS pressure response between the nominal power level and that of the EPU;

- The RC pressure response does not actuate the PORV or code safety valves due to the system response to the transient;

- The steam generator outlet pressure, biased higher at the EPU conditions, essentially maintains the initial bias throughout the transient, and comes to equilibrium at the initial setpoint. There is no TBV actuation; and

- The plant thermal-hydraulic responses are consistent with those expected (as described above) and predictable at the EPU conditions.

The acceptance criteria defined in FSAR Section 4.1 states that neither the PORV (i.e., "relief valve") or turbine bypass valves be actuated in the performance of this operational transient. Therefore, the plant response for the 10% step load decrease transient is acceptable for the EPU.

<u>Ramp Power Change Transients</u>

<u>Load Decrease from 100% to 8%</u>

This transient is the design power unloading cycle. The transient is defined with the system power starting at 100% power, decreasing at rates up to 10% per minute between 100% and 20% power, and rates up to 5% per minute between 20% and 13% power. In the range from 13% to 8% power, the reactor power is manually decreased at rates up to 3/4% per minute. See Figures 2.12.2-7, 2.12.2-8, and 2.12.2-9.

The DPT simulation analyzes the ramp power change from 100% power to approximately 20% power at a rate of 10% per minute over a period of 8 minutes. Power is held constant for 200 seconds when the rods are jogged in at a rate to simulate 0.75% per minute down to 8% power. Steady-state is then achieved.

The results of the DPT simulation for the 100% to 8% ramp load decrease transient indicate the following:

- RCS pressures do not actuate the PORV or code safety valves;

- The secondary steam pressure and flow rates are controllable as expected for the EPU conditions and there is no TBV actuation; and

- The plant thermal-hydraulic responses are consistent with those expected (as described above) and predictable at the EPU conditions.

The acceptance criteria defined in FSAR Section 4.1 states that neither the PORV (i.e., "relief valve") or turbine bypass valves be actuated in the performance of this operational transient. Therefore, the plant response for the ramp load decrease transient is acceptable for the EPU.

<u>Load Increase from 8% to 100% power</u>

This transient is the design power loading cycle. The transient is defined with the system power starting at approximately 8% power. The reactor power is manually increased to 13% power at a rate of 3/4% per minute, then placed in automatic control and power is then increased at rates up to 5% per minute between 13% and 20% power, and at a rate of 10% per minute between 20% and 90% power, and at a rate of 5% per minute between 90% and 100% power. See Figures 2.12.2-7, 2.12.2-8, and 2.12.2-9.

The DPT simulation analyzes the ramp power change from 8% power to approximately 20% power at a rate of 0.75% per minute. Power is held constant for 200 seconds when the system is put into automatic control and the power is increased to 100% at a rate of 10% per minute. Steady-state is then achieved.

The results of the DPT simulation for the 8% to 100% ramp load increase transient indicate the following:

- RCS pressures do not actuate the PORV or code safety valves;
- The secondary steam pressure and flow rates are controllable as expected for the EPU conditions and there is no TBV actuation; and
- The plant thermal-hydraulic responses are consistent with those expected (as described above) and predictable at EPU conditions.

The acceptance criteria defined in FSAR Section 4.1 states that neither the PORV (i.e., “relief valve”) or turbine bypass valves be actuated in the performance of this operational transient. Therefore, the plant response for the ramp load increase transient is acceptable for the EPU.

2.12.2.3 Conclusion

The effects of the proposed EPU on the plant capability of meeting its response to design basis operational transients have been evaluated. The acceptance criteria of no TBV or PORV (i.e., “relief valve”) actuation during the performance of these events have been demonstrated with these simulations. It is concluded that the effects of the proposed EPU on the plant operational capability will not be significant and that the changes that are necessary to achieve satisfactory results at EPU are consistent with the plant's design basis.

Therefore, these simulations indicate that the responses of the plant to operational transients at the proposed EPU are acceptable with respect to the plant capability of meeting its design basis operational transients and the plant will continue to meet the requirements of FSAR Section 4.1.

2.12.2.4 References

None

Figure 2.12.2-1: 10% Step Load Increase Power Response

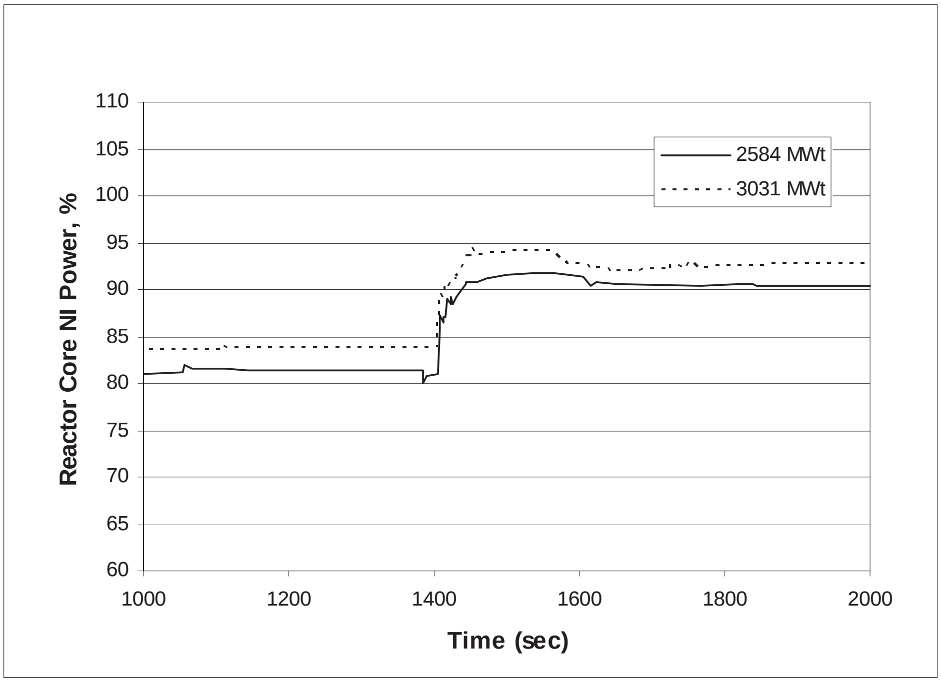


Figure 2.12.2-2: Step Load Increase Primary Pressure Response

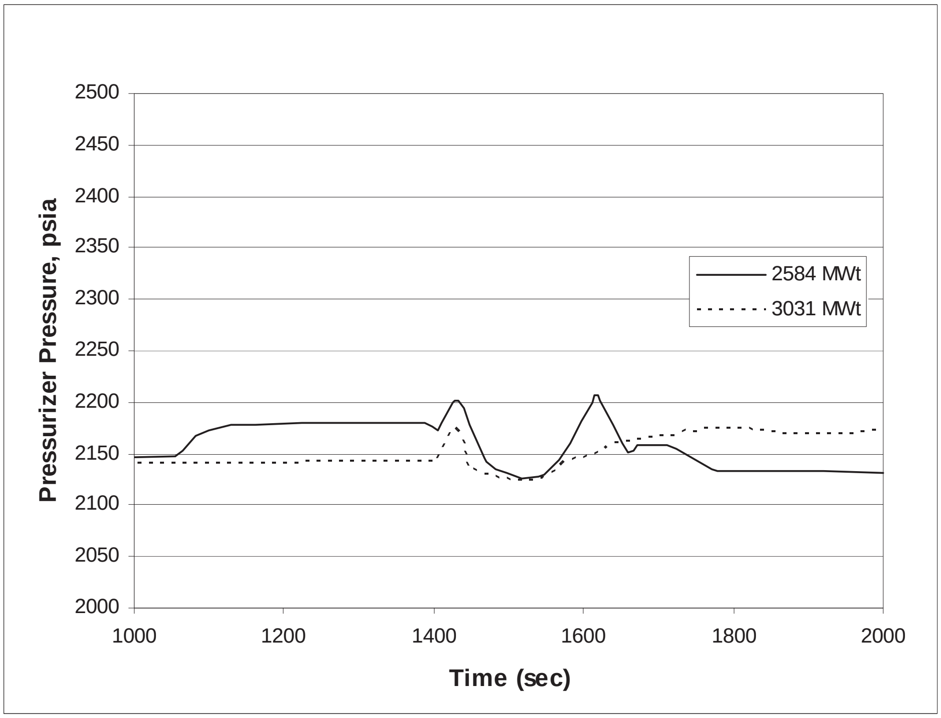


Figure 2.12.2-3: Step Load Increase Secondary Pressure Response

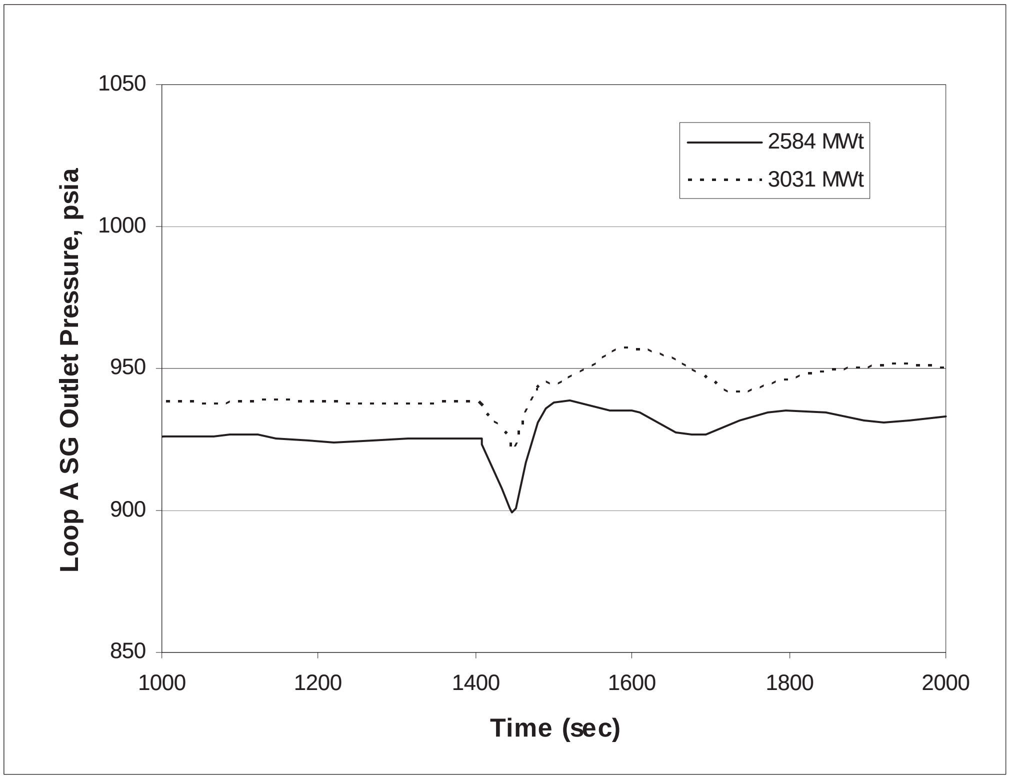


Figure 2.12.2-4: Step Load Decrease Power Response

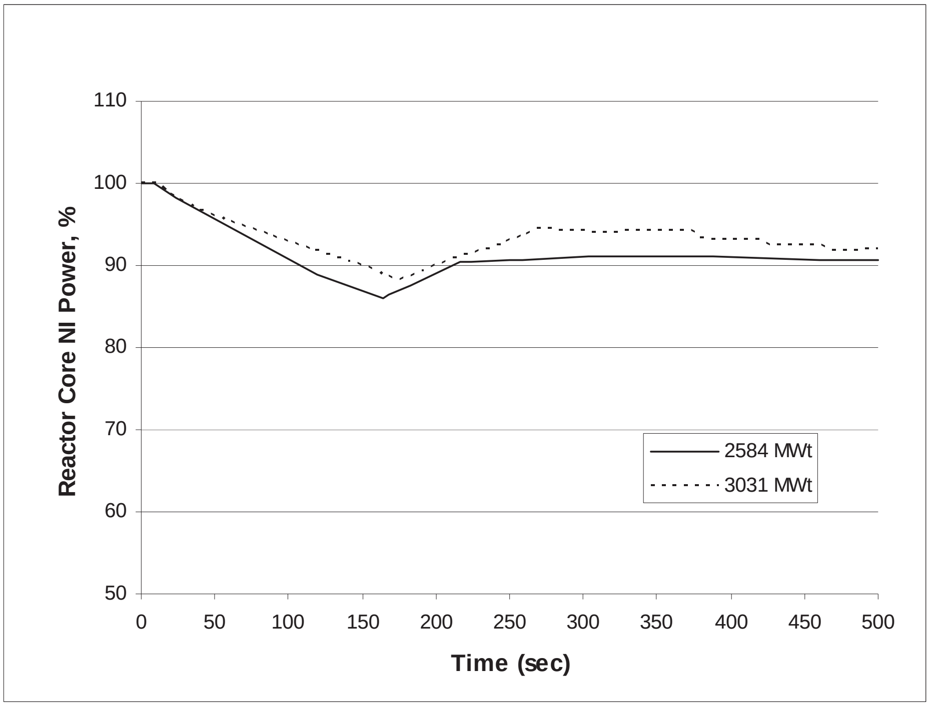


Figure 2.12.2-5: Step Load Decrease Primary Pressure Response

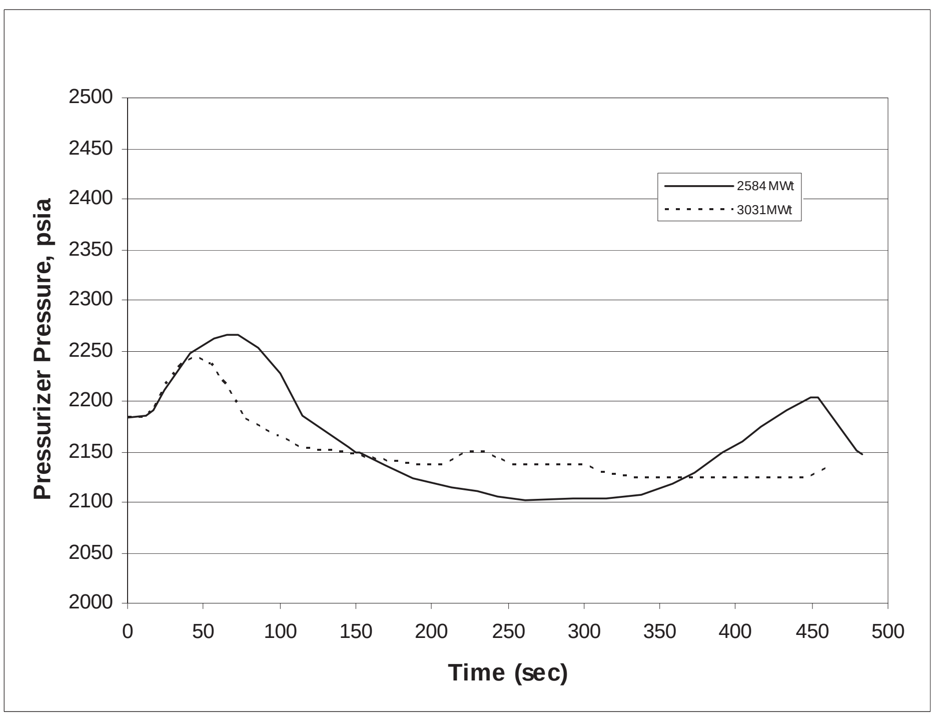


Figure 2.12.2-6: Step Load Decrease Secondary Pressure Response

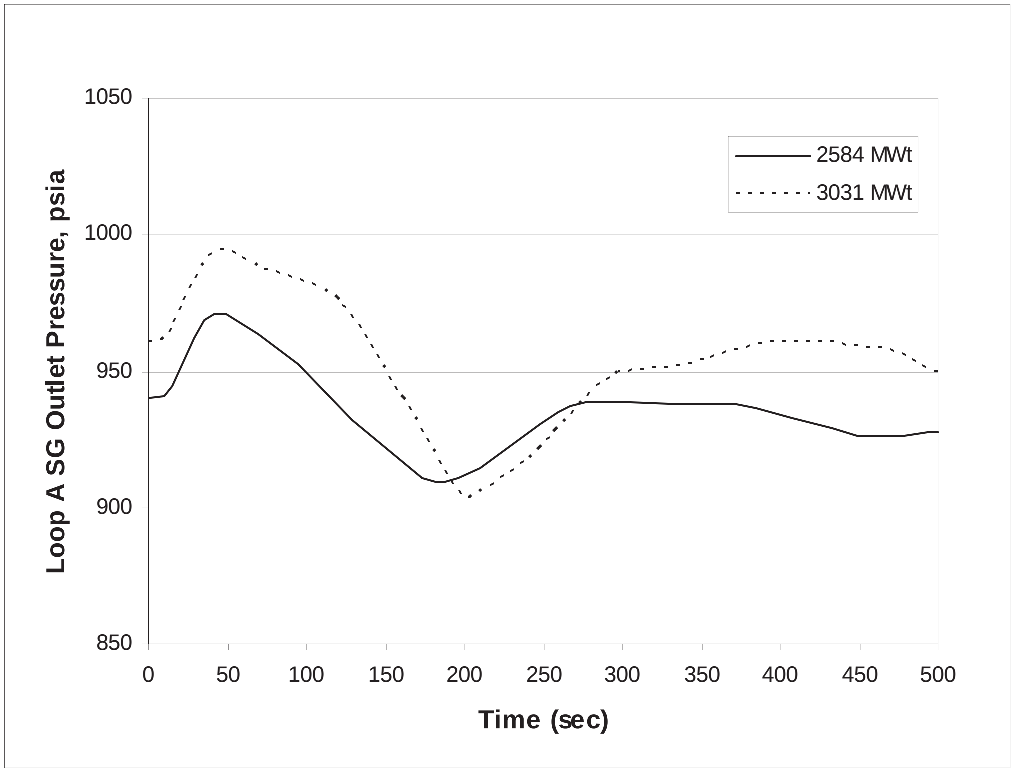


Figure 2.12.2-7: Ramp Power Change, Core Power Response (100% - 8%, 8% - 100%)

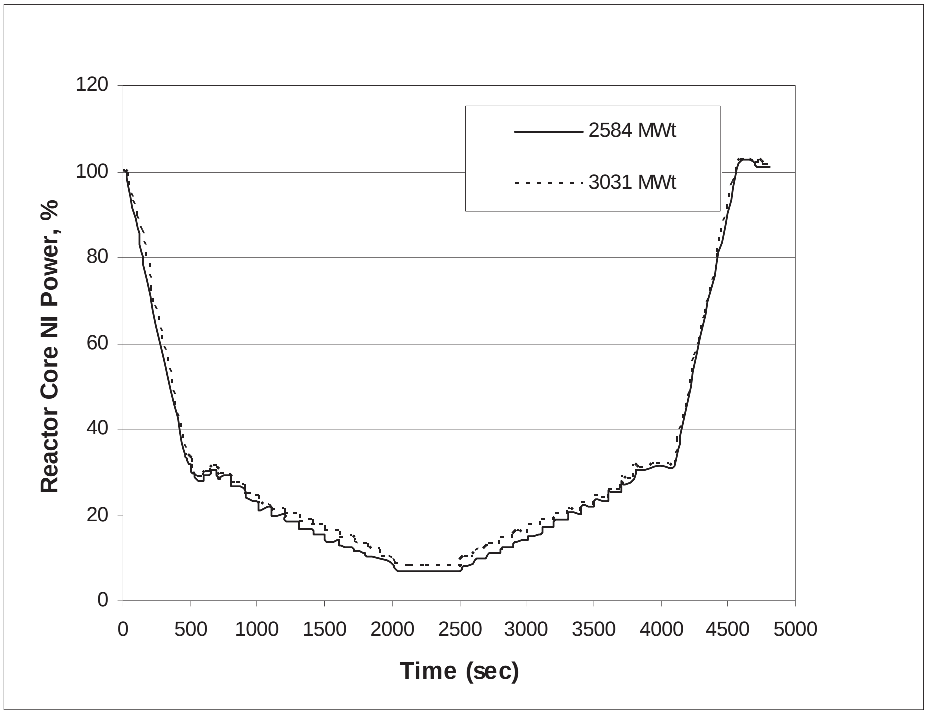


Figure 2.12.2-8: Ramp Power Change, Primary Pressure Response (100% - 8%, 8% - 100%)

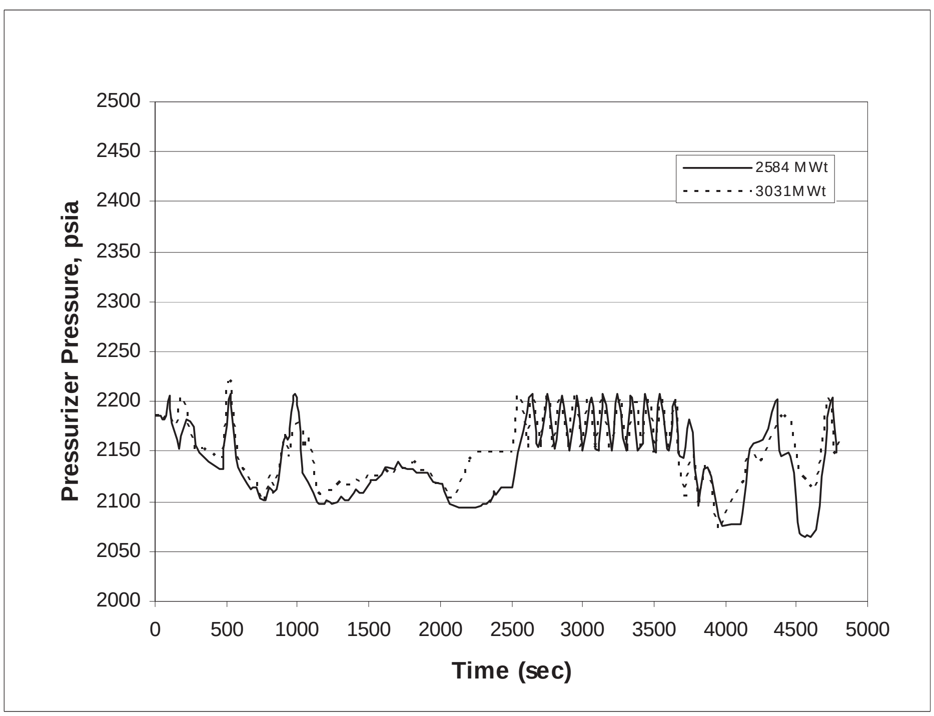
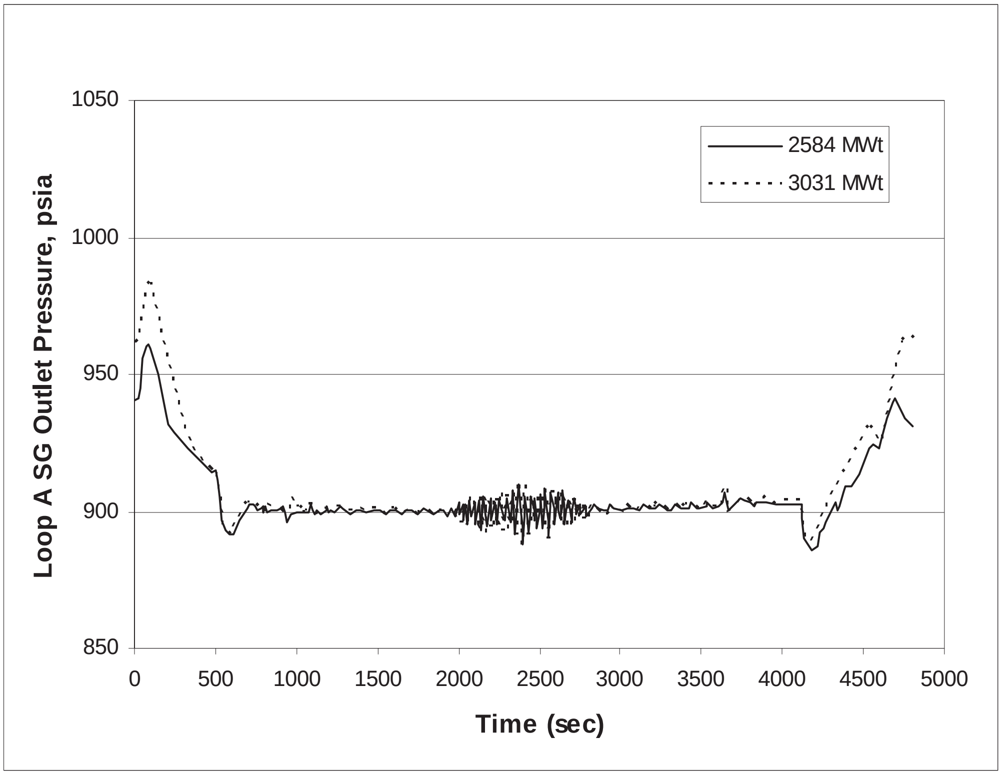


Figure 2.12.2-9: Ramp Power Change, Secondary Pressure Response (100% - 8%, 8% - 100%)



2.13 Risk Evaluation

2.13.1 Regulatory Evaluation

CR-3 conducted a risk evaluation to (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of Standard Review Plan (SRP) Chapter 19, special circumstances are present if any issue would potentially rebut the presumption of adequate protection provided by the licensee to meet the deterministic requirements and regulations. The CR-3 review covered the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the CR-3 review covered the quality of the risk analyses used to support the application for the proposed EPU. This included a review of the actions to address issues or weaknesses that may have been raised in previous peer reviews of the CR-3 individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE), or by an industry peer review.

CR-3 Current Licensing Basis

The 1993 IPE (Level 1 PSA model) was first reviewed and accepted for the intended purpose (response to Generic Letter 88-20) and has been updated to ensure the model represents the as-built and operated plant. The updates addressed plant modifications, data updates, and modeling improvements.

The Level 1 probabilistic safety analysis (PSA) was expanded to meet IPEEE requirements, and was submitted and accepted for that purpose. The CR-3 IPEEE contains the most current information for external events.

The CR-3 PSA model history, through the 2006 model, was described as part of CR-3 License Amendment Request (LAR) 295. Approval of the LAR was documented in CR-3 License Amendment 229 (Accession Number ML081060231). Therefore, only changes to the PSA model after 2006 are presented in this submittal.

The model includes the fault trees, basic event data, and software configuration files required to generate CDF and LERF results. The PSA is intended to be a best estimate tool. It does not necessarily assume design basis conditions. The PSA can credit non-safety related equipment for accident mitigation when justified, and allows multiple failures of safety-related equipment based on probability. The current PSA model of record is based on internal events and internal flooding only. CR-3 plant programs are currently based upon the 2006 revision of the model and will be updated to the 2009a revision of the PSA model in the near future.

PSA Model 2008 Update

The CR-3 PSA model update 2008 was completed in February 2009. This revision was performed to incorporate the American Society of Mechanical Engineers (ASME) gap self-assessment findings, update plant specific data, and incorporate fault tree logic to support the fire PSA model.

PSA Model 2009 Update

The CR-3 PSA model update 2009 was completed in July 2009. This revision was performed to incorporate the ASME gap self-assessment findings associated with the Human Reliability Analysis (HRA), and incorporate fault tree logic to support the fire PSA.

PSA Model 2009a Update (Pre-EPU)

Prior to including the EPU modifications into the PSA model, a couple of deficiencies or opportunities for improvement were noted in the 2009 base model. These items were not related to EPU, however, they were included in the 2009a model and EPU model to be consistent in modeling the plant as designed and operated.

PSA EPU Model

The model used for this evaluation is called the EPU model. This model is an update of the 2009a model including those EPU modifications that affect plant risk. Comparisons to the change in risk are made between 2009a model and the EPU model.

2.13.2 Technical Evaluation

Introduction

PSA Scope/Description

CR-3 has an at-power PSA model that includes:

- Internal Events
- Internal Floods
- Fire
- Level 2 Analysis
- LERF Analysis

The CR-3 Internal Events PSA model uses small event tree/large linked fault tree methodology. Event trees are developed for each unique class of identified internal initiating events, and top logic is developed to link these functional failures to system-level failure criteria. Fault tree modeling components and human failure events are developed for each of the systems identified in the top logic.

Model 2009a PSA Results

The CR-3 CDF is currently 3.4 E-06 which is generally lower than other similar units. The reasons for the lower CDF are as follows:

- Byron Jackson N-9000 Reactor Coolant Pump (RCP) seals are installed and are assumed to maintain their integrity as long as they have seal injection, or seal cooling, or the RCPs are tripped. This greatly reduces the likelihood of an RCP seal failure causing a LOCA.

- Offsite power is supplied from a 230 kV switchyard that has feeds from the grid and from three fossil plants onsite. CR-3 outputs to a separate 500 kV switchyard. Based on this, dependent loss of offsite power events occurring due to trip initiators is not considered a credible event.

- CR-3 has a third non-safety related diesel that can power an engineered safeguards (ES) bus that adds additional redundancy for loss of offsite power scenarios.

- CR-3 emergency diesel generators (EDGs) are not dependent upon a cooling water supply. The EDGs at CR-3 are air cooled machines.

- CR-3 maintains a diverse secondary cooling capability, including automatically actuated steam and diesel driven emergency feedwater pumps (EFP-2 and EFP-3), a backup motor driven pump powered from the ES bus (EFP-1), and a backup motor driven pump that is powered from normal offsite power or the alternate emergency diesel generator (FWP-7).

- CR-3 has three high head injection/makeup pumps each capable of providing adequate primary cooling via the pressurizer power-operated relief valve or pressurizer safety valves at full Reactor Coolant System (RCS) pressure. The High Pressure Injection (HPI) pumps also have diverse support systems. Two of the pumps have backup cooling and one can be powered from either ES 4160 kV bus.

- CR-3 has separate safety-related service water systems for the decay heat removal system and nuclear services support for other systems. The Nuclear Services System also has a third non-safety related train that can cool normal loads.

- CR-3 has a dedicated chiller installed for 10 CFR 50, Appendix R (fire), considerations that is not dependent on service water.

<u>Description of Analyses and Evaluations</u>

An evaluation was performed to determine the impacts of the EPU on CR-3 plant risk as reflected in the CR-3 PSA. The evaluation included the following:

- Potential impacts of the EPU project on the CR-3 PSA models due to hardware modifications, setpoint changes, and procedure changes to be implemented as part of the EPU project were identified.

- A determination of the quality and technical adequacy of the CR-3 PSA to support the risk significance of these changes was performed.

- A detailed, section-by-section, review of each of the PSA notebooks was performed to identify where the PSA models or documentation could be impacted by the EPU.

- Where the potential for a change impacting plant risk was identified, the affected notebook was revised to address the impact of EPU on the analyses.

EPU-Related Modifications Considered

Several modifications to CR-3 are required to implement the EPU. Since the base model was from 2009a, this evaluation included modifications that have been or will be installed during refueling outages 16R (2009/2010) and 17R. This assessment evaluates the change in CDF and LERF that can be expected as a result of proposed modifications to CR-3 for the implementation of the EPU. Modifications considered in this review include each of those in Appendix E.

Evaluation of EPU Impacts to CR-3 PSA Internal Events

This section describes evaluation of the potential effects of the EPU to the overall risk for CR-3 as evaluated by the PSA. Each of the issues, identified as having the potential for a numerical impact on the PSA, is evaluated to determine the expected effect. The identified impacts are included in the CR-3 PSA models and combined effects quantified to determine the risk change expected as a result of the EPU. A summary of the risk evaluation is provided including any insights for each discussion topic.

Internal Initiating Events

The CR-3 Internal Events PSA includes loss of coolant accident (LOCA), steam generator tube rupture (SGTR), loss of offsite power (LOOP), secondary line break, and transient initiators. For internal event initiators, the underlying contributors to these initiating events were reviewed to determine the potential effects of the EPU on the initiating event frequencies. The results of these evaluations are summarized in the sub-sections that follow.

<u>Loss of Coolant Accident (LOCA)</u>

These frequencies (all sizes) are determined by the potential for passive pipe failures. The EPU does not involve changes to the RCS piping. However, changes to piping that could lead to an interfacing system LOCA (ISLOCA) are planned.

For CR-3, LOCA events other than ISLOCAs are defined as pipe breaks of the RCS piping. The frequency of all pipe break LOCA events is taken from NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995. These are rare events and no change in RCS piping (other than for ISLOCA) is expected as a result of the EPU. Therefore, the pipe failure frequency values for small, medium, large, vessel rupture LOCAs are not affected by the EPU.

A LOCA can also occur as a result of an RCS pressure excursion that results in a stuck open pressurizer power operated relief valve (PORV) or safety relief valve (SRV). The CR-3 PSA includes such consequential LOCAs following transient events that involve spurious actuation of HPI. The frequency of spurious HPI is a random event that is not affected by the EPU. Therefore, no change in consequential LOCA frequency is expected as a result of the EPU.

The ISLOCA frequency analysis for CR-3 considers all pathways that connect low design pressure systems to the RCS and lead outside containment. Installation of the LPI cross-tie and the boron precipitation line result in an additional pathway with the potential for an ISLOCA. These additional pathways were evaluated and the CR-3 specific ISLOCA frequency updated. The results of this update are shown below:

Basic Event	Event Description	Base Value	Percent Increase	Updated Value
IE_V	ISLOCA - DHR Drop Line and Injection Lines	5.16E-9	7%	5.49E-9

Steam Generator Tube Rupture (SGTR)

The frequency of a SGTR is independent of power level. Although the EPU will result in increased steam flow and minor changes in primary and secondary side temperatures and pressures, no change in SGTR frequency is expected as a result of the EPU.

Loss of Offsite Power (LOOP)

No changes to the switchyard or related equipment were identified for the EPU project. The LOOP frequency at CR-3 is calculated by considering LOOP events that have occurred throughout the industry and performing a Bayesian update to determine the frequency. The method used to determine LOOP frequency does not change and no new failure modes related to loss of offsite power have been identified for the EPU project. Therefore, it is concluded that there are no changes to the frequency of LOOP as a result of the EPU.

Secondary Line Break

The frequency of secondary line breaks for the CR-3 PSA is based on a Bayesian update of generic failure data. Although slightly higher steam and feedwater flows will occur after the EPU, plant flow accelerated corrosion (FAC) programs are deemed adequate to prevent any changes in pipe failure frequency. No specific failure mechanisms or failure locations for secondary line breaks were identified. An inadvertent opening of ADV spuriously or by the Fast Cooldown System (FCS) would cause similar plant impact and response as a steam line break, therefore this event is categorized with the secondary line break. A Bayesian updated generic industry value is used for a secondary line break initiating frequency. The addition of the FCS does not change the method used to determine nor the value of the secondary line break frequency. Therefore, it is concluded that there are no changes to the steam line/feed line break frequency as a result of the EPU.

Transients Initiators

An assessment of each transient initiating event included in the CR-3 PSA is performed to determine if the frequency of any transient initiating event could be impacted by any component or system changes.

Reactor/Turbine Trip

The reactor/turbine trip initiating event represents several trip contributors that do not have important direct, unique effects on the need for, or the availability of, plant systems. The frequency for a reactor/turbine trip is developed using CR-3-specific operating history. No planned changes were identified that would have a direct impact on transient frequency. The addition of the FCS or the Inadequate Core Cooling Monitoring (ICCM) System does not change the method used to determine reactor/turbine trip frequency. Therefore, it is concluded that there are no changes to the reactor/turbine trip frequency as a result of the EPU.

Loss of Main Feedwater (MFW)

There are several control system and hardware faults that can lead to a loss of MFW to both steam generators. The loss of MFW directly trips both the reactor and the turbine, and actuates Emergency Feedwater (EFW) to provide backup cooling flow to the steam generators. If the EFW System responds as designed, the subsequent plant response is very similar to that of the turbine trip event. The frequency for loss of MFW is developed using CR-3-specific operating history. No planned changes were identified that would have a direct impact on transient frequency. The modifications to the Feed Water and Condensate System do not change the function of the various components or plant response. The additional condensate feedwater heaters (CDHE-7A and 7B) provide a new parallel flowpath for condensate to supply the feedwater, and do not have any impact on loss of MFW. Therefore, it is concluded that there are no changes to the loss of MFW frequency as a result of the EPU.

Excessive Feedwater

An excessive feedwater initiating event can result from control failures in the Integrated Control System (ICS) or from MFW System hardware failures, and will lead to overcooling of the RCS. The Excessive Feedwater frequency at CR-3 is a Bayesian update of the generic industry value from NUREG/CR 6928. The spurious actuation of the new ICCM System could impact emergency feedwater but has no impact on ICS or the MFW System. Therefore, the addition the ICCM System does not change the method used to determine nor the value of the excessive feedwater event frequency. Therefore, it is concluded that there are no changes to the excessive feedwater frequency as a result of the EPU.

Spurious ES Actuation

A spurious ES actuation signal can have many possible outcomes, including a reactor trip, HPI actuation, reactor building isolation, and main-steam isolation. If the spurious actuation originates with a high reactor building pressure signal on 2-of-3 channels, the impact can include all these outcomes. The spurious actuation can be due to human error or instrument or relay failure with one channel in test. The Spurious ES frequency at CR-3 is calculated by a combination of event types from NUREG/CR 6928 and NUREG/CR 5750 and performing a Bayesian update to determine the frequency. The addition of the FCS or the ICCM System does not change the method used to determine Spurious ES frequency. No planned changes were identified that would have a direct impact on spurious ES actuation calculation methodology. Therefore, it is concluded that there are no changes to the spurious ES actuation frequency as a result of the EPU.

Loss of ES 4160 V Bus “A”

A loss of the 4160 V ES bus “3A” causes a loss of one train of ES equipment. Such an effect would not result in a reactor trip since most of the equipment powered from the 4160 V ES bus is not needed during power operations. However, the CR-3 Improved Technical Specifications (ITS) requires plant shutdown if an inoperable 4160 V ES bus cannot be restored within eight hours. The CR-3 probabilistic risk assessment (PRA) assumes that plant shutdown always occurs shortly after a loss of either 4160 V ES bus. No planned changes were identified that would affect the ES buses. Therefore, it is concluded that no changes to the loss of the ES 4160 V Bus “A” frequency will be expected as a result of the EPU.

Loss of ES 4160 V Bus “B”

A loss of the 4160 V ES bus “3B” causes a loss of one train of ES equipment. Such an effect would not result in a reactor trip since most of the equipment powered from the 4160 V ES bus is not needed during power operations. However, CR-3 ITS requires a plant shutdown if an inoperable 4160 V ES bus cannot be restored within eight hours. The CR-3 PRA assumes that plant shutdown always occurs shortly after a loss of either 4160 V ES bus. No planned changes were identified that would affect the ES buses. Therefore, it is concluded that no changes to the loss of the ES 4160 V Bus “B” frequency will be expected as a result of the EPU.

Loss of Service Water

The Nuclear Services Closed Cycle Cooling System, or Service Water (SW) System, removes heat from various components and transfers this heat to the Nuclear Services Raw Water (RW) System. The CR-3 PSA assumes that a manual trip occurs shortly after any loss of service water in order to protect plant equipment. No plant changes affecting service water design or operation were identified. Therefore, it is concluded that there are no changes to the loss of service water frequency as a result of the EPU.

Loss of Raw Water

The RW System is made up of two sub-systems, the Nuclear Services Seawater (RW-SW) sub-system (cooling the SW System) and the Decay Heat Seawater sub-system (RW-DC) (cooling the DC System that cools the Decay Heat System (DH)). The RW-DC System is a standby system and cannot cause a plant trip, but can be unavailable at the time of a plant trip due to common start failures or loss of the intake structure. The RW-SW system includes one normally running pump and two safety-related standby pumps. A loss of the normally running pump and failure of the two standby pumps to start, would result in a manual trip and would eventually cause an automatic reactor trip following RCP trip. No plant changes affecting Raw Water System design or operation were identified. Therefore, it is concluded that there are no changes to the loss of raw water frequency as a result of the EPU.

Loss of Battery Backed Bus “A”

A loss of main vital DC bus “A” does not directly cause a reactor/turbine trip. However, control power to important equipment is lost and, due to the loss of control to many redundant safety functions, the CR-3 ITS requires a plant shutdown to commence in two hours if the main DC distribution panel was not recovered. Therefore, this event is included in the CR-3 PSA as a manual shutdown. No plant changes affecting battery backed bus “A” design or operation were identified. Therefore, it is concluded that there are no changes to the loss of battery backed bus “A” frequency as a result of the EPU.

Loss of Battery Backed Bus “B”

A loss of main vital DC bus “B” does not directly cause a reactor/turbine trip. However, control power to important equipment is lost and, due to the loss of control to many redundant safety functions, the CR-3 ITS requires a plant shutdown to commence in two hours if the main DC distribution panel was not recovered. Therefore, this event is included in the CR-3 PSA as a manual shutdown. No plant changes affecting battery backed bus “B” design or operation were identified. Therefore, it is concluded that there are no changes to the loss of battery backed bus “B” frequency as a result of the EPU.

Loss of Battery Backed Bus “C”

Plant response to a loss of non-1E DC power (bus DPDP-1C) is similar to a loss of MFW except that the DC power failure blocks the normal turbine trip which leads to an overcooling with HPI actuation. No plant changes affecting the battery backed bus “C” design or operation were identified. Therefore, it is concluded that there are no changes to the loss of battery backed bus “C” frequency as a result of the EPU.

Loss of Makeup (MU)

A loss of normal makeup would result in a decrease in pressurizer level. If normal makeup cannot be quickly reestablished using the alternate pump, the operators terminate normal letdown. If the loss of makeup is due to a flowpath failure, multiple injection paths exist to makeup to the RCS and could be implemented as needed until normal makeup is restored or the plant is shut down. If a loss of makeup is a result of loss of all three MU/HPI pumps, then a reactor trip or shutdown would be required, and feed-and-bleed cooling capability would be unavailable. The installation of makeup tank bypass valve is designed such that valve failure will not prevent flow to the pumps. Therefore, it is concluded that there are no changes to the loss of normal makeup frequency as a result of the EPU.

Station Blackout (SBO)

The frequency for a SBO is not calculated as a separate, individual event. Rather, the SBO frequency is determined through the quantification process using initiating events discussed above and system logic models to estimate the probability of losing all alternating current (AC) power. Any changes to system logic required as a result of the EPU are discussed in the sections that follow and will be included as necessary in the quantification process. Therefore, no changes are needed to address the SBO frequency for the EPU.

Anticipated Transient Without Scram (ATWS)

Similar to the discussion for SBO, ATWS frequency is not calculated directly but is determined through the overall quantification process. Therefore, no changes are needed to address the ATWS frequency for the EPU.

Internal Events Accident Sequence Analysis and Event Trees

This section describes the assessment of how the EPU could affect the event tree analysis for the CR-3 PSA. An event tree analysis is done for each initiating event category discussed above. The event trees are constructed by identifying front line safety systems and operator actions that either respond to the initiating events or mitigate failures of other frontline systems. The event tree models are delineated to lead to safe states, transfers to other trees, or core damage states.

In the subsections that follow, the potential for the EPU to affect the accident sequence and event tree analysis for each event tree is evaluated and discussed.

Large LOCA

The large LOCA event tree model requires at least one core flood tank (CFT) injection, low-pressure injection by one of two decay heat pumps in the low pressure injection (LPI) mode with suction from Borated Water Storage Tank (BWST) and recirculation using one of two decay heat pumps with suction from reactor building sump. The success criteria for the decay heat removal is continued supply of cooling water to at least one train of the LPI System operating in the recirculation mode. Although the post-EPU power levels are higher, evaluation of success criteria for the higher power levels indicates no changes in the success criteria or timing information. Therefore, there are no changes to the large LOCA event tree analysis as a result of the EPU.

Medium LOCA

The current medium LOCA event tree model requires high-pressure injection and Emergency Core Cooling System (ECCS) recirculation. As a result of the higher power levels, success criteria evaluations show that for the medium LOCA events near the larger end of the spectrum, core damage could occur without injection from at least one CFT. To conservatively bound all medium LOCA events, the accident sequence analysis for medium LOCAs was changed to require injection from at least one CFT for all medium LOCA events. The success criteria now becomes injection by one of two CFTs and actuation of one train of FCS and HPI by at least one of three HPI pumps, drawing suction from the BWST and providing flow via at least three injection nozzles, or HPI by at least two of three HPI pumps drawing suction from the BWST and providing flow via at least two injection nozzles. Long-term inventory control is achieved through continued LPI injection to the RCS with the suction source switched to the reactor building emergency sump prior to depleting the BWST inventory. Success criteria for decay heat removal is the continued supply of cooling water to at least one train of the LPI System operating to supply suction to the HPI system in the high pressure recirculation (i.e., “piggy-back”) mode.

The atmospheric dump valve (ADV) modification includes the FCS feature to support rapid depressurization of the RCS for a medium break LOCA. Once a MBLOCA has occurred, the FCS automatically initiates (which controls the ADVs to maintain main steam pressure at a lower value), the feature provides a more-rapid RCS cooldown.

Small LOCA (SBLOCA)

The small LOCA event tree model requires high-pressure injection, secondary side decay heat removal, RCS pressure control, and ECCS recirculation. Although the post-EPU power levels are higher, evaluation of success criteria for the higher power levels indicates no changes in success criteria or timing information. Therefore, there are no changes to the SBLOCA event tree analysis as a result of the EPU.

- Success criteria for the control of RCS pressure is:
    - Continued RCS heat removal via feedwater after the reactor trip (as below for decay heat removal); or
    - Pressure relief via opening of one of the following: the pressurizer PORV or
    - At least one of two pressurizer SRVs.
- Success criteria for control of RCS inventory is:
    - HPI by at least one of three HPI pumps, drawing suction from the BWST and providing flow via at least three injection nozzles, or HPI by at least two of three HPI pumps drawing suction from the BWST and providing flow via at least two injection nozzles; and
    - Long-term control of inventory loss by continued injection to the RCS, supplied from the LPI pumps, with the suction source switched to the reactor building emergency sump prior to depleting the BWST inventory.
- Success criteria for decay heat removal is:
    - Heat removal via at least one SG with the thermal center in the SG raised and with flow provided by at least one of the following options:
        - One of two EFW pumps actuated automatically or manually aligning and starting the spare EFW pump or manually aligning and starting the auxiliary feedwater pump, or

    - If heat removal via the SGs is lost, feed-and-bleed cooling from the flow to the RCS by at least one of three HPI pumps drawing suction from the BWST, discharging through the pressurizer PORV, and
- Long-term makeup to the RCS by one of three HPI pumps in the High Pressure Recirculation mode (as outlined above).

Steam Generator Tube Rupture (SGTR)

The accident progression in the SGTR event tree model includes high-pressure injection to maintain RCS inventory followed by operator action to cooldown and isolate the faulted steam generator and depressurize the RCS to stop the loss of primary coolant. Timing analyses for the SGTR events are based on the design-basis timing shown in the FSAR Section 14.2.2.2. Since these times have not changed as a result of the EPU, there are no changes to the SGTR event tree analysis as a result of the EPU.

Interfacing System LOCA (ISLOCA)

The ISLOCA event tree model assumes that core damage occurs following the ISLOCA initiating event. No EPU modification will change the mitigation of an ISLOCA, therefore, there are no changes to the ISLOCA event tree analysis as a result of the EPU.

Reactor Vessel Rupture

The reactor vessel rupture event tree model assumes that core damage occurs following the initiating event. No EPU modification will change the mitigation of a reactor vessel rupture, therefore, there are no changes to the reactor vessel rupture event tree analysis as a result of the EPU.

Transients

The transient event tree model is used for all transient-type events including special initiating events, LOOP, and SBO. The transient event tree includes reactivity control, RCS pressure control, RCS inventory control, and decay heat removal (i.e., with either EFW, auxiliary feedwater (AFW), or primary feed-and-bleed cooling). Although the flow required from EFW or AFW is higher as a result of the EPU, the higher flow can be supplied by the existing pumps so no change in the event tree is needed. The EPU will result in changes in steam generator mass and decay heat levels, so initiation of feed-and-bleed cooling is affected. For the EPU model, analyses show that if all secondary side heat removal is lost, reactor coolant released through the pressurizer PORV will cause containment pressure to rise and actuate high-pressure injection, effectively initiating feed-and-bleed cooling without any operator action.

Included in the transient event tree analyses is the SBO analysis. The increased power levels for post-EPU conditions will result in less time available to recover offsite power before core damage. The shorter time is accounted for within the convolution analysis by changes to the non-recovery probability values. However, no changes in the overall accident progression would occur.

Anticipated Transient Without Scram (ATWS)

For the PSA model, a nominal value for Moderator Temperature Coefficient (MTC) was used in the base-case analyses of response to a failure to trip. No changes to reactivity characteristics affecting the ATWS accident sequence modeling were identified for post-EPU conditions. Therefore, there are no changes to the ATWS event tree analysis as a result of the EPU.

Data Analysis

The existing plant specific data was evaluated and no EPU-specific data changes relevant to the plant specific data were identified. The model changes for the EPU PSA modeled changes, such as the LPI System cross-tie and boron precipitation line, EF recirculation valve, use generic, standard manual and check valves failure rates. The ADVs are being modified to be safety-related with a backup air supply and automatic initiation circuit. Although additional data gathering for new equipment is required as part of the long-term PSA maintenance and update process, there is no reason to presume that the failure rates or maintenance unavailability for new components will be either higher or lower than the existing, similar plant components performing the same function. Therefore, there are currently no changes to the data analysis expected as a result of the EPU.

Offsite Power Non-Recovery Probability

As part of the process of assessing the frequency of core damage accidents initiated by a LOOP, it is necessary to account for the potential for offsite power to be recovered. The time available to recover offsite power is based on a complete loss of AC power and steam generator makeup. For the 2009a PSA model, analyses performed with the MAAP 3.0b code show that core damage can be avoided if RCS makeup is initiated within 60 minutes following a complete loss of power. For post-EPU conditions, updated thermal-hydraulic calculations performed with the MAAP 4.0.6 code show that core damage can occur if RCA feed-and-bleed cooling is not initiated within 55 minutes following a loss of all AC power and secondary makeup.

The offsite power non-recovery probability values were re-evaluated to account for the shorter time available for power recovery.

Systems Analysis

This section describes the assessment of how the EPU could affect each of the system analyses performed for the CR-3 PSA. In the subsections that follow, the potential for the EPU to affect each of the system analyses is evaluated and discussed. Success criteria in the PSA model is on an accident sequence basis, not on a system basis. Thus, success is discussed in the accident sequence section.

Switchyard

The switchyard is divided into two voltage levels (500 kV and 230 kV). The 500 kV switchyard consists of breakers, disconnect switches, motor-operated disconnect switches, and connecting bus work arranged in a “ring bus” configuration. The 500 kV switchyard provides physical connections to the grid from the CR-3 and fossil unit, CR-5, generators. The switchyard can also be used, under certain conditions, to supply power to the plant through the main transformers via the auxiliary transformer in a “backfeed” alignment.

The 230 kV switchyard consists of similar components, and connecting bus work arranged in a “breaker and a half” scheme. The 230 kV switchyard provides connections for fossil units CR-1, CR-2, and CR-4 to the grid, and supplies power to the CR-3 startup transformers, the offsite power transformer and the backup ES transformer. The Crystal River 230 kV substation serves as the preferred source of power for reactor plant startup, normal operations, emergency operations, and plant shutdown. No changes to the switchyard or associated systems are planned as a result of the EPU, and no changes to the system success criteria were identified per Section 2.3.2, Offsite Power System. Therefore, no changes to the switchyard transformer system model were required as a result of the EPU.

<u>6900 V Electric Power</u>

Power from the switchyard is provided to the plant distribution system. It is comprised of the 6900 V and 4160 V systems. The 6900 V bus distribution system consists of two separate buses and related breakers. The 6900 V distribution system provides power to the reactor coolant pumps, auxiliary feedwater pump FWP-7 and the reactor building chiller, with the latter two being 4160 V AC loads supplied from the 6900 V AC bus through a step down transformer. The EPU does not change the 6900 V system and how the PSA models the system or system interactions.

The AC Power System model includes the AC power supplies from the switchyard to the 120 V AC instrument power. The AC Onsite System provides motive and control power to various loads modeled in the PSA.

The AC Onsite System and its functions can be divided by voltage levels. The Class 1E Electrical System provides reliable electrical power to engineered safeguards, critical, and essential systems that shut down the reactor and limit the release of radioactive material during all modes of operation and shutdown conditions, including any design basis event. The Class 1E Electrical System is comprised of two redundant electrical subsystems that provide power to redundant engineered safeguards trains. Physical separation of the subsystems is such that they are not affected by a common fire or common mechanical damage.

In addition to the 6900 V and 4160 V distribution, power is provided to the reactor building maintenance support building, emergency feedwater pump non-safety motor control center (MCC), industrial cooling system cooling tower, and 480 V switchgear/bus MTSW-4. A separate offsite 12.5 kV line supplies power to this switchgear and to an alternate supply for two instrument air compressors. A 12.5 kV/480 V transformer mounted at the switchgear furnishes the 480 V supply to the bus.

<u>4160 V Electric Power</u>

Two 4160 V buses supply engineered safeguards loads and have emergency power backup from the emergency diesel generators. The 4160 V distribution provides power to non-safety related 4160 V motors and to the 480 V distribution. The 4160 V Best auxiliary bus can be powered by manually initiating the alternate AC diesel generator EGDG-1C, which also can support FWP-7. The balance of plant 4160 V distribution system is comprised of two unit 4160 V buses, a transformer, and associated breakers. The EPU does not change the 4160 V system and how the PSA models the system or system interactions.

480 V Electric Power

The 480 V Distribution System is divided into safety (Emergency Safeguards ES) and non-safety distribution. The 480 V ES Distribution System consists of two 480 V buses and associated breakers, two 4160 V/480 V transformers, and seven 480 V MCCs plus one emergency feedwater pump MCC and associated breakers. No changes to the 480 V power buses are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, there are no changes to the 480 V system models as a result of the EPU.

Air Handling System (PSA System Model AH or HVAC)

The Air Handling System (AH or HVAC System), as modeled in the PSA, is a compilation of various portions of several different plant ventilation systems. The AH System is composed of the cooling mechanisms associated with the emergency feedwater isolation and control (EFIC) rooms, the control complex, the EDG rooms, and the Decay Heat Closed Cycle Pump Cooling System. No changes to the AH or HVAC or associated systems are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the AH or HVAC system models are required as a result of the EPU.

Reactor Building Spray (PSA System Model BS)

The reactor Building Spray (BS) System consists of two redundant subsystems. Each subsystem contains one reactor building spray header, a pump, associated piping, valves, and instrumentation. No changes to the BS System are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the BS system model are required as a result of the EPU.

Core Flood (PSA System Model CF)

The Core Flood (CF) System is a passive injection system that provides direct injection of the contents of the CFTs into the reactor vessel for a large break LOCA that rapidly depressurizes the RCS. No changes to the CF System are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the CF System models are required as a result of the EPU.

Chilled Water (PSA System Model CH)

The Chilled Water (CH) System is composed of two supply systems: 1) the Control Complex Chilled Water System, and 2) the Appendix R Chilled Water Cooling System. These systems provide convectional cooling as required or desired during all modes of plant operation. Both of the chilled water system subsystems are closed loop systems. No changes to the chill water systems are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the CH System models are required as a result of the EPU.

Decay Heat Closed Cycle Cooling (PSA System Model DC)

The Decay Heat Closed Cycle Cooling (DC) System removes heat from various components and transfers this heat to the decay heat Raw Water System (RW). The system consists of two independent closed cycle loops. In each loop, water exits the DC pump to a supply header and is directed to each of the six parallel system heat loads. The DC water then flows to a return header, where it is cooled in the

DC heat exchanger by the Decay Heat Raw Water System and returned to the DC pump suction. No changes to the DC System are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the DC System models are required as a result of the EPU.

Decay Heat Removal (PSA System Model DH)

The Decay Heat Removal (DH) System provides normal heat removal operation following plant cool down and provides low pressure makeup following a LOCA in the LPI and both low and high pressure recirculation modes of operation.

The DH System consists of two pumps, two heat exchangers, a BWST, interconnecting piping, and motor-operated control and isolation valves required for normal and emergency system operation. The DH System is comprised of two independent and redundant cooling trains. Each train is capable of providing 100% of the heat removal requirements for a normal reactor shutdown, LOCA emergency cooling. Each train may take suction from the BWST, the reactor building sump or the RCS “B” hot leg (decay heat drop line). The EPU LOCA analysis has required modification to the DH System for the PSA. These changes result in changes to the PSA that consist of new basic events to reflect the potential failure of the new LPI injection cross tie and a potential ISLOCA flowpath for the long term boration flowpath. The LPI injection lines and a long term boration flowpath have been added to the DH system model to reflect the potential failure of these passive components. The manual isolation valves were assigned a two year standby failure rate and the check valves have a demand failure rate assigned consistent with similar valves in the DH System. The overall effect on the CR-3 CDF and LERF due to these changes was a negligible increase in CDF and LERF. Both sets of number are presented in the PSA Results section.

DC Power System (PSA System Model DP)

The DC Electrical (DP) System consists of batteries, battery chargers, and DC distribution panels. The DC distribution system is divided into a Class 1E (vital) portion and a non-1E (non-vital) portion. The vital DC Distribution System provides a source of reliable continuous power for DC pump motors, equipment control, and instrumentation that is important to the safe shutdown of the plant. The Fast Cooldown System added new DC power panels which were included in the EPU PRA model.

Emergency Feedwater Initiation and Control (PSA System Model EC)

The Emergency Feedwater Initiation and Control (EFIC) System provides for initiation of emergency feedwater (EFW), control of once through steam generator (OTSG) level when EFW is in use, isolation of any depressurized OTSG, termination of EFW to preclude OTSG overfill, and control of the ADVs. The system consists of four input process channels that are powered from separate vital power buses. Each channel receives dedicated inputs from level and pressure instruments on each OTSG, as well as digital logic input signals from the Reactor Protection System (RPS), ATWS Mitigating System Actuation Circuitry (AMSAC) and Engineered Safeguards Actuation System (ESAS). Therefore, no changes to the EC system model are required as a result of the EPU.

Emergency Feedwater and Auxiliary Feedwater (PSA System Model EF)

The EFW System provides makeup to the OTSGs in the absence of the MFW System. Once actuated, it is controlled by the EFW EFIC System.

The EFW System is in standby during power operation and consists of a dedicated EFW tank, three EFW pumps, and valves, piping, and controls required for system operation. Each EFW pump driver is diverse. EFP-2 is a turbine-driven pump. EFP-3 is a diesel-driven pump. Both EFP-2 and EFP-3 respond automatically to an initiation signal. EFP-1 is an electric motor-driven pump that does not initiate automatically, but can be actuated manually. All pumps are 100% capacity and all pumps take water from the dedicated EFW tank. The condensate storage tank (CST) serves as a backup to this water source. Additionally, EFP-1 and EFP-2 can be supplied from the condenser hotwell. Water leaving the discharge of the EFP-3 and EFP-1 pumps flow to a common header and are connected to both OTSGs. EFP-2 flows into separate headers for both OSTGs. Headers of each train converge prior to injecting into the applicable OTSG. Steam to the EFP-2 turbine (EFTB-1) is supplied from either OTSG. Steam is admitted by opening either of two DC-powered motor-operated valves. One valve (ASV-5) is automatic and the other (ASV-204) is manually actuated. Exhaust steam is dumped to the atmosphere. In addition to the EFW pumps, an AFW pump is also present (FWP-7) and can be manually actuated from the control room to provide an additional redundant water makeup source. The supply to this pump is normally from the CST, but it can also take suction from the condenser hotwell. FWP-7 is an electric motor-driven pump. It is powered normally by plant power, but can also be supplied from diesel generator EDG-1C.

The EPU will require changes to ensure that the EFW pumps can deliver the required rated flow to the steam generators. Additional flow to the steam generators will be generated, when required, by closing newly installed isolation valves in the minimum recirculation lines for the EFW pumps. Some design basis accidents will require closing the valves for the first 10 minutes of the accident to increase EFW flow from approximately 550 gpm to 660 gpm. However, PRA best-estimate analyses (as opposed design basis analysis) performed for the EPU show that the additional flow is not required to prevent core damage. Therefore, the EFW minimum flow isolation does not change the PRA success criteria of the system and therefore the new recirculation valves are not modeled.

Therefore no changes to the EF system model are required as a result of the EPU.

Emergency Diesel Generators (PSA System Model EDGs)

The Diesel Generator (emergency generator or EG) System model addresses the two emergency diesel generators, the Alternate AC diesel generator EGDG-1C, and support systems that supply onsite emergency power in the absence of offsite power to the safety-related buses. No changes to the EDGs or associated systems are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the EDG system model are required as a result of the EPU.

Engineer Safety Actuation System (PSA System Model ES)

The ESAS model addresses the signal generation associated with initiation of safety-related equipment in response to plant conditions. The system consists of sensors, transmitters, relays and associated components necessary to detect plant conditions necessary for actuation of engineered safeguards systems and equipment. The logic is designed with two trains (A and B) that respond to signal inputs based on a two-of-three channel actuation. No changes to the ESAS configuration form or function or associated systems configuration, are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the ES System model are required as a result of the EPU.

Compressed Air (PSA System Model IA)

The Compressed Air System provides high pressure control air for various plant pneumatic valves. The CR-3 ADV identified as MSV-25 and MSV-26 will be made safety-related to allow rapid cool-down in cases of some SBLOCAs. The modification resulted in changes in the Instrument Air (IA) System modeling, which adds new solenoids and backup safety-related air supply for supporting the ADVs. The failure rates for the new components are the same as the similar component in the IA System. The result of this change to the CR-3 PSA is a negligible improvement in the CDF and LERF.

Miscellaneous System (PSA System Model MS)

Various support systems are used in the PSA model for limited support functions which do not warrant a separate system notebook. These include plant water systems used for cooling and bearing flush for seawater and circulating water pumps, and makeup to the BWST. No changes to the MS Systems are planned as a result of the EPU and no changes to the systems' success criteria were identified. Therefore, no changes to the MS Systems model are required as a result of the EPU.

Makeup and Purification System (PSA System Model MU)

The Makeup and Purification (MUP) System provides for inventory and water chemistry control of the reactor coolant, and for emergency makeup (high pressure injection -HPI). The system consists of three letdown coolers, a pressure-reducing block orifice, a liquid radiation monitor, two prefilters, two mixed bed demineralizers, two post filters, a batch controller, a makeup tank, three makeup pumps and their associated lube oil pumps, two seal injection filters, two seal return coolers, and various air- and motor-operated control and isolation valves required for system operation and interface with other systems. As part of the EPU project, a three-way bypass valve around the makeup tank will be installed. However, the design of the bypass valve will not block flow to the make-up pumps. Therefore, no changes to the MU System model are required as a result of the EPU.

Power Conversion System (PSA System Model PCS)

The Power Conversion System (PCS) model includes the Main Steam, Condensate, Feedwater, and Circulating Water Systems. The description of the PCS is divided into two functions: steam removal and feedwater addition. The steam removal portion addresses the main steam and condenser including the Circulating Water System. The feedwater addition function includes the Feedwater and Condensate Systems. The Main Steam, Condensate, and Feedwater System will be extensively modified to support the EPU. However most of the changes are limited to increasing the size or capacity of the various components. The changes that are required to be reflected in the CR-3 PSA model for the PCS as a result of the EPU are very limited in scope. These changes consist of two new heat exchangers (CDHE-7A and 7B) in the condensate flowpath that provides an additional flowpath of condensate in the event of a spurious isolation of condensate valves, new flow control valves (CDV-364 and CDV-365) downstream of the condensate pumps and new recirculation valves (CDV-366 and CDV-367). The heat exchangers are to be used to capture previously rejected heat from the Moisture Separator Reheater Drains. The new condensate heat exchangers are neglected as a potential flow path. Adding the additional path would result in a negligible reduction in CR-3 CDF and LERF. The new flow control valves are added as a potential failure of their respective condensate pump trains. The new recirculation valves are not modeled as they are sized to maintain a minimum pump discharge of 2500 gpm which is small enough compared to the pump capacity that the ability of the pumps to remove decay heat would not be challenged by a flow diversion through the recirculation lines. Other changes are modifying the ADVs (MSV-25 and MSV-26) to be safety-related with a safety-related air supply. The ADV changes have been incorporated into the CR-3 PSA model for the EPU. These modifications result in a negligible reduction in the CR-3 CDF and LERF.

Reactor Building Cooling (PSA System Model RBC)

The reactor building emergency cooling system limits post-accident ambient pressures and temperatures to design values. The reactor building air recirculation and cooling units, backed up by the reactor building spray system, are used during post-accident conditions. When operated in conjunction with the Reactor Building Spray System, the reactor building emergency cooling system can provide design basis heat removal capability. No changes to the reactor building cooling system are planned as a result of the EPU and no changes to the systems' success criteria were identified. Therefore, no changes to the RBC System model are required as a result of the EPU.

Reactor Coolant System (PSA System Model RCS)

The RCS model addresses the pressurizer relief valves and the reactor protection system (RPS). The relief valves are provided to mitigate pressure challenges in the RCS and are used to remove heated water from the RCS during "feed-and-bleed" core cooling operations. The RPS is responsible for reactor shutdown following a reactor trip condition. This EPU revision does not impact the RCS as modeled in the CR-3 PSA system model. Therefore, no changes to the RCS or RPS system models are required as a result of the EPU.

Decay Heat Sea Water System (PSA System Model RWDC)

The Nuclear Services and Decay Heat Seawater (RW) System is comprised of two sub-systems, the Nuclear Services Sea Water (RW-SW) subsystem (cooling the Nuclear Service Closed Cycle Cooling (SW) System) and the Decay Heat Sea Water subsystem (RW-DC) (cooling the Decay Heat Closed Cycle Cooling (DC) System that cools the decay heat (DH) System). The RW-DC system is safety-related and serves as the primary means of transferring heat from the DH System to the ultimate heat sink (Gulf of Mexico). Although the RW-DC and RW-SW Systems are two different systems, the pumps share a common suction pit (one per train) and inlet piping along with common piping returning the water to the ultimate heat sink. No changes to the RW-DC or the RW-SW systems or associated systems are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the RWDC system models are required as a result of the EPU.

Nuclear Service Sea Water (PSA System Model RWSW)

The RW-SW System is safety-related and serves as the primary means of transferring heat from the SW system to the ultimate heat sink (Gulf of Mexico). The RW-SW is an open system comprised of normal duty and emergency duty pumps. The normal duty is a single pump supplying the system, whereas the emergency duty has two pumps each providing cooling to the RW/SW heat exchangers. Although the RW-DC and RW-SW Systems are two different systems, the pumps share a common suction pit (one per train) and inlet piping along with common piping returning the water to the ultimate heat sink. No changes to the Raw Water Service Water system or associated systems are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the RW-SW system models are required as a result of the EPU.

<u>Nuclear Services Closed Cycle Cooling System (PSA System Model NCCC)</u>

The Nuclear Services Closed Cycle Cooling System, or Service Water (SW) System, removes heat from various components and transfers this heat to the nuclear services raw water system (RW-SW). The SW System is a closed loop system. The system consists of four heat exchangers, a normal duty pump, two emergency duty pumps, two booster pumps, a surge tank, two filters, a radiation monitor, a post-accident sampling system cooler, a demineralizer, and system heat loads. The system also contains a number of safety-related valves used to align or isolate SW to selected components during accident/emergency conditions. The system removes heat from various components and transfers the heat to the RW System. One pump is a normal duty pump (SWP-1C) that is normally in operation. The other two pumps (SWP-1A and SWP-1B) are emergency duty pumps that align either on low header pressure or ES actuation. The low pressure start is blocked if the supporting diesel is running. The emergency duty pumps differ from the normal duty pump. Each emergency duty pump "assembly" is comprised of two half-sized pump units driven by one full-sized motor, and has twice the capacity of the normal duty pump. No changes to the Nuclear Services Closed Cycle Cooling System or associated systems are planned as a result of the EPU and no changes to the system success criteria were identified. Therefore, no changes to the NCCC system models are required as a result of the EPU.

<u>INADEQUATE CORE COOLING MITIGATION SYSTEM (ICCMS)</u>

The ICCMS is a new system that has two mitigation functions. First function is to actuate on a sustained loss of subcooling coincident with a reactor trip, which trips the reactor coolant pumps (RCPs) to protect core inventory and raises the OTSG level setpoint to promote natural circulation. The second function is initiate fast cool down system on a sustained loss of subcooling coincident with a reactor trip and a sustained indication of inadequate HPI flow which opens the main steam atmospheric dump valves and controls them at a reduced steam generator pressure. The ICCMS consists of three initiation channels of instrumentation and two actuation trains. Each actuation train monitors outputs of all three initiation channels and a train actuation occurs on any two initiation channels or a manual initiation. The actions to trip the RCPs and raise the OTSG level setpoint were previously manual actions in the PRA, so adding an automatic function that performs the same function will cause a reduction in the CDF.

<u>FAST COOLDOWN SYSTEM (FCS)</u>

A Fast Cooldown System (FCS) is being added to help mitigate a PRA Medium Break LOCA (i.e., design-basis Small Break LOCA). The system will automatically depressurize the secondary side of the OTSG, which facilities rapid depressurization the RCS thereby improving ECCS flow into the core. The break size that the FCS will mitigate is around 0.15 ft2 which is classified as a medium break LOCA in the PRA. Plant specific PRA analysis indicate that the FCS system is not required to successfully respond to a medium break LOCA, however, the success criteria is conservatively changed to require FCS. Adding the FCS requirement to the success criteria for a medium LOCA will introduce new failure modes and therefore cause the CDF contribution from medium break LOCAs to increase. The FCS is a new system that utilizes both of the safety-related ADVs. When fast RCS cooldown is required, control of ADVs is automatically transferred to the FCS pressure controllers upon an FCS actuation signal. The FCS controller automatically modulates the ADVs to maintain the associated OTSG pressure at a nominal FCS setting. The FCS also includes DC power and air supply for the system operation.

Human Reliability Analysis (HRA)

This section describes the assessment of how the EPU could affect the HRA performed for the CR-3 PRA. Over 70 unique post-initiator operator actions are developed for the CR-3 PRA. The vast majority of these events are not impacted by the EPU.

Summarized in Table 2.13-1 are the changes made to the probability values of human failure events (HFEs) in the 2009a PRA model. Changes to the probability values are related to reduce time available to perform the action given the higher decay heat values expected post-EPU.

The Human Error Probability (HEP) for the HRA events were quantified with the Electric Power Research Institute HRA Calculator using the HCR/ORE/THERP method. The methodology assigns median HEP for timing for control room personnel. Reducing the system time window from 60 to 55 minutes changes the timing factor for the human error probability resulting in a higher failure rate for actions. Reducing the System Time Window from 60 minutes (used in the current analyses) to 55 minutes, the HEP for these events are recalculated with the results shown in Table 2.13-1, Human Error Probability (HEP), 60 Min vs. 55 Min System Time Window Comparison.

HRA Dependency Evaluation

The dependent HFE probability values were updated to account for the revised HEPs shown in Table 2.13-1. None of the timing changes expected for post-EPU conditions was determined to affect the dependency level for existing HFE combinations. The updated dependent HEPs were included in the quantification process.

Turbine Missile Generation

Changes to the turbine implemented as part of the EPU do not result in any increase to turbine missile frequency (see Section 2.5.1.2.2, Turbine Generator).

Seismic Events

An IPEEE seismic analysis was not performed for the external events IPEEE. The basis for not performing a seismic evaluation was that seismic events at CR-3 would be a probabilistically low contribution to overall risk and that the resolution of Unresolved Safety Issue A-46, Seismic Qualification of Equipment in Operating Plants, is considered to be sufficient to address the seismic aspects of the IPEEE. Given the low risk of seismic events at CR-3, it is concluded that the EPU results in a negligible change to the risk posed by seismic events.

Internal Fire Events

Based upon a review of the EPU related modifications most have a negligible impact on fire risk. For example, replacement of turbine bypass valves or atmospheric dump valves or heat exchangers would not impact the fire CDF due to the low heat release rates of the added components, and the negligible impact on fire ignition frequency. Similarly, negligible changes in overall fire loading were identified. Although the 17R EPU design modifications are not mature enough to have the required information such as cable routing to accurately quantify the effects on the fire PRA model, the increase to fire CDF is not expected to be significant.

The evaluation of both ADVs spurious opening within a short mission time causing an overcooling event in both SG is in the range of 4E-14 for their failure which is below the threshold for modeling in the internal events PRA model. The lockout modification will have an insignificant increase in the failure of the FCS by introducing additional contacts that could fail open. In the case of a fire causing both ADVs to spuriously open, a lockout design is required to disable the ADVs in the event of a significant fire in the control complex. At the time of the PRA LAR evaluation, the cable routing and component location had not been finalized, therefore the impact on fire risk could not be quantified. Also the timing analysis for the new operator action to lock out the ADV actuation had not been performed, therefore the fire HRA value could not be determined.

Since only negligible changes in fire ignition frequency and fire loading occur as a result of the EPU, an evaluation of the affect on Fire PRA based upon HRA timing was preformed. The updated EPU HRA values were applied to the fire PRA model. The fire HRA methodology was applied to the EPU HRA values and the results were incorporated in the recovery rule file. Since the HRA timing changes due to EPU are insignificant, it is concluded that there is no significant change in overall fire risk as a result of the EPU.

High Winds, Floods, and Other External Events

The external events high winds, tornadoes, external floods, and transportation and nearby facility accidents were evaluated as part of the IPEEE. Using bounding analyses, these evaluations determined that these events would have minimal impact on the overall risk of core damage at CR-3. For the EPU, no changes were identified that would impact these external events.

Level 2 Analysis

The CR-3 PSA model includes a full Level 2 PSA model as well as a LERF model. LERF is quantified as a subset of the total release determined from the Level 2 PSA model. The Level 2 and LERF models employ the same logic models as the Level 1 model. The logic changes made to address the EPU for Level 1 are carried into the Level 2 models.

Shutdown Risk

A quantitative shutdown risk model is not maintained for CR-3. Therefore, a quantitative assessment of risk cannot be performed. For shutdown modes, a defense-in-depth risk management scheme is used and shutdown safety assessments are performed deterministically following the guidelines in NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management. Implementation of the shutdown safety assessments is done procedurally using checklists. One of the considerations in the checklists is the time to the onset of boiling following a loss of shutdown cooling. While the time to boiling is decreased early in an outage given the higher decay heat levels post-EPU, existing procedures consider the shorter times when the curves used to determine boiling are revised. No other changes related to the EPU affect any layer of defense considered in the shutdown safety assessments. Therefore, it is concluded that the EPU has no significant impact on shutdown risk.

PSA Results

The CR-3 PSA internal events and Fire PRA models were quantified after incorporating the changes for EPU as described above. The results of the quantification are shown in Table 2.13-2, Risk Results Without Risk Reduction Modifications, and were contrasted with the results from the 2009a model and the Fire PRA for CR-3.

Contribution to core damage by initiating events is given in Table 2.13-3. The dominate small LOCAs sequences are the failure of long term cooling. The reactor vessel rupture is a single event fault. The transients include reactor trips, loss of feed water, and over feed events.

In summary, the proposed CR-3 EPU:

- is consistent with the defense-in-depth philosophy and is maintained with the proposed EPU as discussed in Reg. Guide 1.174;
- maintains sufficient safety margins; and
- results in a small increase in core damage frequency and Large Early Release Frequency. These changes are consistent with the intent of the Commission's Safety Goal Policy Statement.

2.13.3 Conclusion

CR-3 has reviewed the assessment of the risk implications associated with implementation of the proposed EPU and concludes that the potential impacts associated with the implementation of the proposed EPU are adequately modeled and/or addressed. CR-3 further concludes that the results of the risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, CR-3 finds the risk implications of the proposed EPU acceptable.

2.13.4 References

1. U.S. Nuclear Regulatory Commission, Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance, NUREG-0800, Standard Review Plan Chapter 19.0, Revision 1, November 2002.

Table 2.13-1
Human Error Probability (HEP)
60 Min vs. 55 Min System Time Window Comparison

Basic Event	Event Description	Base Value (60 Min)	Updated EPU Value (55 Min)	Percent Increase
AHU4KVXY (60)	Operators Fail to Power 4 kV ES Bus from Alternate Offsite Source	8.7E-03	1.4E-02	61%
AHUE3ABY (60)	Operators Fail to Switch Power Source to ES MCC 3AB	1.1E-02	1.6E-02	45%
AHUEGDGY (60)	Operators Fail to Start EGDG Manually	9.9E-02	1.4E-01	41%
AHUMT2HY (60)	Operators Fail to Align MTSW-2G To MTSW-2H	1.6E-01	2.2E-1	38%
HHUMBACY (60)	Operators Fail to Switch MUP-1B Power Source	1.1E-01	2.4E-1	118%
HHUMPSBY (T)(60)	Operators Fail to Start Non-ES Selected Makeup Pump (Transients)	4.2E-03	5.0E-3	19%
QHUEFW9Y (60)**	Operators Fail to Raise OTSGs Level or Achieve Adequate Flow Given Loss of Adequate Subcooling Margin	2.2E-03	2.3E-03	5%

** The ICCM System will automatically raise OTSG level; therefore the existing HRA action will be “ANDED” with the auto function which makes it a recovery action. AREVA analysis indicate this action must be done within a 10 minute time window to maintain core temperature within limits.

Table 2.13-2
Risk Results Without Risk Reduction Modifications

	Base Prior to EPU	EPU Results	Delta Increase
Core Damage Frequency	3.4E-06	3.5E-06	1.0E-7
Large Early Release Frequency	1.2E-07	1.2E-07	-
Non Multi Compartment Fire CDF	3.28E-5	3.29E-5	1E-7

Notes:

The change in CDF and LERF both fall inside the Region III Acceptance Guidelines of RG 1.174. Refer to Figure 3 and Figure 4 of Regulatory Guide 1.174 for Acceptance Guidelines.

Table 2.13-3
Contribution to CDF by Initiating Events

IE		2009a Model	EPU Model
IE_S	Small LOCAs	43.0%	30.5%
IE_T	Transients	19.8%	20.4%
IE_Z	Reactor vessel rupture	14.8%	14.8%
E_F	Internal Flooding	10.0%	10.0%
IE_M	Medium LOCA	2.4%	14.3%
IE_A	Large LOCA	6.6%	6.6%
IE_R	SGTR	3.2%	3.2%
IE_V	ISLOCA	0.2%	0.2%

Table 2.13-4
EPU HRA with Fire Methodology Applied

HRA	Description	EPU Fire	Pre-EPU Fire
AHU4KVXY (60)	Operators Fail to Power 4 kV ES Bus from Alternate Offsite Source	1.0	1.0
AHUE3ABY (60)	Operators Fail to Switch Power Source to ES MCC 3AB	1.9E-02	1.4E-02
AHUEGDGY (60)	Operators Fail to Start EGDG Manually	1.0	1.0
AHUMT2HY (60)	Operators Fail to Align MTSW-2G To MTSW-2H	2.4E-01	1.9E-01
HHUMBACY (60)	Operators Fail to Switch MUP-1B Power Source	2.6E-01	1.3E-01
HHUMPSBY (T)(60)	Operators Fail to Start Non-ES Selected Makeup Pump (Transients)	1.7E-2	1.6E-02
QHUEFW9Y	Operators Fail to Raise OTSGs Level or Achieve Adequate Flow Given Loss of Adequate Subcooling Margin	2.2E-03	2.2E-03

2.14 The Effects of EPU on the Renewed Licensing and License Renewal Programs

2.14.1 Regulatory Evaluation

The CR-3 License Renewal Application (LRA) was submitted to the NRC for review and approval on December 16, 2008 (ADAMS Ascension Number ML090080054). The LRA reflects the CR-3 plant configuration and current licensing basis immediately prior to its submittal in December 2008. The application will be kept current in accordance with 10 CFR 54. In particular, 10 CFR 54.21(b) requires that each year following the submittal of the license renewal application and at least three months before the scheduled completion of the NRC review, an amendment to the renewal application must be submitted. The amendment to the renewal application identifies any changes in the facility that materially affects the contents of the license renewal application, including the FSAR supplement.

CR-3 Current Licensing Basis

Since the CR-3 LRA is being concurrently reviewed by the NRC, it has not been incorporated into the CR-3 licensing basis. The EPU License Amendment Request (LAR) does not rely on the approval of the LRA nor does the LRA rely on any aspect of the EPU being approved. Similarly, the approval of the LRA prior to the EPU LAR would not have any impact on the EPU review.

2.14.2 Technical Evaluation

Introduction

Since the LRA is not yet approved, there is limited overlap between to two activities. Those aspects which might have some level of potential overlap are addressed below.

Description of Analyses and Evaluations

Both projects impact the Environmental Qualification (EQ) of Electrical Equipment. The evaluations in support of the LRA focus on an extended period of normal operations (60 years as opposed to 40 years). The license renewal impacts were fully evaluated in support of that application and the design documents were conservatively updated to reflect that impact without waiting for NRC approval of the LRA. EPU adds additional EQ impact. Primarily, the normal and accident dose rates are higher. The EPU impacts are discussed in Section 2.3.1, Environmental Qualification of Electrical Equipment.

Both projects impact various aspects of reactor vessel, and to a lesser extent, other Reactor Coolant System materials issues. These materials impacts are discussed in the appropriate sections of the LRA and this submittal. Reactor vessel material reviews are dependent on integrated neutron fluence. Both the time at power (measured in effective full power years and related directly to the period of extended operation) and power level (EPU) impact the neutron fluence. Such fluence evaluations and associated materials analyses are costly, time consuming, and complex. Therefore, the analyses performed in support of the LRA as well as EPU addressed the combined impacts. Nevertheless, approval of either request is not dependent on approval of the other and simply reflects more conservative materials limitations than would be otherwise required for either project alone. Additionally, maintaining the existing Low Temperature Overpressure (LTOP) Power Operated Relief Valve (PORV) setpoint until this time continues to provide appropriate low-temperature protection for EPU conditions. Implementation of the revised LTOP PORV setpoint will require a separate LAR submittal requesting NRC approval of an amendment to the CR-3 Improved Technical Specifications, specifically Section 3.4.11. Submittal of this LAR will be made at least 12 months prior to reaching 27.5 EFPY (relative to EPU power).

Another area of potential overlap is License Renewal Scope. One EPU impact on scope is the addition of the Low Pressure Injection Cross-Tie and Hot Leg Injection flow paths (refer to Appendix E for discussion of this modification). These additional flow paths will likely be added to the License Renewal scope. Similarly, any future modification would be reviewed for its potential impact to the license renewal scope.

One final area of potential overlap is associated with License Renewal required programs. The programs necessary to support the period of extended operation are created or modified, as necessary, to meet 10 CFR 54 requirements. None of these programs are impacted at the programmatic level by EPU. The technical content of some program implementing details (e.g., related to EQ, Flow Accelerated Corrosion) will change as a result of different EPU conditions. Each of these EPU impacts are discussed in appropriate detail in other sections.

2.14.3 Conclusion

As noted above, the EPU LAR in does not rely on the approval of the LRA nor does the LRA rely on any aspect of the EPU LAR being approved. The approval of the LRA prior to the EPU LAR would not have any impact on the EPU. Both can be and have been evaluated as independent activities.

2.14.4 References

None

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #309, REVISION 0

APPENDIX A

SAFETY EVALUATION REPORT COMPLIANCE

Appendix A: Safety Evaluation Report Compliance

A.1 Safety Evaluation Report Compliance Introduction

This Appendix is a summary of NRC-approved codes and methods used in Section 2.8.5, "Accident and Transient Analyses" for the CR-3 EPU. The appendix addresses compliance with the limitations, restrictions, and conditions specified in the approving safety evaluation of the applicable codes and methods.

Table A.1-1 presents an overview of the Safety Evaluation Reports (SER) by codes and methods. For each SER, the applicable report subsections and appendix subsections are listed.

References

A.1-1 BAW-10193-PA (Proprietary), Rev. 0, RELAP5/MOD2-B&W For Safety Analysis of B&W Designed Pressurized Water Reactors.

A.1-2 BAW-10156-A, Rev. 1, "LYNXT, Core Transient Thermal-Hydraulic Program."

A.1-3 BAW-10192PA (Proprietary), Rev. 0, and BAW-10192-A, Revision 0, (Nonproprietary), "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants."

A.1-4 ANP-2788P, Rev.0 (Proprietary), “Crystal River 3 Rod Ejection Accident Methodology Report” (Submitted as an attachment to License Amendment Request #307, DPR-72, Approved January 28, 2010 as Amendment No. 237).

Table A.1-1: Safety Evaluation Report Compliance Summary

Subject	Topical Report (Reference) / Date of NRC Acceptance	Code(s)	Limitation, Restriction, Condition	Report Section(s)	Appendix Section
Non-LOCA Safety Analysis	BAW-10193-P-A (Reference A.1-1) 10/15/1999	RELAP5/MOD2-B&W	Yes	2.8.5.1.2, 2.8.5.2.1, 2.8.5.2.3, 2.8.5.2.4, 2.8.5.3.1, 2.8.5.3.2, 2.8.5.4.1, 2.8.5.4.2, 2.8.5.4.3, 2.8.5.4.4, 2.8.5.4.5, 2.8.5.4.6, 2.8.5.6.2	A.2
Transient DNB Analysis	BAW-10156A-01 (Reference A.1-2) 1/4/1993 Revised SER conclusions accepted 3/26/1993	LYNXT	Yes	2.8.5.3.1, 2.8.5.3.2	A.3
App K LOCA	BAW-10192-P-A (Reference A.1-3) 2/18/1997	RELAP5/MOD2-B&W REFLOD3 CONTEMPT BEACH	Yes	2.8.5.6.3	A.4
Control Rod Ejection	ANP-2788P (Reference A.1-4)	RELAP5/MOD2-B&W LYNXT COPERNIC NEMO-K	No	2.8.5.4.6	N/A

A.2 Non-LOCA Analysis (RELAP5/MOD2-B&W)

Table A.2-1: RELAP5/MOD2-B&W for Non-LOCA Events

Limitations, Restrictions, and Conditions
1. RELA 5/ 2 ill e applied consistent ith the current application o A and TRA 2. In addition the pu p coastdo n characteristics ill e deter ined via RELA 5/ 2 pu p odel. Justification: AREVA performed the RELAP5/MOD2-B&W analyses with no change to the basic methodology described in the licensee's Safety Analysis Report. The pump coastdown characteristics for the one- and four-pump coastdown transients were determined via the RELAP5/MOD2-B&W pump model.
2. I analysis is or an IE T a ne L F ench ar analysis ust e su itted to and revie ed y the R . Justification: The steam generators incorporated into the CR-3 plant design are not the IEOTSGs design. Therefore, no benchmark analyses were required to support the CR-3 EPU.
3. A 1 16 A Use o the allis and U TF para eters at the tu e undle and stea generator plenu inlet is accepta le. The para eters used in the FL odel or any other application ust e validated and the validation revie ed and approved y the sta or that application see A 1 16 A pages 5 33 and 5 335 . Justification: This is a limitation on the junction at the inlet to the tubes in a recirculating steam generator plant. It is not applicable to the geometry in the B&W-designed plants and is therefore not used.

A.3 Transient DNB Analysis (LYNXT)

Table A.3-1: LYNXT for Transient DNB Analysis

Limitations, Restrictions, and Conditions
1. hen L T Rev. 1 is applied to the stea line rea analysis and i the uncertainty in cross lo resistance is not s all then the i plicit algorith ust e used. *This restriction as clari ied in correspondence ollo ing issuance o the ER. The resulting revision to ER onclusion 1 as that the use o the algorith should e li ited y the range o application o ey para eters and not analysis type. The application o the L T Rev. 1 algorith is restricted to the ollo ing ranges* *ass lu a solute . to 3. l /hr t^2 here denotes 1 ^{6}* *yste ressure 5 to 3 psia* *Local heat lu . to . tu/hr t^2* Justification: The PV algorithm was used for the loss of flow analyses described in Section 2.8.5.3.1 and 2.8.5.3.2. More restrictive ranges of applicability have been incorporated into the LYNXT version (V29.0) used for the CR-3 EPU, such that conditions outside the range of applicability would be flagged as an error. For the loss of flow analyses for the CR-3 EPU, the analyses remained within the range of applicability and no further justification is required.
2. hen L T Rev. 1 incorporating the algorith is used the licensee is responsi le or veri ying the ade uacy o the cross lo resistance chosen henever reverse and recirculation lo s are o served in the analysis. Justification: The loss of flow events do not result in reverse or recirculation flows up to the time the minimum DNBR is observed. Therefore, no further justification is required for the crossflow resistance chosen.
3. A ong the 11 F correlations availa le in L T Rev.1 only the correlation has een validated ith L T Rev 1 hen the algorith is used. There ore only the can e used or analysis ith L T Rev.1 hen the i plicit algorith is used. The applications are restricted to the ranges o applica ility o these correlations. *This restriction as clari ied in correspondence ollo ing issuance o the ER. The resulting revision to ER onclusion 3 as that additional F correlations ay e used provided that the application o the correlation is restricted to its range o applica ility.* Justification: The revision to SER Conclusion 3 specifically mentions the B&W-2, BWC, BWCMV, and W3 CHF correlations as acceptable for use with the PV algorithm, with each of these correlations having been previously approved by the NRC. The revision to Conclusion 3 notes that comparisons between the LYNXT PV algorithm produced essentially the same BWC DNBRs (and thus essentially the same local conditions) as the original COBRA-IV-I implicit solution. It is further noted that if local conditions are the same between the two algorithms then the DNBR predictions for any CHF correlation will be the same. The BHTP CHF correlation was used for the CR-3 EPU. Use of the

Limitations, Restrictions, and Conditions
BHTP correlation is acceptable because it has been approved by the NRC (Reference A.3-1), and because it has been shown that the PV algorithm produces essentially the same local conditions when compared to previously accepted COBRA-IV-I implicit solution. The use of an approved CHF correlation, applied within the range of applicability produces an acceptable result. The range of applicability of the BHTP CHF correlation is defined in Reference A.3-1. The LYNXT code (V29.0) flags operation outside the range of applicability and denotes the results as invalid. The CR-3 EPU analyses remained within the range of applicability for the BHTP CHF correlation.
. *This sa ety evaluation restricts the use o the algorith to the analysis o the Flo oastdo n Transient and the stea line rea analysis. ther transient analyses to hich the algorith s ay e applied ill re uire appropriate ench ar ing y the licensee.* *This restriction as clari ied in correspondence ollo ing issuance o the ER. The resulting revision to ER onclusion states that the ER conclusion is to e replaced y revised onclusion 1 su ari ed a ove .* Justification: No further justification required.

References

A.3-1 AREVA NP Topical Report BAW-10241-P-A, Revision 1, "BHTP DNB Correlation Applied with LYNXT." (Acceptance Letter Dated 7/25/2005, [Rev. 1], 9/29/2004 [Rev.0].

A.4 RELAP5/MOD2-B&W for LOCA Events

Table A.4-1: RELAP5/MOD2-B&W for LOCA Events

Limitations, Restrictions, and Conditions
1. The L A ethodology should include any R restrictions placed on the individual codes used in the evaluation odel E . Justification: For LBLOCA analyses, the RELAP5/MOD2-B&W (includes BEACH), the REFLOD3B and CONTEMPT codes are utilized. For SBLOCA analyses, only the RELAP5/MOD2-B&W and CONTEMPT (for CFT analyses only) codes are utilized. The NRC restrictions placed on the codes used in the BWNT LOCA EM have been reviewed. All items were in compliance with the NRC restrictions except for the BEACH initial cladding temperature. An additional FLECHT benchmark was performed to extend the range of validity for the BEACH initial clad temperature. The additional benchmark has been approved by the NRC (BAW-10166P-A, Rev.5).
2. The guidelines code options and prescri ed input speci ied in Ta les 1 and 2 in oth olu e I and olu e II o A 1 1 2 should e used in L L A and L A evaluation odel applications respectively. Justification: Compliance with Table 9-1 & 9-2 in Volume I (LBLOCA) of BAW-10192P-A and Table 9-1 & 9-2 in Volume II (SBLOCA) of BAW-10192P-A has been verified and documented for EM applications including the CR-3 EPU. Compliance to the Table 4 restrictions for the LBLOCA analyses and the Table 6 restrictions for the SBLOCA analyses was also verified and documented within the LBLOCA and SBLOCA analyses performed for EPU. These tables also include inputs and restrictions placed on the individual codes that make up the BWNT LOCA.
3. The li iting linear heat rate or L A li its is deter ined y the po er level and the product o the a ial and radial pea ing actors. An appropriate a ial pea ing actor or use in deter ining L A li its is one that is representative o the uel and core design and that ay occur over the core li eti e. The radial pea ing actor is then set to o tain the li iting linear heat rate. For this de onstration calculations ere per or ed ith the a ial pea o 1. . The general approach is accepta le or de onstrating the L A li its ethodology. o ever as uture uel or core designs evolve the asic approaches that ere used to esta lish these conclusions ay change. ARE A ust revalidate the accepta ility o the evaluation odel pea ing ethods i 1 signi icant changes are ound in the core elevation at hich the ini u core L A argin is predicted or 2 the core aneuvering analyses radial and a ial pea s that approach the L A L R li its di er apprecia ly ro those used to de onstrate Appendi co pliance. Justification: This restriction is related only to LBLOCAs, which set the LOCA linear heat rate limits. The axial and radial peaks used in the LBLOCA analyses are in accordance with the EM, with an axial peaking factor of 1.7 for all elevations and linear heat rates analyzed. Section I.A of Appendix K to 10 CFR 50 requires that LOCA analyses are performed with the maximum peaking factors allowed by the technical specifications. It also requires that a range of power distribution shapes and peaking factors representing power distributions that may occur over

Limitations, Restrictions, and Conditions
the core lifetime be studied, and the one selected should be that which results in the most severe calculated consequences for the spectrum of postulated breaks and singe failures analyzed. Compliance with 10 CFR 50 Appendix K.I.A can be achieved through control of maximum radial or axial peaks versus core elevation. However, limiting the core design allowable radial or axial peaks because of LOCA results may be difficult if not impossible given the variations observed in the calculated PCTs. For example, axial peaks for the core inlet-skewed shapes must be greater than 1.7, while core exit-skewed shapes must be less than 1.7, in order to assure that limiting PCTs are predicted for either case. Moreover, each influences the core mid-plane peak and its saddle-shaped PCT behavior. The mid-plane PCTs can worse with either increase or decrease in axial peak. These considerations either make the core design too restrictive or drastically complicate the LOCA analyses needed to demonstrate compliance. Four steps were developed to show compliance or to define a LOCA linear heat rate (LHR) limit penalty. These steps provide guidance for determining those locations where the resulting axial peaking may be outside the bounds (i.e., augmented) of those considered in the LOCA analyses. These four steps are summarized below. 1. The fuel burnup must be compared to the LOCA LHR limits versus burnup. If the burnup is on the PCT-limited portion of the LOCA limit curve (≤ 40,000 MWd/mtU), then proceed to Step 2. If the burnup range is on the pin pressure-limited portion of the curve (> 40,000 MWd/mtU), the restriction is met without any other conditions. That is, no axial peaking checks or linear heat rate limit adjustments are needed for pin pressure-limited LHRs. 2. If the burnup is on the PCT-limited portion of the curve, then the power distribution analysis LOCA margins must be checked at all core elevations. If there is less than 5% LOCA margin, proceed to Step 3. If there is more than 5% margin, the restriction is met and no further checks are needed because the PCT at the maximum power distribution LHR will be lower than the BWNT LOCA EM PCT. 3. If the burnup is on the PCT-limited portion of the curve and there is less than 5% LOCA margin, then variations in the augmented peaking factor versus the 1.7 axial used in the LOCA analyses must be considered. The axial peak must be 1.65 or greater for 0 to 4 ft power peak elevations, 1.7 +/- 0.05 for 4 to 8 ft elevations, and 1.75 or less for 8 to 12 ft elevations. If these axial peaks are in compliance, the restriction is met and no further checks are needed. If they are not met, then proceed to Step 4 for the LOCA LHR limit reductions. 4. If the burnup is on the PCT-limited portion of the curve, there is less than 5% LOCA margin, and the axial peak is not in compliance, then the power distribution analysis must assign a LOCA LHR limit penalty to ensure that the BWNT LOCA EM PCT (based on the given LHR and axial peaking factor of 1.7) is not underpredicted. The LHR limit penalty compensates for the known deviation between the augmented axial peak and the required peak. Appropriate LHR reductions are thus provided for those core locations where the augmented axial peaks are not bounded by the LOCA analyses. Applying the necessary LHR reduction to the core

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operating limits ensures that the consequences of the LOCA are not underpredicted.
. The echanistic E ypass odel is accepta le or cold leg transition . 5 t^2 to 2. t^2 and hot leg rea calculations. The non echanistic E ypass odel ust e used in the large cold leg rea ≥2. t^2 ethodology since the de onstration calculations and sensitivities ere run ith this odel. Justification: As outlined in BAW-10192P-A Volumes I and II, different bypass models are used for large break and small break analyses. The nonmechanistic ECCS bypass model is used in large break analyses (≥ 2.0 ft). The mechanistic ECCS bypass model is used for cold leg transition (0.75 ft to 2.0 ft), hot leg, and all smaller sized cold leg breaks. As presented in Sections 4.2 and A.6.3 of Volume II of the EM, the minimum break size range for cold leg transition breaks is determined based on those breaks that show initial clad DNB, which is the 0.5 ft^2 break for the Mark-B-HTP CR-3 with ROTSGs at EPU power level.
5. Ti e in li e L A li its ust e deter ined ith or sho n to e ounded y a speci ic application o the R approved evaluation odel. Justification: Time-in-life studies for LBLOCAs have determined that similar LOCA limits are obtained from BOL to approximately 40 to 45 GWd/mtU. After approximately 40 to 45 GWd/mtU, the LHR must be dropped to reduce both the pin pressure and stored energy in the fuel. The plant-specific applications will use BOL conditions in the average channel to conservatively bound the fuel stored energy and oxide thickness for metal water energy generation for the entire range of fuel pin burnup. For the hot channel, conditions appropriate to the time-in-life will be used. Time-in-life calculations for SBLOCA applications, which use a conservative composite set of reactivity parameter bounding for all TILs, are not required unless the fuel pin heatup is sufficient to cause cladding rupture. For the EPU analyses, AREVA used a method to explicitly examine times in life and the likelihood of rupture and its effect on the PCT for each case. The method used three supplemental pins with a plastic weighted heating ramp rate option, BOL fuel temperatures, and BOL initial oxide thicknesses. The hot channel is set to the pin pressure limit at EOL. The three supplemental pins use pin pressures consistent with BOL and two pressures roughly uniformly distributed between the BOL and EOL values. Clad rupture at cladding temperatures less than approximately 1600 F allows increased cooling because of the clad surface area increase. At these temperatures the metal-water reaction is not significant, therefore rupture is a beneficial event that if avoided will produce higher PCTs. For higher cladding temperatures where the metal-water reaction contributes to the peak clad temperature, the pin pressure variation will ensure that clad rupture is obtained at the most limiting time during the transient. To maximize the cladding temperatures, the BOL fuel stored energy and BOL oxide thicknesses are used. While these assertions are based on studies performed with Zr-4 cladding, they are equally applicable to M5 cladding, because the rupture behavior and metal-water reaction are not significantly different between the cladding materials. A pure TIL calculation (with TIL-specific reactivity inputs, fuel stored energy, pin pressure, and cladding oxide thickness consistent with the TIL that produces the worst rupture time) would be

Limitations, Restrictions, and Conditions

performed if the composite case is judged to be overly conservative. The consistent case would also use the plastic-weighted normalized heating ramp rate to predict the fuel pin swell and rupture performance.

6. L A li its or three pu p operation ust e esta lished or each class o plants y application o the ethodology descri ed in this report. An accepta le approach is to de onstrate that three pu p operation is ounded y our pu p L R li its.

Justification: A LBLOCA analysis of three operating RCPs at a core power of 80 percent full power was performed to demonstrate that three-pump operation is bounded by four-pump LHR limits. The hot channel three-pump peak LHR limit is set equivalent to the 100 percent power 4-pump LHR limit. Because this analysis is performed at a power level less than 95 percent, a positive MTC of +1 pcm/F is considered. The analysis showed that the consequences of the 4 RCP full-power LOCA LHR limit analyses bound those during 3 RCP operation; therefore, the 4 RCP operation LHR limits remain valid for the 3 RCP operation.

Three-pump SBLOCA analyses are not performed because the core power is reduced but the ECCS capacity is not reduced. Therefore, four-pump full-power SBLOCA PCTs will bound the PCTs for similar three-pump partial power cases.

. The li iting E con iguration including ini u versus a i u E ust e deter ined or each plant or class o plants using this ethodology.

Justification: The number of ECCS pumps that are available following a LOCA is dependent upon the offsite power status, number of diesel generators operating, or the equipment failures postulated. Higher ECCS flow is very beneficial for SBLOCAs, but is not as significant for LBLOCAs for the time frame of concern, which is prior to the availability of pumped injection. During a LBLOCA, more ECCS flow can refill the vessel slightly faster, but it can also reduce the containment pressure. Lower containment pressures increase the core steam binding effect and retard the quench front advancement.

The limiting LBLOCA ECCS configuration is a single ECCS train for CLPD breaks. For this application, the minimum containment pressure derived from a maximum ECCS flow configuration was used as a boundary condition in combination with the minimum ECCS flow configuration. This composite approach conservatively considers the worst containment pressure with the minimum ECCS refill capacity to ensure that LBLOCA calculated consequences are bounding for any combination of available ECCS pumps.

In addition to the LBLOCA analyses, a minimum containment pressure analysis using the guidance for LBLOCA applications was performed specifically for use in the SBLOCA CFT line break analyses to ensure that the CFT line break analysis was also conservative.

Limitations, Restrictions, and Conditions
. For the s all rea odel the hot channel radial pea ing actor to e used should correspond to that o the hottest rod in the core and not to the radial pea ing actor o the 12 hottest undles. Justification: There are twelve assemblies modeled in the hot bundle, and each pin is peaked to the hot pin radial value.
. The constant discharge coe icient odel discharge coe icient = 1. re erred to as the igh or Lo rea oiding or ali ed alue should e used or all s all rea analyses. The odel hich changes the discharge coe icient as a unction o void raction i.e. the Inter ediate rea oiding or ali ed alue should not e used unless the transient is analy ed ith oth discharge odels and the inter ediate void ethod produces the ore conservative result. Justification: This restriction is related only to SBLOCA analyses. A constant discharge coefficient is used for SBLOCA analyses. Verification of this input is performed for each SBLOCA analysis.
1 . For a speci ic application o the ARE A s all rea L A ethodology the rea si e hich yields the local a i u T ust e identi ied. In light o the di erent possi le ehaviors o the local a i u ARE A should usti y its choice o rea si es in each application to assure that either there is no local a i u or the si e yielding the a i u local T has een ound. rea si es do n to . 1 t2 should e considered. Justification: This restriction is related only to SBLOCA analyses. The SBLOCA break spectrum (down to 0.005 ft) is performed to determine the local maximum PCT. The break sizes analyzed are chosen to ensure that the local peak has been appropriately defined. The full spectrum of break sizes performed for the Mark-BHTP fuel at EPU conditions covers this requirement.
11. designed plants have internal reactor vessel vent valves R s that provide a path or core stea venting directly to the cold legs. The T L A evaluation odel credits the R stea lo ith the loop stea venting or L L A analyses. The possi ility e ists or a cold leg pu p suction to clear during lo do n and then re or during re lood e ore the evaluation odel analyses predict average core uench. ince the REFL 3 code cannot predict this re or ation o the loop seal ARE A is re uired to run the RELA 5/ 2 syste odel until the hole core uench to con ir that the loop seal does not re or . This de onstration should e per or ed at least once or each plant type raised loop and lo ered loop and e udged applica le or all L L A rea si es. Justification: This restriction is related only to LBLOCA analyses. This verification analysis has been performed using the RELAP5 system model for the 177-FA LL plant design. The results of that analysis confirmed that a loop seal does not reform prior to whole core quench. Since these results were obtained using the 177-FA LL model, it can be concluded that Restriction #11 of the evaluation model is met for the CR-3 plant.

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APPENDIX B

ADDITIONAL CODES AND METHODS

Appendix B: Additional Codes and Methods

Numerous analytical codes were used to support the proposed CR-3 Extended Power Uprate. These have been reviewed against the codes currently described in the FSAR. The codes listed below do not currently appear in the FSAR, and are identified for the NRC's information, along with their functional application. All of these computer codes have been determined by CR-3 to be appropriate for use in their respective applications.

CODE	DESCRIPTION
ANSYS	ANSYS is a commercially available general purpose finite element program and is used for structural, seismic and thermal analyses for the replacement steam generators.
ASPEN	Short circuit analysis is run using the ASPEN software package, at post-EPU conditions. This software is used for Grid Stability evaluation.
ATAPP	The computer program ATAPP is a transient analysis post-processor developed by B&W. This computer program is used to calculate stress ranges from the output of the finite element program ANSYS.
AutoPipe	AutoPipe software package is used to perform the pipe stress analysis.
CAFTA	It is a computer based fault tree analysis tool used for Probabilistic Safety Assessment (PSA).
CASMO-4	CASMO-4 performs cell criticality calculations and burnup depletion calculations. This code was used to determine the reactivity effects of tolerances and fuel depletion for spent fuel pool storage.
COMPARE-MOD 1	COMPARE-MOD1 performs transient analysis of the thermodynamic conditions in zero velocity or stagnant volumes connected by flowing junctions with provision for mass and energy addition. This software program is used for calculating HELB pressurization in the IB.
COPERNIC	This software program is used to define cladding corrosion/oxidation properties for accident analyses.
dcVOLTPRO	dcVOLTPRO is used for performing dc system analyses.
Digital Power Train (DPT)	The Digital Power Train (DPT) computer simulation tool is used to predict the performance and behavior of the major components in the B&W nuclear steam system for a wide range of plant conditions and operation. This software program is used to support development of test cases during startup.

CODE	DESCRIPTION
DORT	This software program is used in performing Reactor Vessel Fluence Evaluation.
EasyFIV	This program is used to predict the flow induced vibration response of Steam Generator tubes subjected to single or two-phase cross flow. The program analyzes the potential for Fluid-Elastic Instability (FEI) and Vortex Shedding Resonance.
ETAP	ETAP Electrical Analysis Software is used to generate Electrical Power Distribution System Analysis Results for Safety Related design activities at CR3.
FIVDYNA	FIVDYNA is a non-linear fretting wear program developed by B&W and it accounts for the effects of tube-to-support gaps and tube pre-loads. This software program is used to predict the work rates and the dynamic response of a single tube in a steam generator tube bundle subjected to fluid cross-flow.
FTREX	It is a quantifier program for fault trees and event trees which is employed as a default fault tree solver to convert the core damage fault tree to the minimum cut-sets. This software program is used for Probabilistic Safety Assessment (PSA).
GALE	This software program is used to determine the source-term of normal operations gaseous and liquid effluents.
GT STRUDL	This software program is used to perform design and analysis of the pipe supports at EPU conditions.
MCNP4a	MCNP4a is a three-dimensional continuous energy Monte Carlo code developed at Los Alamos National Laboratory. This software program is used for the criticality analysis of the loaded storage racks in the Spent Fuel Pool.
NRCDOSE	This software program is used to provide conservative estimates of doses to the public to demonstrate compliance with10CFR50, Appendix I criteria for routine effluent releases.
ORIGEN2	ORIGEN2 was used to calculate decay heat from the fuel assemblies in the spent fuel pool and to calculate the source term to be used in dose assessments.
PDBURST	PDBURST is a computer program that calculates the probability that an assumed crack in a turbine disk will grow to the critical depth. This software program is used for Turbine Missile Probability analysis.
PDMISSILE	PDMISSILE is a computer program that calculates the probability of casing penetration given a disk burst up to 120 percent of the rated speed. This software program is used for Turbine Missile Probability analysis.

CODE	DESCRIPTION
PRAQuant	It is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run. This software program is used for Probabilistic Safety Assessment (PSA).
PSS/E	This software is used to simulate single and multiple contingency events on the transmission system, using base cases that model all of Florida and a partially equivalence model of the "outworld" (Southern Company and beyond). This software program is used for Grid Stability evaluation.
QRecover	It is a tool to automatically manipulate cut-sets based on a set of rules. This software program is used for Probabilistic Safety Assessment (PSA).
RELAP5/MOD3	This software program is used to determine replacement steam generators cross flow load and tube support plate loads during MSLB and the tube tensile loads during LBLOCA.
RELAP5/MOD3.2	The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. This software program is used to calculate the IB environmental temperatures for a spectrum of MSLBs.
RETRAN 3D MOD003.1	This software program is used to determine Steam Generator Tube Tensile Loads during MSLB.
RPM	This Radiological Plant Model (RPM) software program is used to perform dose calculations for Equipment Qualifications.
STRATUS	This software program is used for the analysis of tube sheets with triangular perforation patterns. This program calculates stress ranges, based on the method described in the ASME B&PV Code Section III, Appendix A-8000, from the output of the finite element program ANSYS.
THEDA-2	The THEDA-2 program is used to calculate the secondary side properties, such as temperature, quality, void fraction, velocity, mixture enthalpy, and density.
WinSteam	WinSteam is used to calculate steam properties including specific volume, enthalpy, specific heat capacity, viscosity and thermal conductivity using the IAPWS-IF97 formulations.
XOQDOQ	This NRC software program is used for calculating long term atmospheric dispersion factors which are used in calculating dose to the public from routine releases of gaseous radioactive effluents.

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APPENDIX C

ASSOCIATED TECHNICAL REVIEW GUIDANCE

Appendix C
Matrix 1

Scope and Associated Technical Review Guidance

Materials and Chemical Engineering

Areas of Review	Other Guidance
Reactor Vessel Material Surveillance Program TR Section 2.1.1	FSAR Section 3.2.4 Section 4.2 Section 4.3 Section 4.4 Table 4-3
Pressure-Temperature Limits and Upper-Shelf Energy TR Section 2.1.2	FSAR Section 4.2 Section 4.3
Pressurized Thermal Shock TR Section 2.1.3	FSAR Section 4.3.3
Reactor Internal and Core Support Materials TR Section 2.1.4	FSAR Section 3.2.4 Table 3-1
Reactor Coolant Pressure Boundary Materials TR Section 2.1.5	FSAR Section 3.2 Table 3-1 Section 4.1 Section 4.2.1 Section 4.2.2 Section 4.3.2 Section 4.3.3 Section 4.3.4 Table 4-3 Table 4-4 Table 4-5 Table 4-6 Table 4-9
Leak-Before-Break TR Section 2.1.6	FSAR Section 4.2.6 Section 5.2.3 Section 5.2.4 Section 5.4.3. Section 7.1.1 Section 7.1.2 Section 14.2.2

Areas of Review	Other Guidance
Protective Coating Systems (Paints) - Organic Materials. TR Section 2.1.7	FSAR Section 1.3.2 Section 1.7.1 Section 1.11 Section 5.2.2 Table 1-3 Table 5-7 Table 5-8 Table 14-44 Table 14-45
Flow-Accelerated Corrosion TR Section 2.1.8	GL 89-08 EPRI NSAC-202L-R3
Steam Generator Tube Inservice Inspection TR Section 2.1.9	FSAR Section 4.2.2 Section 4.3.2 Section 4.3.4 Section 4.4.
Steam Generator Blowdown System TR Section 2.1.10	FSAR Section 4.2.6 Section 4.3.4 Section 5.4.4 Table 5-9
Chemical and Volume Control System TR Section 2.1.11	FSAR Section 5.2.1 Section 9.1 Section 9.2 Section 11 Section 14

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MATRIX 2

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Mechanical and Civil Engineering

Areas of Review	Other Guidance
Pipe Rupture Locations and Associated Dynamic Effects TR Section 2.2.1	FSAR Section 4.1 Section 4.2 Section 4.3 Section 5.1 Section 5.2 Section 5.4 Table 4-24
Pressure-Retaining Components and Component Supports, TR Section 2.2.2 TR Section 2.2.2.1 NSSS Piping, Components and Supports TR Section 2.2.2 2 BOP Piping Components and Supports TR Section 2.2.2.3 Reactor Vessel and Supports TR Section 2.2.2.4 Control Rod Drive Mechanism and Supports TR Section 2.2.2.5 Steam Generators and Supports TR Section 2.2.2.6 Reactor Coolant Pumps and Supports TR Section 2.2.2.7 Pressurizer and Supports	FSAR Section 1 Section 2 Section 3 Section 4 Section 5 Section 9 Section 10 Section 11 Table 4-19 Table 4-20 Table 4-21
NSSS Design Transients TR Section 2.2.2.8	FSAR Section 4.1 Section 4.2 Section 4.3 Section 4.6 Section 5.2.1 Table 4-8
Reactor Pressure Vessel Internals and Core Supports TR Section 2.2.3	FSAR Section 1 Section 3 Section 4.2.2
Safety-Related Valves and Pumps TR Section 2.2.4	FSAR Section 4 Section 5 Section 6 Section 7 Section 8 Section 9 Section 10

Areas of Review	**Other Guidance**
Seismic and Dynamic Qualification of Mechanical and Electrical Equipment TR Section 2.2.5	FSAR Section 1 Section 2 Section 3 Section 4.1 Section 4.2 Section 5 Section 6 Section 7 Section 8.2 Section 9.7 Section 9.8 Section 10.2 Section 10.5 Section 14.2.2
Incore Instrumentation Guide Tubes TR Section 2.2.6	FSAR Section 1 Section 3 Section 4.2 Section 5.1 Section 7 Section 9.3 Section 12.6 Section 14

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Electrical Engineering

Areas of Review	Other Guidance
Environmental Qualification of Electrical Equipment TR Section 2.3.1	FSAR Section 7 Section 8 Section 9.7
Offsite Power System TR Section 2.3.2	FSAR Section 8
AC Onsite Power System TR Section 2.3.3	FSAR Section 8
DC Onsite Power System TR Section 2.3.4	FSAR Section 8
Station Blackout TR Section 2.3.5	FSAR Section 14.1.2.9 Section 8.2

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MATRIX 4

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Instrumentation and Controls

Areas of Review	Other Guidance
Reactor Protection, Safety Features Actuation, and Control Systems TR Section 2.4.1	FSAR Section 1.4 Section 3.2 Section 4 Section 6 Section 7 Section 10.5 Section 14
Engineered Safeguards Actuation System (EFSAS), TR Section 2.4.2.1	FSAR Section 7
Emergency Feedwater Initiation and Control (EFIC) TR Section 2.4.2.2	FSAR Section 7.2.4 Section 7.5.2
Remote Shutdown System TR Section 2.4.3,	FSAR Section 7.4.6
Control Rod Drive Control System (CRDCS). TR Section 2.4.4.1,	FSAR Section 7 Section 9
Integrated Control System (ICS) TR Section 2.4.4.2	FSAR Section 4 Section 5 Section 7 Section 10

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Plant Systems

Areas of Review	Other Guidance
Flood Protection TR Section 2.5.1.1.1	FSAR Section 2.4.1 Section 2.4.2 Section 5.2.1 Section 6.2 Section 8.2.3 Section 9.5.2 Section 10.6.5 Section 12.6
Equipment and Floor Drains TR Section 2.5.1.1.2	FSAR Section 6.4 Section 8.2 Section 9.6 Section 11.2
Circulating Water System TR Section 2.5.1.1.3	FSAR Section 2.4 Section 9.5 Section 10.1 Section 11.2 Section 14.1
Internally Generated Missiles TR Section 2.5.1.2.1	FSAR Section 1.3 Section 4.2 Section 5.2.3 Section 5.2.4 Section 5.4 Section 6.1.2 Section 6.2.2 Section 8.2.2 Section 8.2.3 Section 9.5 Section 9.9 Section 10.5
Turbine Generator TR Section 2.5.1.2.2	FSAR Section 1 Section 5.4.3 Section 7 Section 10
Protection Against Postulated Piping Failures in Fluid Systems Outside Containment TR Section 2.5.1.3	FSAR Section 5.4.3 Section 5.4.4 Section 7.1.1 Section 7.1.2 Section 9.7.2

Areas of Review	**Other Guidance**
Fire Protection TR Section 2.5.1.4	FSAR Section 9.8
Reactor Coolant Drain Tank TR Section 2.5.2	FSAR Section 5.1.1 Section 11
Fission Product Control Systems and Structures TR Section 2.5.3.1	FSAR Section 11
Main Condenser Evacuation System TR Section 2.5.3.2	FSAR Section 10.2.7 Section 11.4.2
Turbine Gland Sealing System TR Section 2.5.3.3	FSAR Section 10.2.1
Spent Fuel Pool Cooling and Cleanup System TR Section 2.5.4.1	FSAR Section 9.3 Section 9.6 Table 9-6 Table 9-7
Nuclear Services (SW) and Decay Heat Seawater (DC) System TR Section 2.5.4.2	FSAR Section 1.3.2 Section 2.3 Section 2.4 Section 2.5.4 Section 4.2.2 Section 4.2.4 Section 5 Section 6 Section 7.1.1 Section 7.1.3 Section 9.1.2 Section 9.3 Section 9.5 Section 9.7.2 Section 10.5. Section 11.4.2
Reactor Auxiliary Cooling Water Systems TR Section 2.5.4.3	FSAR Section 1.3.2 Section 2.3 Section 2.4 Section 2.5.4 Section 4.2.5 Section 5 Section 6 Section 7.1.3 Section 9.4.2 Section 9.5 Section 10.5 Section 11.4.2
Ultimate Heat Sink TR Section 2.5.4.4	FSAR Section 1.3.2 Section 1.9.4 Section 2.4 Section 6 Section 9

Areas of Review	**Other Guidance**
Emergency Feedwater System TR Section 2.5.4.5	FSAR Section 5.1 Section 5.2.4 Section 5.4 Section 6 Section 7 Section 10 Section 14
Main Steam TR Section 2.5.5.1	FSAR Section 4 Section 5 Section 6.3.5 Section 7.2.3 Section 7.2.4 Section 10
Main Condenser TR Section 2.5.5.2	FSAR Section 4.2 Section 4.3 Section 7.2.3 Section 9.3 Section 9.5 Section 10
Steam Dump System TR Section 2.5.5.3	FSAR Section 10.2.1 Section 10.5.2
Condensate and Feedwater TR Section 2.5.5.4	FSAR Section 4 Section 10 Section 11.2 Section 14
Gaseous Waste Management Systems TR Section 2.5.6.1	FSAR Section 11.2.2
Liquid Waste Management Systems TR Section 2.5.6.2	FSAR Section 11.2.1
Solid Waste Management Systems TR Section 2.5.6.3	FSAR Section 11 Section 11.2.5 Table 11-4
Emergency Diesel Engine Fuel Oil Storage and Transfer System TR Section 2.5.7.1	FSAR Section 8.2.3
Light Load Handling System (Related to Refueling) TR Section 2.5.7.2	FSAR Section 9.6

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Containment Review Considerations

Areas of Review	Other Guidance
Primary Containment Functional Design TR Section 2.6.1	FSAR Section 4 Section 5 Section 6.2 Section 6.3 Section 7.3 Section 7.4 Section 11 Section 14
Subcompartment Analyses TR Section 2.6.2	FSAR Section 14.2.2
Mass and Energy Release Analysis for Postulated Loss of Coolant Section TR Section 2.6.3.1	FSAR Section 5.1 Section 5.2 Section 6 Section 14.
Mass and Energy Release Analysis for Secondary System Pipe Ruptures TR Section 2.6.3.2	FSAR Section 6 Section 14
Combustible Gas Control in Containment TR Section 2.6.4	FSAR Section 14.2.2
Containment Heat Removal TR Section 2.6.5	FSAR Section 5.5 Section 6.2 Section 6.3
Pressure Analysis for ECCS Performance Capability TR Section 2.6.6	FSAR Section 14.2.2

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Habitability, Filtration, and Ventilation

Areas of Review	Other Guidance
Control Room Habitability System TR Section 2.7.1	FSAR Section 1.4.11 Section 1.9 Section 2.2.3 Section 5.1 Section 7.4.5 Section 9.7.2 Section 9.7.3 Section 14.2.2
Engineered Safety Feature Atmosphere Cleanup TR Section 2.7.2	FSAR Section 5 Section 9.7.2 Section 11
Control Room Area Ventilation System TR Section 2.7.3.1	FSAR Section 9.7 Section 11
Spent Fuel Pool Area Ventilation System TR Section 2.7.4	FSAR Section 9.7 Section 11
Auxiliary and Radwaste Area and Turbine Areas. Ventilation Systems TR Section 2.7.5	FSAR Section 9.7 Section 11
Engineered Safety Feature Ventilation System TR Section 2.7.6	FSAR Section 1.6 Section 1.7 Section 2.3 Section 2.4 Section 2.5 Section 5 Section 8.2 Section 9.7 Section 11 Section 14
Reactor Building Ventilation Systems TR Section 2.7.7	FSAR Section 5 Section 6 Section 7 Section 8 Section 9 Section 11 Section 14

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SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Reactor Systems

Areas of Review	Other Guidance
Fuel System Design TR Section 2.8.1	FSAR Section 3 Section 4.1
Nuclear Design TR Section 2.8.2	FSAR Section 3.1.2 Section 3.2.2
Thermal and Hydraulic Design TR Section 2.8.3	FSAR Section 3.1.2 Section 3.2.3
Functional Design of Control Rod Drive System TR Section 2.8.4.1	FSAR Section 3.2 Section 5.1 Section 5.4 Section 7.1 Section 7.2 Section 9.1 Section 9.5 Section 9.7 Section 14.1 Section 14.2 Table 3-27
Overpressure Protection During Power Operation TR Section 2.8.4.2	FSAR Section 4 Section 7.1
Overpressure Protection During Low Temperature Operation TR Section 2.8.4.3	FSAR Section 4 Section 7.1
Residual Heat Removal System TR Section 2.8.4.4	FSAR Section 6.1 Section 5.1 Section 5.3 Section 5.4 Section 6.2 Section 6.3 Section 7.1 Section 7.4 Section 9.3.1 Section 9.4 Section 14.1 Section 14.2

Areas of Review	**Other Guidance**
Non LOCA Analyses Introduction TR Section 2.8.5.0	FSAR Section 14.1.2 Section 14.2.2
Decrease In Feedwater Temperature, Increase In Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve TR Section 2.8.5.1.1	FSAR Section 4 Section 7.1.1 Section 7.1.2 Section 7.1.3
Steam System Piping Failures Inside and Outside Containment TR Section 2.8.5.1.2	FSAR Section 5.4.4 Section 10.2.1 Section 10.5.1 Section 14.0.2 Section 14.2.2
Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Steam-Pressure Regulator Failure TR Section 2.8.5.2.1	FSAR Section 4 Section 7.1.2 Section 7.1.3 Section 7.2 Section 14.1.2
Loss of Non-emergency AC Power to the Station Auxiliaries TR Section 2.8.5.2.2	FSAR Section 4 Section 7.1.2 Section 7.1.3 Section 14.1.2
Loss of Normal Feedwater Flow TR Section 2.8.5.2.3	FSAR Section 4 Section 7.1.2 Section 7.1.3 Section 14.1.2 Section 14.2.2
Feedwater System Pipe Breaks Inside and Outside Containment TR Section 2.8.5.2.4	FSAR Section 4.3.3 Section 6.1.1 Section 6.1.2 Section 6.2 Section 6.3 Section 7.2 Section 10.5 Section 14.1.2 Section 14.2.2
Loss of Forced Reactor Coolant Flow TR Section 2.8.5.3.1	FSAR Section 4 Section 7.1.2 Section 7.1.3 Section 7.2.1 Section 14.1.2

Areas of Review	Other Guidance
Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break TR Section 2.8.5.3.2	FSAR Section 7.2 Section 14
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition, TR Section 2.8.5.4.1	FSAR Section 14.1.2
Uncontrolled Control Rod Assembly Withdrawal at Power TR Section 2.8.5.4.2	FSAR Section 14.1.2
Control Rod Misoperation, TR Section 2.8.5.4.3	FSAR Section 14.1.2
Startup of an Inactive Loop at an Incorrect Temperature TR Section 2.8.5.4.4	FSAR Section 14.1.2
Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant TR Section 2.8.5.4.5	FSAR Section 14.1.2
Spectrum of Rod Ejection Accidents TR Section 2.8.5.4.6	FSAR Section 14.2.2
Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory TR Section 2.8.5.5	FSAR Section 14.1.2
Decrease in Reactor Coolant Inventory TR Section 2.8.5.6	FSAR Section 14.2.2
Inadvertent Opening of Pressurizer Pressure Relief Valve TR Section 2.8.5.6.1	FSAR Section 14.2.2
Steam Generator Tube Rupture TR Section 2.8.5.6.2	FSAR Section 4.2.4 Section 7.1.2 Section 7.1.3 Section 14.2.2
Emergency Core Cooling System and Loss of Coolant Accidents TR Section 2.8.5.6.3	FSAR Section 14.2.2

Areas of Review	**Other Guidance**
Anticipated Transients Without Scram, TR Section 2.8.5.7	FSAR Section 7.2.2 Section 7.5
New Fuel Storage TR Section 2.8.6.1	FSAR Section 3.2.4 Section 5.4.3 Section 9.6.1 Section 9.6.2 Table 9-13 Table 9-14
Spent Fuel Storage TR Section 2.8.6.2	FSAR Section 2 Section 3.2.4 Section 5.1.1 Section 5.4.3 Section 9.3 Section 9.6 Table 9-13 Table 9-14
Loss of Decay Heat Removal at Midloop TR Section 2.8.7.1	FSAR Section 9.4.2

APPENDIX C
MATRIX 9

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Source Terms and Radiological Consequences Analyses

Areas of Review	Other Guidance
Source Terms for Radwaste Systems Analyses TR Section 2.9.1	FSAR Section 11 Section 14 Table 14-28 Table 14.29 Table 14-31 Table 14-40 Table 14-41 Table 14-52
Radiological Consequences Analyses TR Section 2.9.2	FSAR Section 11.1.1 Section 14.2.2 Table 14-30 Table 14-59

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MATRIX 10

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Health Physics

Areas of Review	**Other Guidance**
Occupational and Public Radiation Doses TR Section 2.10.1	FSAR Section 2.2 Section 2.3 Section 2.6 Section 11.1 Section 11.2 Section 11.3 Section 11.4 Section 11.5

APPENDIX C
MATRIX 11

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Human Performance

Areas of Review	Other Guidance
Human Factors TR Section 2.11.1	FSAR Section 12.2.3 Section 14.2.2

APPENDIX C
MATRIX 12

SCOPE AND ASSOCIATED TECHNICAL REVIEW GUIDANCE

Power Ascension and Testing Plan

Areas of Review	Other Guidance
Transient Performance TR Section 2.12.2	FSAR Section 4.1.1 Section 7

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #309, REVISION 0

APPENDIX D

CORE BORIC ACID DILUTION CONTROL FOR CR-3 AT EPU CONDITIONS

PROPRIETARY

AREVA NP Inc.

Engineering Information Record

Document No.: 51 - 9161089 - 000

Core Boric Acid Dilution Control for CR-3 at EPU Conditions

PROGRESS ENERGY FLORIDA, INC.

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APPENDIX E

MAJOR PLANT MODIFICATIONS

Appendix E Major Plant Modifications

I. BACKGROUND

Physical plant changes required to support EPUs are required for various reasons. Some are necessary to support the efficient electrical output of the unit to maximize the benefit of increases in rated thermal power (RTP). Others are necessary to support or compensate for analytical impacts of the RTP increase. The purpose of this attachment is to address the full range of modifications that relate to the CR-3 EPU to allow the reader a more complete understanding of the overall scope for the CR-3 EPU Project.

Progress Energy Florida, Inc., is a regulated public utility subject to the oversight of the Florida Public Service Commission. State laws provide unique funding mechanisms for certain activities including new nuclear generation and environmental improvements. Each has appropriately restrictive standards that must be met. Improvements in plant reliability do not meet such standards and those costs are recovered under base rates as normal capital improvements. Therefore, in order to keep the funding appropriately separate, the CR-3 EPU Project has limited EPU modification activities to those necessary to license or operate the plant at the higher power levels and does not include other improvements.

For CR-3, two major modifications often associated with EPUs were implemented independent of the EPU Project.

- Replacement of the three Generator Step-Up transformers: One failed in-service and the others were exhibiting unacceptable performance characteristics. They were replaced during Refuel 15 (15R, Fall 2007) and have been operating well since. When they were replaced, they were sized to accommodate EPU conditions (1200 mVA).

- Replacement Once-Through Steam Generators (OTSGs): These were installed during Refuel 16 (16R, Fall 2009). One of the key characteristics of the replacement generators is to replace Alloy 600 tubing with Alloy 690 tubing. The replacement generators were originally specified and designed to nearly EPU conditions (3010 MWt) and were re-evaluated to EPU conditions (3030 MWt, which is the core RTP plus Reactor Coolant Pump (RCP) power). The installation project for the replacement OTSGs was expanded to include replacement of Alloy 600 material in the Reactor Coolant System (RCS) by replacing a significant portion of the RCS Hot Leg. Furthermore, the attached main feedwater riser piping was replaced with new flow accelerated corrosion resistant materials).

1.0 POWER UPRATE MODIFICATIONS

The CR-3 EPU Project is being designed and implemented over several years, including three refueling outages (15R, 16R, and 17R) and intervening operating cycles:

- In 15R (2007), CR-3 installed Leading Edge Feedwater Flow Meters and other plant improvements in support of a Measurement Uncertainty Recapture uprate. This was licensed and implemented early in 2008.
- In 16R (2009-2011), CR-3 installed a number of physical plant changes to improve the unit's thermal efficiency. The significant changes are discussed briefly below.
- In 17R, the balance of the EPU related modifications will be installed and are discussed briefly below.

These modifications also improve plant margins at existing power levels. CR-3 has or intends to implement these modifications under 10 CFR 50.59. Thus, the installation of these modifications does not require prior NRC approval via this License Amendment Request (LAR). However, a number of the modifications are credited in the safety analysis revisions necessary to support EPU conditions. The most significant of these are detailed in Section 1.3 of this Appendix, entitled, “Significant Modifications Explicitly Credited in the EPU Safety Analysis.” Further, separate Enclosures are attached for three of the more complex modifications. Greater detail is provided for these modifications because they are credited in the EPU supporting analysis, and impact post-accident operator response requirements. Included in those Enclosures are aspects of the modifications that do require prior NRC approval as summarized in Attachment 1 of this LAR.

1.1 MODIFICATIONS INSTALLED IN 16R

The integrated impact of all major project modifications (EPU and OTSG replacement) installed in 16R was assessed with a 10 CFR 50.59 review and all the necessary design, licensing and analytical changes associated therewith. There was one change in the planned modification installation schedule. The low pressure turbine replacements were moved from 16R to 17R. The 16R modifications include the following:

1.1.1 ELECTRICAL GENERATOR AND RELATED UPGRADES

The Main Generator and Exciter were respectively upgraded and replaced by Siemens, Inc. The Main Generator components replaced included the stator core, stator winding, parallel rings, main leads, rotor, bushings, current transformers, and hydrogen coolers. The result was a generator with the following fundamental ratings: 1200 mVA at a power factor of 0.93 lagging, 22KV, 75 psig and 1800 rpm. To accommodate the generator upgrades, the Exciter was also replaced. In addition, the Turbine Lube Oil Cooler tube bundle was replaced to accommodate increased heat loads generated as a result of the pending power uprate; and the Hydrogen Cooler heat load removal capability was increased by replacing it with a new, higher capacity Cooler.

1.1.2 ISO-PHASE BUS DUCT COOLER REPLACEMENT

CR-3 installed a replacement for the existing Isolated Phase Bus Duct (IPBD) Cooler to provide increased cooling capability to support the EPU generator output rating of 1200 mVA. The new cooler includes two 100% capacity fans and motors with automatic swap-over capability for improved reliability. This modification also replaced the 5kV Non Segregated Bus from the startup transformer to the 4160V Unit Busses, 3A and 3B, to provide a full 2000 amp capacity connection to each bus. The balance of the bus was determined to retain adequate margin.

In addition, the current transformers on the low voltage bushings of the Generator Step-up Transformers were replaced due to inadequate thermal rating for operation at the higher EPU rating. Also replaced were the existing bus duct grounding straps so as to be consistent with the new 1200 mVA rating.

1.1.3 HEAT EXCHANGER UPGRADES

- CR-3 installed replacement heat exchangers, sized for EPU conditions, for the existing Condensate/Feedwater Heat Exchangers (CDHEs), CDHE-3A and CDHE-3B.

- CR-3 installed replacement Moisture Separator Reheaters (MSRs), sized for EPU conditions, for MSR-3A, MSR-3B, MSR-3C, and MSR-3D.

1.1.4 SECONDARY COOLING SYSTEM UPGRADES (SC PUMPS AND HEAT EXCHANGERS)

- CR-3 installed larger pump impeller/shaft assemblies and drive motors on each of the two Secondary Cooling pumps (SCPs), SCP-1A and SCP-1B, to achieve the necessary flow to support EPU conditions.
- CR-3 installed two new, higher capacity Secondary Cooling Heat Exchangers (SCHEs), SCHE-1A and SCHE-1B. The new SCHEs were designed to meet the increased heat loads with nominal 100 degree Fahrenheit SC outlet temperatures.

1.1.5 MSR SHELL DRAIN HEAT EXCHANGERS ADDITION

CR-3 installed MSR Shell Drain Heat Exchangers, CDHE-7A and CDHE-7B, to capture heat lost from the MSRs shell drains and transfer the heat from the drains into the Condensate System to improve plant efficiency. By adding these heat exchangers to the system, the condensate will cool the drains before they enter the condenser, and recover heat by discharging the heated condensate from the heat exchangers to the condensate line before it enters the deaerator.

1.1.6 TURBINE BYPASS VALVE REPLACEMENT

CR-3 replaced the 6-inch Turbine Bypass Valves (TBVs) with 12-inch valves, and replaced existing associated piping, as needed, to support the larger valves. The modification bounds the expected post-R17 EPU conditions with both the impact of EPU and the replacement OTSGs. The bypass capacity to the condenser is approximately 22.7% RTP at EPU full power turbine trip conditions.

1.1.7 ICS SCALING & FUNCTION CURVES EXITING 16R

The Integrated Control System (ICS) System was modified to support the increase in the electrical generation of the station resulting from plant modifications implemented during the 16R phase of the EPU. Several modules within the ICS which process the unit load demand (ULD) are referenced to a nominal 100% power electrical output value. These modules were rescaled to reflect post R16 electrical output. Total feedwater demand is referenced to the unit load demand. Modules within the FW subsystem were rescaled to reflect the revised post R16 unit load demand.

1.2 MODIFICATIONS BEING INSTALLED IN 17R

Each modification will be finalized and implemented in accordance with the CR-3 design change process to meet the applicable design and licensing basis requirements as outlined in this submittal.

1.2.1 TURBINE (HIGH PRESSURE AND LOW PRESSURE) REPLACEMENTS

The two low pressure and one high pressure turbines will be replaced in the 17R outage. The replacements are being provided and installed under contract by Siemens, Inc.

The existing Alstom welded-rotors low pressure turbines will be replaced with the Siemens advanced shrunk-on 18 m^2 disk rotor design in order to achieve improved efficiency, increased electrical generation, increased component life/reliability (due to reduced stress corrosion cracking potential), and reduced inspection/maintenance costs. The modification scope includes replacement of the inner casing assemblies, guide blade carrier, stationery blade rings, rotor, and blades. The modification also includes either upgrading or replacing the LP1 to LP2 jackshaft, glands, gland seal housing, bearing pedestal oil seals, oil seal rings, inlet flow guide rings, coupling guards, coupling bolt assemblies, exhaust flow guides, exhaust hood spray nozzles/piping, turbine bearings, lead-free (stainless steel) rupture disks, rupture diaphragm cover, and turning gear spacer. Modifications to the outer casing as well as existing mechanical systems and components (i.e., piping and hangers) are also necessary to support fit-up of the new components..

In order to ensure proper on-line monitoring of the new low pressure turbines, the following new systems will be installed with the turbine replacement: a low pressure turbine blade tip vibration monitoring system, a turbine generator torsional vibration monitoring system, and a grid fault monitoring system.

The existing Westinghouse 296 high pressure partial-arc turbine will be replaced with the Siemens monoblock 296FA full-arc design, sized to support the increased steam flow anticipated under EPU conditions. The modification scope also includes replacing/upgrading the coupling spacer, inner cylinder assembly, rotating and stationary blade path seals, inlet seal rings/nuts, gland cases, outer gland seal segments, bearing pedestal oil seal strips, turbine bearings, and blade carrier. Realignment of the outer cylinder, modifications to the steam inlet pipe flanges and the outer cylinder drain lines, and modifications to the Gland Seal System are also necessary to support fit-up of the new components as well as increased spillover capacity.

1.2.2 DEAERATOR RE-RATE AND BYPASS LINE INSTALLATION

Unlike most nuclear power plants CR-3 utilizes a condensate deaerator. It is a mechanical device used to remove dissolved gasses (primarily oxygen) from the Condensate System fluid; provide regenerative heating of the condensate to improve cycle efficiency and serve as a collecting point for returning high pressure drains to the feedwater cycle. CR-3 has determined the deaerator and the associated storage tank are not rated for the anticipated EPU operating conditions. Therefore, both the deaerator and the associated storage tank are being re-rated to the higher thermal-hydraulic conditions that they may experience under EPU conditions. Therefore, the deaerator is also being partially bypassed. With the bypass line installed, adequate oxygen control will be retained through chemical control. To assure ongoing effectiveness of the chemical control, enhanced monitoring capability will be included in this modification.

1.2.3 CONDENSATE AND FEEDWATER SYSTEM ENHANCEMENTS

One of the primary impacts of the EPU is to increase flow through the Main Steam, Condensate and Main Feedwater Systems. The increased flow rate is accomplished by increasing the capacity of all the main drivers (Condensate Pumps, Feedwater Pumps and Feedwater Booster Pumps).

A. CONDENSATE PUMP MOTOR AND CONTROL VALVES

The current variable speed condensate pumps will be replaced with higher capacity motors with direct drive pump impellers. The design flow capacity for each pump increases to 10,020 gpm at 745 ft from 8,000 gpm at 745 ft head for each of the existing pumps. Flow will be controlled via air operated control valves at the pump discharge as opposed to pump speed. This design was selected as it has a better operating history at other nuclear power plants. A condensate pump recirculation subsystem is being installed to provide sufficient flow to meet the modified pump minimum flow requirements at EPU conditions. Additionally, condensate reject flow design will be modified to facilitate startup operations at low condensate flow rates.

The condensate pump and motor replacement modification will also implement a new condensate demineralizer/polisher resin trap design to meet EPU conditions. Since 2007, at the current power level, the condensate demineralizer/polishers have experienced a wide variation in flow and differential pressure across the various combinations of demineralizer/polishers in service. The new “nominal” demineralizer and resin trap flows will not exceed the maximum design value; and the new combined vessel and trap differential pressures will not be excessive; however, little margin exists for the resin trap differential pressure alarm. As a result, CR-3 will implement a new resin trap design to equalize demineralizer flows and increase resin trap differential pressure alarm margin.

B. FEEDWATER BOOSTER PUMPS AND MAIN FEEDWATER PUMPS

Higher flow feedwater booster pumps and motors will replace the existing pumps, motors and support components. Each pump will be capable of delivering 14,084 gpm at 420 psig outlet pressure. The existing pump recirculation subsystem will be modified to provide sufficient flow to meet the replacement pump minimum flow requirements at EPU conditions.

The main feedwater pumps will be replaced with near-equivalent pumps. Each pump has a rated capability to produce 14,604 gpm at 2066 ft TDH. The replacement pumps will be operated at a higher speed to obtain the higher flow-rate. In order to provide adequate protection against damaging over-pressure conditions additional controls will be added to trip the MFW pumps before they exceed down-stream piping design limits.

These modifications will provide acceptable flow margins.

C. FEEDWATER HEATER REPLACEMENT

Replacement of the FWBPs will increase the maximum pressure for the fifth stage Feedwater Heaters (FWHE-2A and 2B) to 420 psig. The current tube-side design pressure is 400 psig and the maximum allowable pressure on the feedwater tube-side nozzles is 405 psig. As a result, CR-3 will replace these heat exchangers with similar heat exchangers to accommodate the higher discharge pressure from the replacement FWBPs. This modification will also include replacement of thermal relief valves (HVV-51 and -52).

FWHE-3A and -3B are also being replaced to support operation at the higher power level (and flows). The design of the replacement heaters will reduce tube velocities during EPU condition to below maximum HEI standard of 10 ft/sec, improve thermal efficiency, increase capacity and increase operational margin.

D. REPLACEMENT OF MOTOR OPERATED VALVES

Replacement of the FWBPs will increase the differential pressure across the various downstream motor operated valves (MOVs). Currently, the calculations for Feedwater Valves, FWV-14 and FWV-15, indicate a margin on the required motor torque of less than 10%. With any increase in differential pressure, this margin will fall further below the desired 10% margin. CR-3 will replace both of these MOVs in 17R with valves capable of performing under the new design conditions; and with margins meeting or exceeding expectations. Furthermore, the response time FWV-14 and FWV-15 will meet the maximum 20 second closure time required to support Main Steam Line Break accident analysis assumptions at EPU conditions.

1.2.4 ICS SCALING & FUNCTION CURVES AND OTHER VALUES EXITING 17R

The Integrated Control System (ICS) System will be modified to support the increase in the electrical generation of the station resulting from plant modifications implemented in support of the EPU. Several modules within the ICS which process the unit load demand (ULD) are referenced to a nominal 100% power electrical output value. These modules will be rescaled to reflect post R17 electrical output. Total feedwater demand is referenced to the unit load demand. Modules within the FW subsystem will be rescaled to reflect the revised post R17 unit load demand.

Additional ICS modification(s) will be made consistent with other EPU modification final design details, modeling and/or testing results that impact ICS functions. Listed below are examples of changes to runback targets or rates associated with the main feedwater pump, feedwater booster pump, and RCP trip runbacks (discussed further in Section 2.12.1, Approach to EPU Power Level and Test Plan) which will be implemented to assure the plant's automatic response to such transients is appropriate and acceptable.

- The loss of RCP runback target will be reduced from 75% to approximately 70%.
- The Main Feedwater Pump trip runback target will be lowered from 50% to approximately 40%.
- The Feedwater Booster Pump trip runback target will be lowered from 50% to approximately 40%.

Additional changes associated with EPU include:

- The Main Steam Header Pressure post-trip bias will be reduced from 125 psig to 95 psig.
- The asymmetric rod runback will be removed from the ICS.

1.2.5 MAKEUP TANK BYPASS

Core reactivity control to restore Axial Power Imbalance (API) and control rod insertion are limited by the rate of boration and RCS dilution through Makeup Tank (MUT-1). More responsive boration control will enhance the ability to promptly respond to changes and to maintain reactivity related operating limits, as required by the CR-3 Improved Technical Specifications (ITS) such as API (ITS 3.2.3) and SDM

(ITS 3.1.1) and Regulating Rod Insertion Limits (ITS 3.2.1). CR-3 is addressing this by installing a bypass line 'around' MUT-1.

1.2.6 STRUCTURAL SUPPORT IMPROVEMENTS

Each of the modifications noted in the preceding discussions will address structural support impacts as needed. However, as discussed in Section 2.2.2.2, BOP Piping Components and Supports, the change in EPU conditions coupled with the response of the systems to revised normal conditions (e.g., turbine stop valves stroking closed against higher flows) changes the loading on various pipe supports (hangers, snubbers, etc.) that would otherwise not be impacted by EPU modifications. Main steam system supports will be modified as identified in Section 2.2.2.2.

1.2.7 EMERGENCY FEEDWATER FLOW INCREASE IMPLEMENTATION

Emergency Feedwater (EFW) flow needs to be increased roughly in proportion to decay heat for EPU conditions. The current EFW flow requirement for Loss of Feedwater is 275 gpm per OTSG. At EPU conditions, this requirement increases to 330 gpm per OTSG. The two original Emergency Feedwater Pumps (EFP-1 and -2) previously operated with such flow rates controlled by the Emergency Feedwater Initiation & Control (EFIC) System. The credited pumps (EFP-2 and EFP-3) can and need to supply this higher flow, but are currently prevented from doing so by continuously in-service recirculation flow paths. In order to increase the flow sufficiently, the recirculation flow will be isolated when the EFW pumps are automatically actuated and flow reaches or on shutdown returns to appropriate setpoints. EFP-1 is not credited for any events requiring the higher flow and thus is not being modified.

1.2.8 RECONCILIATION OF REPLACEMENT STEAM GENERATOR

As noted earlier, the replacement steam generator's design and analytical basis was to a slightly lower than EPU power level. It was necessary to update the design and analytical documentation to reflect the higher power level in the various analyses. In additional necessary change to the 'downcomer' orifice plate setting will be made. The higher power level (and associated higher steam flow) will affect the differential pressure across that plate, and will require adjustment to maintain its settings to appropriate levels in the steam generator.

1.3 SIGNIFICANT MODIFICATIONS EXPLICITLY CREDITED IN THE EPU SAFETY ANALYSIS

The first three of these modifications warrant providing more complete conceptual design documents, which are provided as Enclosures 1, 2 and 3 to this Appendix. The enclosures include current and modified figures, drawings, pictures, and/or more detailed discussions. As noted earlier each modification will be finalized and implemented in accordance with the CR-3 design change process to meet the applicable design and licensing basis requirements as outlined in this submittal.

1.3.1 LOW PRESSURE INJECTION CROSS-TIE/HOT LEG INJECTION IMPLEMENTATION

In addition to its normal shutdown cooling function, the DH System also performs Low Pressure Injection (LPI) functions at CR-3 and supports active boron precipitation control. However, at EPU conditions, these two scenarios are not adequately supported by the current system design and performance. To ensure that the DH System performs all intended functions, CR-3 will modify the system by cross-connecting the two trains inside the Reactor Building. The cross-connect will support performance of these two functions.

a. This modification will improve the response to the Core Flood Tank (CFT) line break with 'loss of the opposite Engineered Safeguards (ES) bus' scenario described below.

   The event postulates a double-ended guillotine break at one of the two core flood nozzles. The High Pressure Injection (HPI) and LPI train associated with a single failed ES bus will not be available for coolant injection. The remaining LPI train, powered by the operating bus, must also be assumed to be unavailable for coolant injection because it may be aligned to the broken CFT line. As a consequence, only one train of HPI and one CFT can be assured. HPI restores the level, but only after some time, and during that time, the core is covered with a two-phase froth. In the event the ES bus is lost without a loss of offsite power a prompt (one minute) RCP trip is also critical to ensuring adequate core cooling. Otherwise the forced reactor coolant flow degrades injection flow and RCS conditions further. A failure to trip the RCPs within one minute could aggravate core level and progress the event to an inadequate core cooling scenario. Currently, CR-3 credits an NRC approved operator action to manually trip the RCPs within one minute. At the current analyzed pre-EPU power level, the 10 CFR 50.46 acceptance criteria are met. However, at EPU conditions additional plant modification is required.

   In addition to the manual trip the RCPs within one minute which will be automated by the ICCMS discussed below and in Enclosure 3, a modification will be installed which cross-connects the two LPI trains and also relocates and balances system resistance such that sufficient flow to both core flood lines from either available LPI pump is assured. The details that support the efficacy of this cross-tie are addressed in Enclosure 1.

b. This modification will also incorporate a tie-in from the proposed LPI cross-connect to the decay heat drop line to allow direct LPI-RCS Hot Leg injection. This flow path will mitigate boron precipitation by inducing reverse flow through the core to flush highly concentrated boron solution in the vessel back to the Emergency Core Cooling System sump where it will mix with lower concentrations.

c. Currently, boron precipitation mitigation is accomplished through two active means. One is reverse flow through an idled LPI pump back to the sump suction (referred to as dump-to-sump). The other is auxiliary pressurizer spray which, under certain limited conditions, credits pressurizer spray flow as forcing relatively less concentrated borated water back into the RCS. These methods are not single-failure proof (thus requiring an exemption granted to CR-3), and require relatively complex operator diagnostics and actions.

   The modified Hot Leg Injection system eliminates the noted shortcomings (it is simple and single-failure tolerant), and is functionally similar to the solution used by other PWRs.

   This flow path will normally be isolated and will be placed in service when appropriate conditions are observed from the Main Control Room. The operator actions required to align the system are to simply open both of the hot leg injection valves (one powered from each power train). The efficacy of this system is further discussed in Section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents. The man-machine interface improvements are discussed in Section 2.11, Human Performance.

1.3.2 ENHANCED SECONDARY COOLDOWN CAPABILITY IMPLEMENTATION

Mitigation of certain Small-Break (SB) LOCAs at EPU conditions requires more flow than a single HPI pump can produce. The flow requirements increase due to increased DH load.

Directly increasing HPI flow would require increased pump size which would be very difficult to accomplish due to physical and other constraints. However, it was recognized that increased primary-to-secondary heat transfer removes heat from the RCS, which leads to reduced pressure which, in turn, leads to increased HPI flow and an earlier discharge of the CFTs. Existing design features and/or operator actions work toward that goal; but are insufficient for EPU conditions. The modification selected to support EPU is a significant, but controlled, depressurization of the secondary system through the Atmospheric Dump Valves (ADVs).

In order to credit the ADVs in the design basis safety analysis, modifications are being made to make them and their controls safety-related. Additionally, sufficient safety-related motive force (bottled air) will be provided to assure achievement of the function until orderly transfer to other alternate air sources takes place. The capacity of the safety-related bottle air supply will be adequate to support the Station Blackout 4-hour coping duration (refer to Section 2.3.5, Station Blackout) and SBLOCA Mission Time. Finally, the increased ADV capacity (provided to support Appendix R cooldown requirements - refer to Section 2.5.1.4, Fire Protection) is credited post EPU for rapid secondary heat transfer. The design is detailed further in Enclosure 2.

This added feature, referred to as the Fast Cooldown System (FCS), will be automatically initiated, within 10 minutes, under specific conditions (i.e., Loss of Adequate Subcooling Margin coupled with inadequate HPI flow, due to, for example, single failure of a HPI pump, its power supply or flow path). The actuation system (ICCMS) is addressed in Enclosure 3. Once initiated, the ADVs will reduce secondary pressure and will be automatically controlled to a nominal 325 psig.

The circuitry will be powered from dedicated direct current (DC) power sources to avoid common mode failure between this and other DC circuits. The design requirement is for no single failure to adversely impact the ADV controls and the associated HPI train. This will include installation of separate analog controllers for each ADV located in the EFIC Room. A simplified circuit diagram is provided in Enclosure 2. The capacity of the DC power supply will be adequate to support a 4-hour coping duration. Operators will be provided with the ability to restore manual control with guidance incorporated into the CR-3 Emergency Operating Procedures.

The SBLOCA analysis (summarized in Section 2.8.5.6.3) credits this feature as being initiated within 10 minutes from the Loss of Adequate Subcooling Margin (LOSCM) with inadequate HPI flow. The increased primary-to-secondary heat transfer from the FCS removes heat from the RCS allowing it to more rapidly depressurize. This increases HPI flow and ultimately leads to timely CFT injection.

1.3.3 INADEQUATE CORE COOLING MITIGATION SYSTEM

The FCS discussed above is initiated when concurrent indications of a LOCA (Loss of Subcooling Margin (LOSM)) and Inadequate HPI flow after a reactor trip. A new analog automatic initiation system is being added to accomplish this function. Furthermore, the existing manual trip of the RCPs (discussed in 1.3.1 and the selection of a higher EFW target steam generator level are also being automated as an output of the Inadequate Core Cooling Mitigation System (ICCMS). This automation of manual actions reduces operator burden and enhances mitigation system reliability. The details of this new system are provided in Enclosure 3.

1.3.4 HIGH PRESSURE INJECTION SYSTEM RESISTANCE

The current high pressure injection system includes fixed resistance throttle valves to passively assure adequate flow to the RCS in the case of an HPI line break. That resistance reduces flow under other SBLOCA conditions and reduces the flow differential between one and two operating HPU pump conditions. The reduction in the flow and differential flow makes it more difficult to determine or recognize inadequate HPI flow conditions. Inadequate HPI flow is the primary parameter used by the Inadequate Core Cooling Mitigation System to actuate the new Fast Cooldown System (See 1.3.2 and 1.3.3 as well as Enclosures 2 and 3). The throttle valves are being opened to reduce the system resistance which facilitates recognition of inadequate flow and otherwise increases HPI flow to the core and improves SBLOCA mitigation performance except for the HPI line break scenarios. The effects of the Fast Cooldown System discussed in 1.3.2 and in Enclosure 2 improve HPI line-break performance sufficiently to not require the existing level of system resistance.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

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ENCLOSURE 1

LPI CROSS-TIE MODIFICATION

LPI CROSS-TIE MODIFICATION

1.0 PURPOSE

Crystal River 3 (CR-3) is pursuing an Extended Power Uprate (EPU) of approximately 15.5%. A limited number of modifications will be required to ensure the plant will be able to mitigate design basis accidents at the increased power level including addition of a Low Pressure Injection (LPI) cross-tie inside containment. LPI is credited with injection of borated water into the reactor vessel for emergency core cooling and reactivity control during certain accident conditions. While the current LPI configuration is adequate for the existing licensed power level, under EPU conditions LPI needs to be modified to improve its performance for the scenarios described below:

- The first is the Core Flood Line Break (CFLB) scenario, wherein a double-ended guillotine break of one of the two Core Flood lines, is postulated. Each Core Flood line is also the flow path of each LPI train into the reactor vessel. Failure of Core Flood line concurrent with limiting single failures reduces available cooling flow below acceptable levels.

- The second is post-LOCA boron precipitation scenario where boron concentrates in the core region due to boiling. Under certain temperature and concentration conditions, boron could precipitate out of solution potentially reducing core flow and thus heat transfer.

This document provides a summary of the proposed design for modifications to the DH/LPI trains at CR-3 necessitated by the planned EPU to a new core power level of 3014 MWt. The additional decay heat requires increased flow to mitigate a CFLB and to control boron precipitation in the core. This proposed design will be supported by safety, thermal-hydraulic, and structural analyses.

The implementation of the modification is expected to be completed in the 17R refueling outage. The modification will include:

- The installation of a cross-connect between the two DH/LPI trains inside the Reactor Building (RB). The cross-connect will perform two functions:

  - Alleviate the concern with a CFLB scenario upon failure of one Engineered Safeguards (ES) bus on the opposite train from the break. This modification will improve the Emergency Core Cooling System (ECCS) response to the scenarios described in Section 2.8.5.6.3.

  - Alleviate post-LOCA boron precipitation concerns. This modification will incorporate a tie-in from the proposed LPI cross-connect to the decay heat drop line to allow direct LPI-hot leg injection. This boron precipitation (BP) mitigation line will provide a flow path to induce reverse flow through the core to flush the highly concentrated boron solution in the vessel back to the ECCS sump where it will mix with lower boron concentrations and buffering solutions.

1.1 BACKGROUND – EXISTING PLANT CONFIGURATION

In addition to its normal shutdown cooling function, the DH System also must perform the following safety functions at CR-3:

- Low Pressure Injection System. During LOCA conditions, the LPI portion of the DH System injects borated water from the Borated Water Storage Tank (BWST) or the RB Sump into the reactor vessel for emergency cooling and reactivity control. The system consists of two separate and independent flow paths (trains) and provides redundancy in active components to ensure required system safety functions will be performed.

- Post-LOCA boron precipitation control. During some LOCA transients, conditions may result in boron becoming concentrated in the reactor core due to boiling in the reactor vessel. Boron could precipitate out of solution, under certain temperature and concentration conditions, potentially blocking core channels and reducing core heat transfer. Successful initiation of an active boron dilution means preventing precipitation from occurring in the core.

Figure 1 shows the current configuration and a postulated CFLB scenario. The event postulates a double-ended guillotine break at one of the two core flood nozzles. The scenario is unique in that it results in the loss of half of the total CFT inventory and flow loss from one LPI train.

During the initiation of LPI flow from the BWST to the core, DH pump flow currently must be maintained below 3500 gpm to maintain EDG operability. Flow control is performed by throttling DHV-110 and DHV-111 to a 3000 gpm setpoint to ensure that with instrument uncertainty and pump recirculation flow, a flow of 3500 gpm at the DH pump is not exceeded.

The current DH / LPI system configuration mitigates the following scenarios associated with this event.

1.1.1 CORE FLOOD LINE BREAK WITH LOSS OF OFF-SITE POWER (LOOP)

A Loss-of-Offsite Power (LOOP) coincident with the turbine trip following reactor trip is postulated for most LOCA scenarios. If a LOOP occurs, the two Emergency Diesel Generators (EDGs) are designed to start to provide power to the Engineered Safeguards (ES) busses. Classically, it is assumed that an EDG fails to start. The HPI and LPI train associated with the failed EDG will not be available for coolant injection. The remaining LPI train, powered by the operating EDG, must also be assumed to be unavailable for coolant injection because it may be aligned to the broken CFT line. As a consequence, only one train of HPI and one CFT can be assured to provide ECCS flow into the RCS.

In the current analysis with the nominal core power of 2609 MWt, HPI flow will match the core boil-off rate at approximately 10 minutes. The combination of residual RCS liquid, intact CFT flow and HPI flow keeps the minimum reactor vessel mixture level just above the top of the core. As decay heat decreases, the HPI flow allows the reactor vessel levels to recover slowly keeping the core continuously covered with a two-phase froth.

At the current analyzed power level, the 10CFR50.46 acceptance criteria are met for this scenario.

1.1.2 CFLB WITHOUT LOOP

Without a LOOP, the electrical power from the EDGs is not required. In this scenario, a failure of an ES bus results in the loss of a train of power and the ECCS equipment available for accident mitigation (one CFT and one HPI pump) is identical to that when an EDG is lost following LOOP. In this case, the RCPs will remain powered and the operators must trip them immediately following a loss of subcooling margin (LSCM). It has been determined that RCP trip within one minute following LSCM will ensure adequate core cooling for this scenario. A failure to trip the RCPs within one minute could exacerbate the RCS inventory liquid loss causing a lower core level and potentially evolving into an Inadequate Core Cooling (ICC) scenario.

Currently, CR-3 credits a NRC-approved operator action to manually trip the RCPs within one minute to meet 10CFR50.46 criteria.

1.1.3 BORON PRECIPITATION CONTROL

During LOCA transients, conditions may result in boron becoming concentrated in the reactor core due to boiling in the reactor vessel. Two flow paths exist to manage the concentration of boron in the core and to prevent precipitation:

- Recirculation Dump-To-Sump (DTS) of RCS fluid from the hot leg to the RB sump using the decay heat drop line. This method reduces the core boron concentration by inducing RCS flow from the top of the core to the sump. Boron precipitation in the reactor core is attenuated by aligning coolant drains from the hot leg through the DH Drop-Line and a series of valves back to the sump through a LPI line with an idle pump. This alignment supports recirculation of a portion of the ECCS injection through the core and out the hot leg instead of only out of the cold leg break.

- Hot leg injection via the Auxiliary Pressurizer Spray (APS). This alternate method injects LPI flow into the pressurizer through the auxiliary spray line. As pressurizer level increases, flow is forced through the hot leg backwards into the reactor vessel lowering boron concentration.

Both BP mitigation methods, however, can be compromised by a single failure of ES MCC-3AB. A single failure exemption was requested and granted by the NRC for this vulnerability.

In addition to single failure concerns, each of these methods has operational limitations and concerns as follows:

- When DTS is utilized, measures must be taken to limit fluid velocity at the RB emergency sump structures to preclude vortexing which could result in air ingestion to ES pumps taking suction from the sump.

- When APS is utilized, its effectiveness is limited by the available flow and decay heat at which a reverse core flow can be achieved. APS will only be effective at higher RCS pressures when the elapsed post-trip time is on the order of 3 to 6 days.

1.1.4 SINGLE FAILURE EVALUATION

Per 10CFR50 Attachment K Section I.D.1, the occurrence of the limiting single failure must be assumed. The limiting single failure for LOCAs is typically any failure that can render an ECCS flow path inoperable.

For the current LPI configuration, a failure of either DH pump (DHP-1A or DHP-1B) to start, or the failure of either motor operated outboard containment isolation valve (DHV-5 or DHV-6) to open on demand may be postulated as the limiting single failure, since either compromises an LPI flow path. Pump failure is not critical to LPI since there is an existing cross-tie outside the RB which could be used to route flow between trains if needed. If either DHV-5 or DHV-6 were to fail closed and the CFLB occurred in the opposite train, however, LPI would not be available to the core.

For the current boron precipitation control configuration, a failure of motor control center ES MCC-3AB would affect the motor operators on valves which permit the alignment for both APS and DTS methods.

At the EPU core power level, the higher decay heat will exacerbate the core coverage condition. As such, for both LOOP and no-LOOP EPU scenarios, the peak clad temperature limits will be challenged if modifications are not made. Therefore, a solution is needed to address these single failures. The DH pumps will not be modified or replaced as part of this modification. System modifications, therefore, must accommodate the existing pump capability and include sufficient margin for component degradation, EDG frequency variation, and measurement uncertainty.

1.1.5 SOLUTION FOR EPU ACCIDENT MITIGATION

A full-flow cross-tie is proposed inside the RB to ensure at least one train of LPI will remain available for a CFLB with a postulated single failure under EPU operating conditions. The cross-tie will be located between mechanical penetrations 342 and 343 as close as practical to the RB annulus wall. Locked-in-place throttle valves (DHV-500 and DHV-600) will be installed upstream of DHV-1 and DHV-2 to limit flow to approximately 3000 GPM per train with both trains operating during a normal cool-down (non-accident DH heat removal). This will ensure adequate normal shutdown cooling and will also provide sufficient line resistance to direct ample cooling to the core through the cross-tie and intact CF line.

To address the concerns with boron precipitation, a branch line will be installed off the LPI cross-tie inside the RB. The branch line will include a parallel MOV isolation path, two check valves in series for primary isolation, as well as a manual isolation valve. The boron precipitation MOVs will be completely redundant and will satisfy IEEE-279 criteria. The power, control, and indication will be fully seismically and environmentally qualified for the service location. The MOV electronics will be installed above the RB maximum postulated flood elevation, while the mechanical components will be qualified for RB spray and full submergence.

Pipe and fittings will satisfy the current CR-3 FSAR paragraph 1.3.2.12.

Valves will satisfy the requirements of CR-3 FSAR Table 6-2.

Additional, topic specific, discussions below elaborate on the design.

1.2 LPI CROSS TIE DESIGN

1.2.1 FLOW CONTROL

The selected configuration includes a passive, normally-open cross-connect line, with inline throttle valves and stop-check valves as shown in Figure 2. The addition of the cross-connect and inline throttle valves allows either DH/LPI pump to supply either CF line. This will ensure adequate resistance exists in the failed train to prevent excessive LPI flow discharge from the severed line, thereby ensuring adequate LPI flow to the reactor at relatively low RCS pressures.

DH flow control for normal (non-accident) cool-down and post-accident scenarios (LPI) will remain essentially unchanged in the new configuration. Currently, flow control valves DHV-110 and DHV-111 are pre-positioned to a required throttle point that represents an indicated accident flow rate of 3000 gpm. This position is indicated by a white permissive light on the MCB. The initial flow control setpoint will be increased from 3000 to 3200 gpm, the pre-positioned throttle point will be revised as needed and the white permissive light switches will be reset. The increase will ensure adequate system flow reaches the core during postulated accident scenarios when pump degradation, instrument inaccuracies, control loop dead-band, and pump recirculation flows are considered in the analyses.

During normal (non-accident) cool-downs, when the pumps are taking suction from the DH drop line, the flow in the first DH train started will be controlled by DHV-110 or DHV-111 to the operator-selected setpoint. If additional cooling is required, the second train may be placed into service. Although DHV-110 and DHV-111 when fully open, are set for 3200 gpm each, the maximum flow will be limited to approximately 3000 gpm/train (6000 gpm total) by the new locked-in-place throttle valves, DHV-500 and DHV-600 when both trains are in service.

In the event of a CFLB with the DH/LPI pumps taking suction from the BWST, the fixed resistance provided by the locked-in-place throttle valves (DHV-500 and DHV-600) will ensure adequate flow is provided to the core through the intact CF line assuming worst case pump degradation and worst case flows through the severed CF line. In order to achieve this result, the upper flow setpoint for DHV-110 and DHV-111 has been increased from 3000 gpm to 3200 gpm resulting in an increase in the upper flow limit for DHP-1A and DHP-1B of 3600 gpm (3500 gpm previously). In a post-accident scenario, once the BWST has reached its minimum level and prior to swapping the LPI suction path to the sump, DHV-110 and DHV-111 will be throttled to 2000 GPM to prevent damaging the LPI pumps due to low suction head (NPSHa < NPSHr). This is no different from the existing system operation. If the HPI pumps are secured after swapping LPI suction to the sump, then the flow setpoint will be reset to 2500 gpm.

1.2.2 SAFETY ANALYSIS FLOW RATES

LPI flow rates credited in the safety analyses at EPU conditions for various accident scenarios are provided in Table 3.1.2-1, LPI Flow Rates (LBLOCA), Table 3.1.2-2, SBLOCA CFT Line Break, and Table 3.1.2-3, SBLOCA CLPD Line Breaks.

1.2.3 VALVE REPLACEMENTS/ADDITIONS

The following valves will be replaced and/or installed in support of this design change:

- <u>Motor Operated Valves DHV-210 and DHV-211</u>

  These valves are currently used to throttle flow during quarterly DH/LPI pump testing and when the DH pumps are used to cool the spent fuel pool (SFP). This modification required these valves be replaced with stop-check valves (DHV-510 and DHV-610) which have significantly higher flow coefficients.

- <u>New stop-check valves DHV-510 and DHV-610</u>

  These valves prevent reverse flow if the new cross-tie line is in use, both containment isolation MOVs (DHV-5 and DHV-6) are open and one LPI pump fails to start.

- <u>BWST Recirculation Throttle Valve DHV-9 and Spent Fuel Pool Throttle Valve DHV-48</u>

  Existing manual gate valves, DHV-9 and DHV-48, are being replaced with manual globe valves to restore the flow control lost by removal of DHV-210 and DHV-211. DHV-9 will remain in its current location while DHV-48 will be relocated downstream slightly to avoid interferences. Both valves will retain the same equipment IDs, pressure/temperature rating, pipe class, etc. Their noun names (equipment descriptions) will change to reflect the valves' new functions, i.e., change from "isolation" to "throttle". Although both DHV-9 and DHV-48 are manually opened in current procedures, they are not used for flow control. This change will impact operating procedures.

- <u>Normally-open flow path through new cross-tie valves DHV-501 and DHV-601</u>

  The cross-tie will have an isolation valve on either side of the BP branch connection. These valves will be normally open under all modes of plant operation except for maintenance evolutions during plant shutdowns. This arrangement also allows for installation flexibility such that one side of the cross-tie can be installed up to its isolation valve without affecting operability of the other DH train. Once completed, the valve can be opened, restoring its respective train to service, and the other train worked similarly. While technically permissible, installation in this manner is subject to plant conditions and Operations approval.

- <u>Throttle valves (DHV-500 and DHV-600)</u>

  These will be manual globe valves, thus avoiding the addition of active components. These globe valves will be adjusted during system flow balancing to attain the desired flow resistance and then locked in place to prevent inadvertent repositioning either by personnel interaction, equipment/tool interaction, or by system vibration. The resistance is to be set such that the 10CFR 50.46 acceptance criteria will be met for all CFLB scenarios via delivery of some LPI flow to the core at lower RCS to containment differential pressures, while limiting the flow lost through a break downstream of the last check valves before the reactor vessel. This will ensure adequate flow through the intact line (and later through the BP line, once opened).

1.3 BORON PRECIPITATION

1.3.1 FLOW RATES

A boron precipitation mitigation connection has been incorporated into the LPI cross-tie design to increase the flow rate delivered to the core to accommodate the EPU power level and eliminate the single failure associated with boron precipitation control at higher mode operation (modes 1, 2, 3). This connection will provide a path for direct hot leg injection during the LPI injection phase to facilitate post-LOCA boron precipitation control.

The proposed boron precipitation connection will provide a new flow path from LPI to the RCS Hot Leg Decay Heat drop line as shown in Figure 3. This flow will enter the hot leg and initiate a reverse flow through the core when flow exceeds the core boil off rate. The reverse flow carries the boron to and out of the break. The BP line will be hydraulically balanced to provide a minimum of 400 gpm at 15 psid (relative to RB pressure) for effective dilution flows at the post-LOCA RCS pressures while taking suction from the sump (during the sump recirculation phase). As a result, the boron concentration in the core will be controlled during the accident.

1.3.2 VALVE ADDITIONS

The following valves will be installed in support of this design change:

- Motor Operated Isolation Valves (DHV-514 and DHV-614)

  Normally closed MOVs will be installed in the BP line to isolate flow during early portions of the accident, when diversion of LPI flow would have an adverse impact on its ability to cool the reactor and BP flow is not required. These valves could be opened by Control Room operators during the later stages of an accident when flow is required to mitigate boron precipitation in the core and RCS pressure has decreased sufficiently. Isolation of these MOVs will be controlled administratively. This configuration will provide additional degradation margin for the LPI pumps, as flow is not being diverted to the hot leg during times when maximum accident mitigation flow is required. The MOVs will be included in the CR-3 GL 89-10 program. All control, indication, and power circuitry will satisfy the current CR-3 licensing basis.

- New check valves (DHV-611 and DHV-612)

  These valves will be installed to prevent reverse flow into the LPI headers and provide a location for the ASME Section XI, Class 1 – Class 2 boundary. These valves will be RCS pressure isolation valves and will be addressed by ITS 3.4.13 and associated Bases.

  DHV-611 also will be credited as the inboard containment isolation valve (CIV) for the BP line. An inboard CIV is required to comply with the requirements for a Type I RB penetration as listed in Section 5.3.2 of the CR-3 FSAR.

  The licensing criterion provided in FSAR 1.4.53 - CONTAINMENT ISOLATION VALVES and repeated in Section 5.3.1, ISOLATION SYSTEM Design Bases, specifies that “Leakage through all fluid penetrations not serving accident consequence limiting systems is minimized by a double barrier so that no single credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building, and various types of isolation valves”. The only criteria pertinent to ES systems is found in the FSAR Section 5.3.1 which states “Fluid penetrations serving ES also meet the design basis and are subject to the Technical Specification operability and surveillance requirements for Containment Isolation Valves.”

  The flow path inside the RB contains inline valves between the RB penetrations and DHV-611. During normal power operation, the piping will be filled with water. There will be no significant differential pressure between the RB atmosphere and the water inside the pipe (if any, pressure would be slightly higher inside the piping due to valve seat leakage). For any scenario in which LPI would be in operation, fluid system pressure would exceed RB atmospheric pressure. Packing leakage through any inline valve, therefore, would be from the system to the building, providing a positive pressure differential and precluding any containment leakage outside the RB.

The inboard check valve CIVs, DHV-1/DHV-2, are excluded from Appendix J testing per FSAR Table 5-9 in the CR-3 FSAR. By extension, Appendix J testing will not be required for DHV-611.

- Isolation valve DHV-615

  This gate valve will be installed to allow isolation of the boron precipitation line for normal decay heat maintenance. Upstream, valves DHV-501 and DHV-601 can be used, thereby completely isolating all components between the LPI cross-tie and the DH drop line. This valve will be in the Class 1 pressure boundary.

Simplified flow paths for the new LPI cross-tie and boron precipitation lines are shown in Figures 2 and 3.

1.4 INSTRUMENTATION & ELECTRICAL

1.4.1 MOVS DHV-514 AND DHV-614

MOVs DHV-210 and DHV-211 (ES Train A and B, respectively) are located in their respective Auxiliary Building (AB) Decay Heat Pump Vaults. They are powered from ES MCC 3A3, Unit 4EG and ES MCC 3B3, Unit 4EG. The control design provides for throttling of the valve via a selector switch located on ES Sections of the MCB or from local pushbuttons located on the MCC. Valve position indications for closed, mid-stroke and open are provided at the MCB and the MCC.

As stated above, MOVs DHV-210 and DHV-211 will be replaced with stop-check valves. The power and control from the existing MOVs will be re-routed into the RB through two new electrical penetrations (converted mechanical penetrations 327 and 328 from the Triangle Room). New MOVs, DHV-514 and DHV-614 (ES Train A and B, respectively) will be installed in the RB at elevation 104' near azimuth 217.5° between the "B" D-Ring wall and the containment outside wall. They will be powered from the same MCC cubicles as existing MOVs, DHV-210 and DHV-211. The new control design will provide for Open/Close operation of the valves via selector switches located on the ES Sections of the MCB or from local pushbuttons located on the MCCs. Valve position indication for closed, mid-stroke and open will be provided at the MCB and MCCs. In the early stages of an accident, position indication is needed to confirm the valves are closed and in the later stages that at least one has opened. The MCB selector switches will be repositioned to support the revised MCB ES Section mimic.

1.4.2 DIFFERENTIAL PRESSURE BETWEEN LPI A AND B TRAINS

New RG 1.97 Category 2 Type D differential pressure strings will be installed between LPI A and B trains. Common pressure taps will be installed in the LPI headers for the redundant differential pressure transmitters (DH-64-DPT and DH-65-DPT). The transmitters and associated isolation valves will be installed outside the West D-Ring wall in the RB. All electrical components will be installed above the maximum postulated flood elevation. Differential pressure indication will be provided in the Control Room, utilizing Dixon dual channel indicators, to aid Operators in HPI termination determination during two LPI pump operation. The differential pressure indication will also be displayed on the SPDS Alpha pages.

Cabling from the new differential pressure transmitters will be routed to the Control Room through newly converted electrical penetrations 327 and 328.

1.4.3 LPI TRAIN PRESSURE INDICATION

New RG 1.97 Category 2 Type D pressure strings will be installed on each LPI train to provide LPI pressure relative to the RB pressure. Pressure sensing lines will utilize the new pressure taps installed for the differential pressure instruments. The transmitters (DH-66-PT and DH-67-PT) and associated isolation valves will be installed outside the west D-Ring wall in the RB. All electrical components will be installed above the maximum postulated flood elevation. Pressure indication will only be displayed on the SPDS Alpha pages. These instrument strings will be used to aid Operators in HPI termination determination during single LPI pump operation.

Cabling from the new pressure transmitters will be routed to the Control Room through new electrical penetrations 327 and 328.

1.4.4 DH RECIRCULATION ULTRASONIC FLOW INSTRUMENTATION

Ultrasonic flow transducers will be permanently installed on the 8" pipe leading to the BWST, downstream of DHV-9. The flow instrumentation will be used to reduce the flow measurement uncertainty associated with DH pumps testing (PT-360) to provide local flow indication for adjustment of globe valve DHV-9. As discussed above, the existing manual gate valve will be replaced by a manual globe valve since DHV-210 and DHV-211 will no longer be available for flow control during testing. Cabling from the ultrasonic flow meter will be routed to an appropriate location near DHV-9 at elevation 95'. A junction box will be mounted in the proximity of DHV-9 large enough to coil an additional cable. A portable, battery powered, hand held electronic ultrasonic flow meter will be connected to the coiled up ultrasonic transducer cables when flow measurement is needed for DH pump testing. The flow meter will be maintained by the calibration lab as a non-safety related, test instrument.

1.5 IMPACT ON OPERATIONS PROCEDURES

- Operational procedures will require revision to include steps and guidance as to when the BP line may be opened. The current time and core exit thermal couple temperature based criteria will be revised to address the post-LOCA EPU conditions.
- Flow adjustment during testing of the DH pump (PT-360) will be controlled locally using DHV-9 and local indication upon completion of this design change. This mode of operation will replace the existing control from the Control Room using DHV-210 and DHV-211.
- When using DH for Spent Fuel Pool cooling, manual throttling of DHV-48 will be required to balance the system in lieu of existing flow control using MOVs, DHV-210 and DHV-211, from the Control Room.
- Operational procedures will require revision to provide guidance on the termination of HPI flow based on the new LPI system configuration and available pressure indications.

1.6 SYSTEM WALKDOWNS

During the R15 outage in November 2007, an AREVA NP walkdown team was deployed to characterize the penetration area inside the Reactor Building. Penetrations 343 and 342 contain the DHV-5 and DHV-6 lines (respectively). This is an area at the 95' elevation near the foot of the South stairway, between

Azimuth 205° and 220°. The actual penetrations are in the overhead, about 11' (Elevation 106') above the 95' floor level.

The boron precipitation line routing was walked down and laser scanned by AREVA NP Metrology Services at the beginning of R16. The information from these two walkdowns was used to construct a detailed model of the RB including its structures, piping and conduit. This model was then used to route new LPI cross-tie and BP line piping. The constructability of the proposed changes was confirmed during a subsequent walkdown, in February 2010, and minor refinements were made to the original design to clear interferences and reduce head loss through the BP line.

The images in the in Figures 4 through 7 show the actual penetrations and their surroundings.

1.7 CODES AND STANDARDS

1.7.1 PIPING

The new DH System piping components will be designed to the same standards as the existing system. From the flow diagram 302-641 Sheet 3, components downstream of and including DHV-5 and DHV-6 and upstream of and including DHV-3 are classified Seismic Class 1 (S1) and Nuclear Class 1 (N1). Components upstream of DHV-5 and DHV-6 and downstream of DHV-41 are classified Seismic Class 1 (S1) and Nuclear Class 2 (N2). The Construction Code requirements applicable for both N1 and N2 piping design is USAS B31.1 "Code for Pressure Piping," 1967 Edition with Addenda B (1971), and Addenda C (1972) with installation, testing, fabrication, and inspection to USAS B31.7, 1969 as defined by SP-5206, where later Code Editions/Addenda are used (as permitted by ASME XI and 10CFR50.55a) reconciliations will be performed.

1.7.2 INSTRUMENTATION

New differential pressure instrumentation (DH-64-DPT and DH-65-DPT), installed between LPI A and B trains, will be safety related, provide RG 1.97 Category 2 Type D differential pressure indication, and will be environmentally and seismically qualified.

New LPI train pressure instrumentation (DH-66-PT and DH-67-PT) will be safety related, provide RG 1.97 Category 2 Type D LPI train pressure indication and will be environmentally and seismically qualified.

MOV's DHV-514 and DHV-614 position switches will be safety related. The switches and the indicating lights will be located on the Main Control Board, and will be seismically qualified.

1.7.3 VALVES

Valves for the DH system will satisfy CR-3 FSAR Table 6-2.

- Valve pressure boundary requirements are stated to be per ANSI B16.5. This standard no longer addresses the construction of valves. In lieu of using B16.5, ASME III (unstamped) valves are specified. ASME III requires conformance to ANSI B16.34 for valve pressure/temperature ratings and valve minimum thickness requirements. ANSI B16.34 is the modern replacement Code for ANSI B16.5. Use of ANSI B16.34 and ASME III are permitted by ASME XI (and 10CFR50.55a), as it is a later edition of the original Code.

- Valve material is required to be inspected (nondestructively examined) to the requirements of USAS B31.7. B31.7 was superseded by ASME III in 1971. ASME III has been specified for nondestructive examination. This is permitted by ASME XI and 10CFR50.55a.
- Valve seat leakage requirements are 2cc/hr/inch of port diameter. Valves that are in critical service, e.g., DHV-611 and DHV-612, have more stringent leakage requirements specified; others, which are not as critical are specified using this criteria.

1.8 POST-MODIFICATION FUNCTIONAL TESTING PLANS

Once installed, the system will receive a flow balance, similar to the HPI system. This is to ensure all valves are properly throttled. The flow balance will be performed using local high-accuracy flow measurement devices as well as high accuracy pressure gauges. Pressure gauges will be installed at predetermined locations in the piping system. Flow devices will be installed in each of the DH lines as well as the BP line.

Motor operated valves will stroke tested as well as electrically tested to verify amperage. A DP test will be performed if necessary, depending on the extent of valve disassembly for installation in the system.

The newly installed electrical penetrations will be tested in accordance with 10CFR50, Appendix J Type B tests to ensure adequate containment pressure boundary integrity is maintained following modification of penetrations 327 and 328.

Electrical components, such as flow transmitters and limit switches will be functional tested to verify proper operation and indication.

There are no unique test requirements to ensure the operability of the proposed modification.

1.9 OPERATING EXPERIENCE

Two other B&W plants have installed LPI cross-ties although their implementation differed somewhat. Both plants still credit 2 minutes for operator action time to trip RCPs on loss of subcooling margin (LSCM), but gained significant cooling for the CFLB scenario.

Referring to Figure 8, the first utilized cavitating venturis (CVs) to provide the passive resistance to limit flow through a broken CF line. Reverse flow through the cross-tie lines is prevented by check valves downstream of each venturi. The use of CVs in these locations (to add resistance to either broken DH line) required splitting the DH inlet lines to both sides of the reactor vessel as well as significant piping additions and modifications inside containment. The CR-3 design will provide similar results using a permanent resistance (DHV-500 and DHV-600) in each train, thereby eliminating the need for the significant piping addition. The CVs installed at this plant have been reported to experience cavitation during some normal operating conditions. The cavitation has resulted in excessive noise and vibration in the system.

Referring to Figure 9, the second station used flow restrictors in conjunction with their single cross-tie (or cross-over) line. The flow path is similar to the design proposed for CR-3, with the exception of not having check valves upstream of the cross-tie line. The flow restrictors are not of the fixed single orifice design; rather, they use stackable plates to adjust the flow resistance. Provision was also made for installing temporary flow instrumentation during outages, to facilitate testing. While the design provides flexibility, it does require opening the system should an adjustment be necessary.

1.10 SUMMARY

This document summarizes the proposed location, layout, and initial design requirements for modifications to the LPI trains at CR-3. The conceptual design for the modification is necessitated by the planned EPU to a core power level of 3014 MWt scheduled to be implemented in the 17R outage. The implementation of the modifications is also planned for 17R.

It is the intent of this design to use proven and standard components with which CR-3 is familiar, in order to reduce the impact on plant operations as a whole. Alternatives were also explored, and operating experience was taken into consideration.

The installation of a cross-connect between the two LPI trains will include the following features:

- The design installs a passive, redundant, normally open LPI cross-tie flow path inside containment. The cross-tie ensures either LPI train will be able to provide flow to either CFT line. Flow resistance for the new cross-tie will be provided by standard globe valves to assure adequate flow to the reactor vessel and limit flow lost through a CFLB. Globe valves were selected based upon their simplicity and ruggedness. Vents, drains, and test connections will be installed in the cross-tie line as required to support the installation and future maintenance.

- The proposed LPI cross-tie will include a new boron precipitation mitigation connection to the decay heat drop line. This design change reduces the reliance on the existing normally closed cross-tie line, which is vulnerable to single failure of ES MCC-3AB, for post-LOCA boron precipitation measures. This line will include normally closed, motor operated isolation valves which will be opened manually by the Control Room operators when needed for boron precipitation control.

- The Cross-Tie and BP line modification will also include the installation of two new redundant differential pressure transmitters (and associated indication strings) between the LPI A and B trains, and two (one per train) new non-redundant header pressure transmitters. All instrument strings will be safety related, RG 1.97 Category 2 Type D compliant.

    1. LPI Train A to train B differential pressure will be displayed in the Control Room to aid Operators in HPI termination determination during two LPI pumps operations. In addition, the differential pressure indication will also be displayed on the SPDS Alpha pages.
    2. Pressure indication for each LPI train will not be redundant. The pressure indication will only be displayed on the SPDS Alpha pages. This indication will be available to aid Operators in HPI termination determination during single LPI pump operation.

- New stop-check valves are required to replace the existing motor operated control valves DHV-210 and DHV-211. Installation of these stop-check valves will prevent reverse flow if the new or existing cross-tie lines are in use.

- The removal of DHV-210 and DHV-211 results in the need to replace the existing gate valves at DHV-9 and DHV-48 with globe valves. This will allow for more precise throttling characteristics during quarterly pump testing and when DH is supplying cooling flow to the SFP.

No impact is expected for normal power operations, since the DH (LPI) System is in standby. For normal shutdown cooling and refueling conditions, the modified configuration will not adversely reduce system capabilities, since the operation of either or both DH trains will ensure adequate flow for these evolutions. Isolation valves for the boron precipitation connection will be shut to allow increased flow through the normal decay heat flow path during shutdown cooling conditions.

LPI flow rates developed for various accident scenarios at EPU conditions are provided below:

TABLE 3.1.2-1 - LPI FLOW RATES (LBLOCA)

Pressure (psia)	Flow Rate (gpm)
14.7	2886
101.0	2886
117.0	2684
124.0	2581

TABLE 3.1.2-2 – LPI FLOW RATES (CF LINE BREAK)

ΔP (P_{RCS} - P_{RB}) (psid)	Total Flow (gpm)
0.0	1438
30.0	931
60.0	419
69.0	238
69.1	0

TABLE 3.1.2-3 – LPI FLOW RATES (SBLOCA LINE BREAKS)

Pressure (psia)[1]	Total Flow (gpm)
14.7	2886
84.0	2886
100.0	2687
125.0	2286
173.0	625
175.0	200

1. Pressure inside the reactor vessel at the core flood nozzle injection point

FIGURE 1 CURRENT DH/LPI CONFIGURATION (SIMPLIFIED)

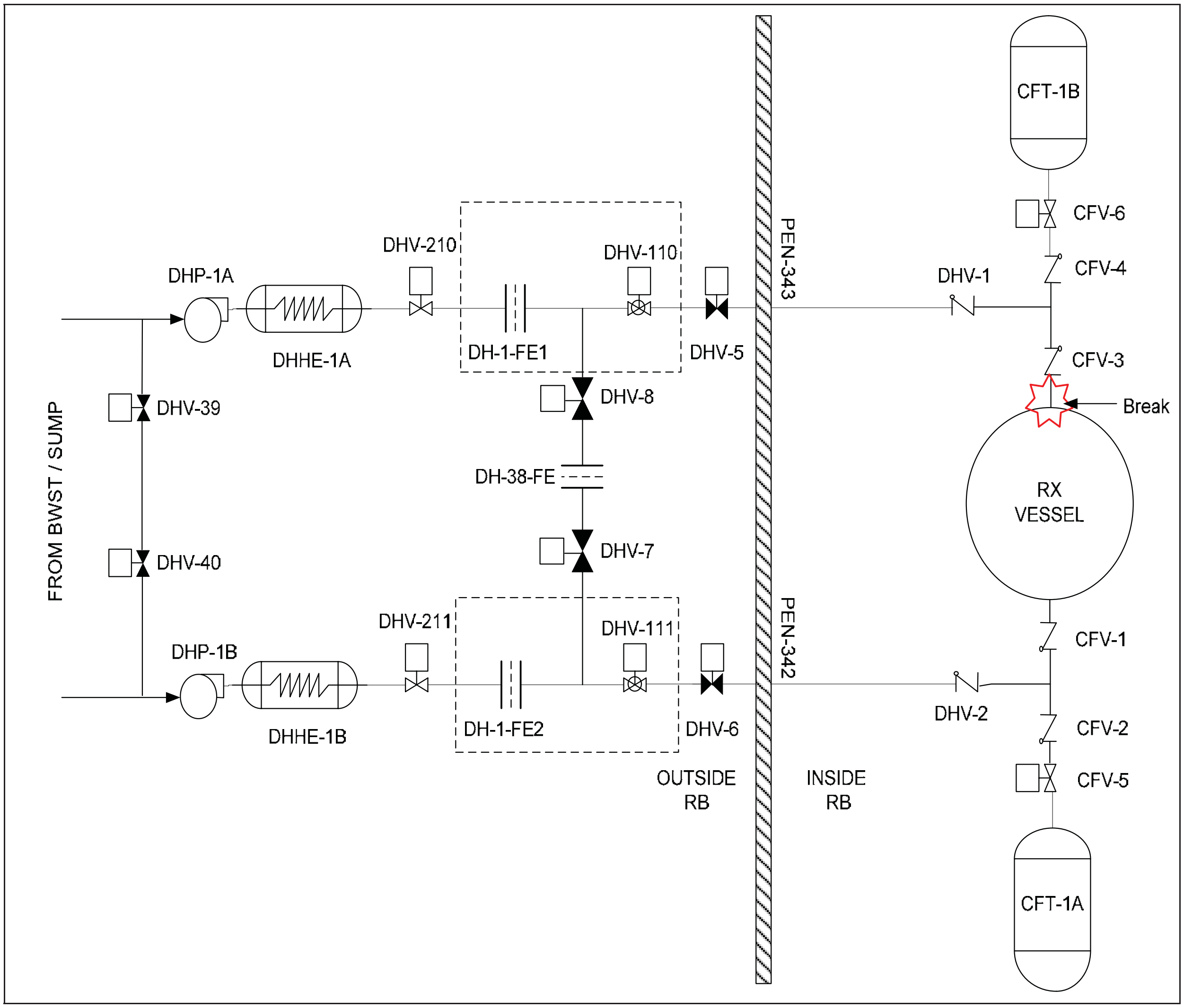


FIGURE 2 PROPOSED DECAY HEAT SYSTEM MODIFICATIONS

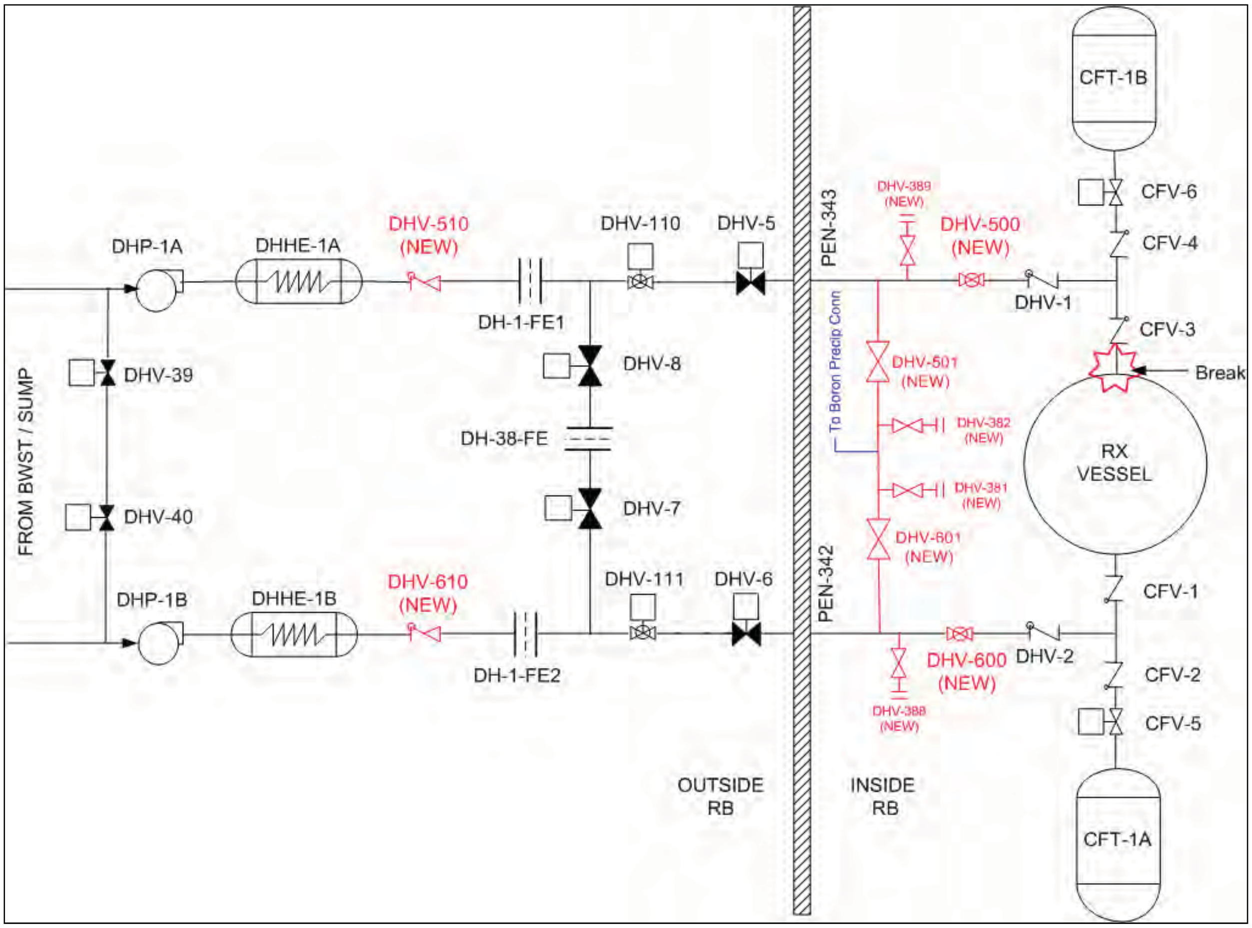


FIGURE 3 PROPOSED LPI CROSS-TIE WITH BORON PRECIPITATION CONNECTION

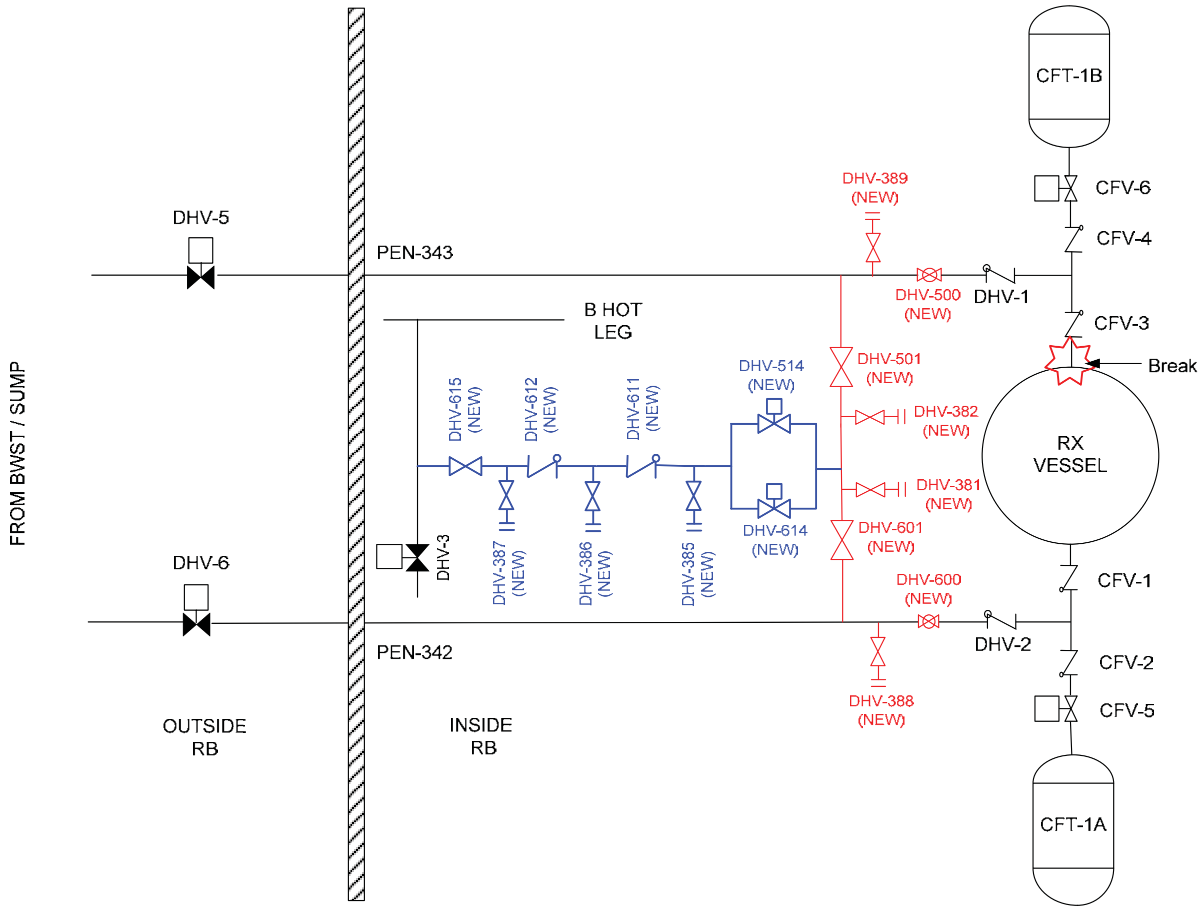


FIGURE 4 PENETRATION 342, LOOKING UP & WEST

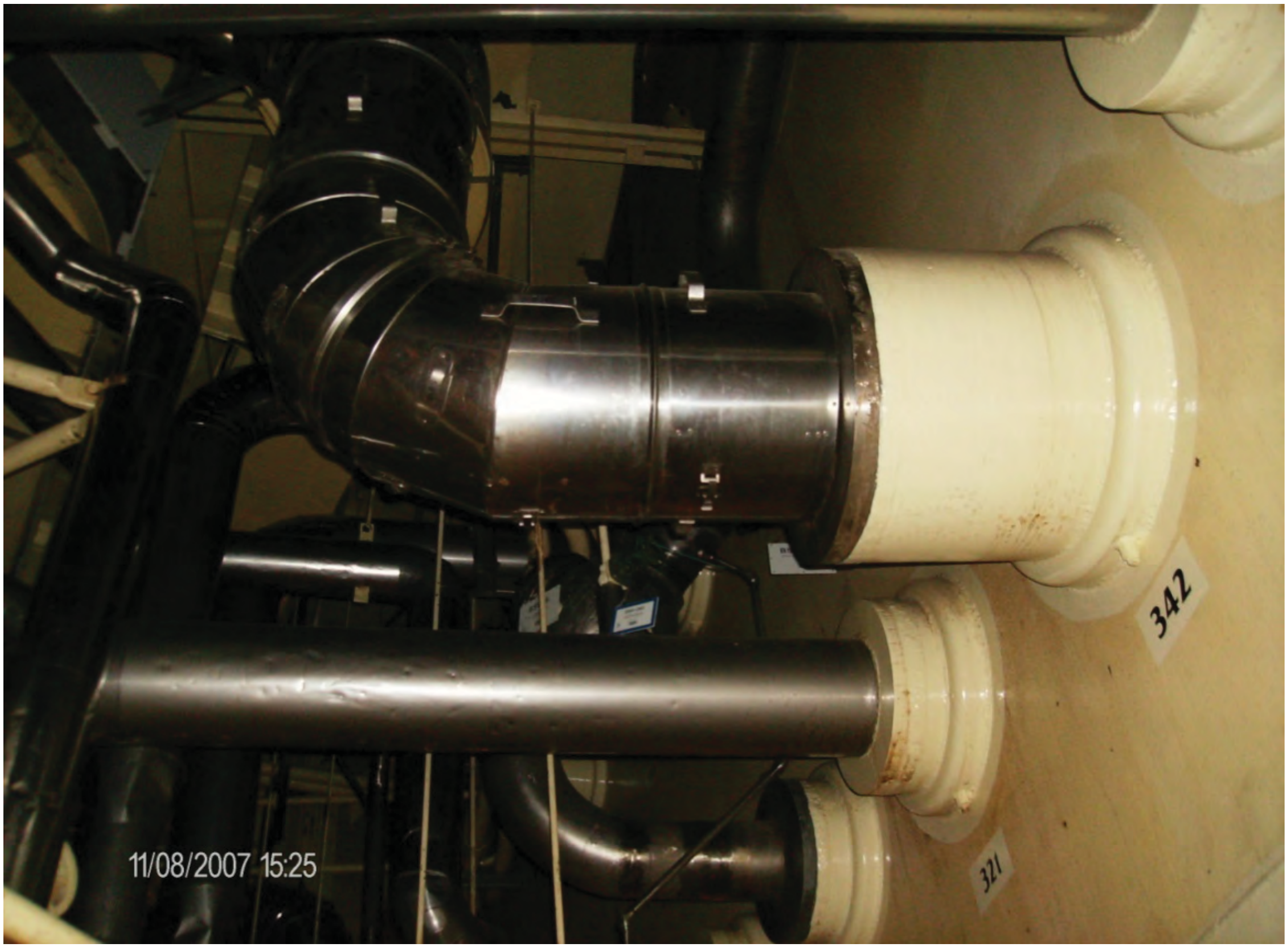


FIGURE 5 PENETRATION 342, LOOKING UP AT 1ST ELBOWS

FIGURE 6 PENETRATION 343, LOOKING UP

FIGURE 7 PENETRATION 343, LOOKING UP & SOUTH-EAST

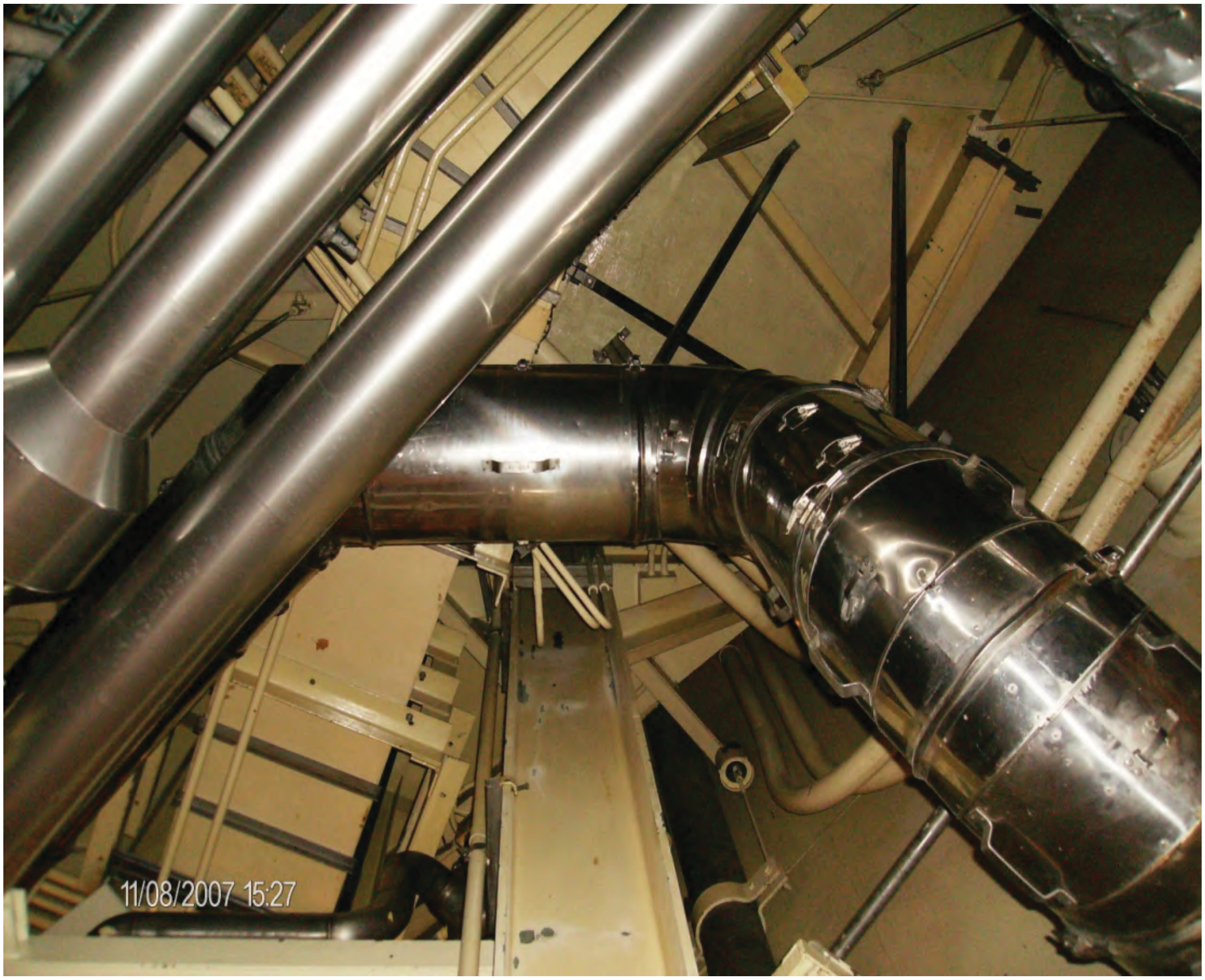


FIGURE 8 – O.E. FIRST PLANT

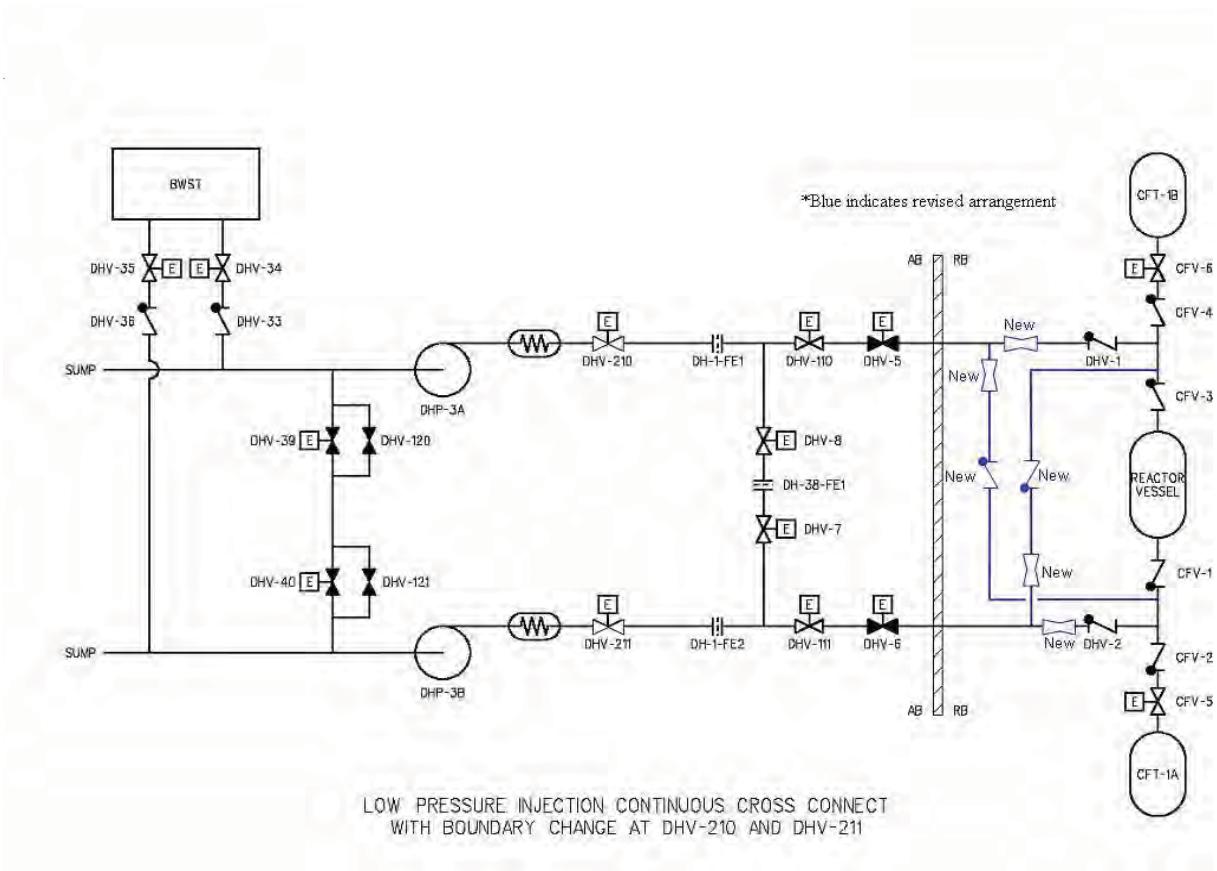
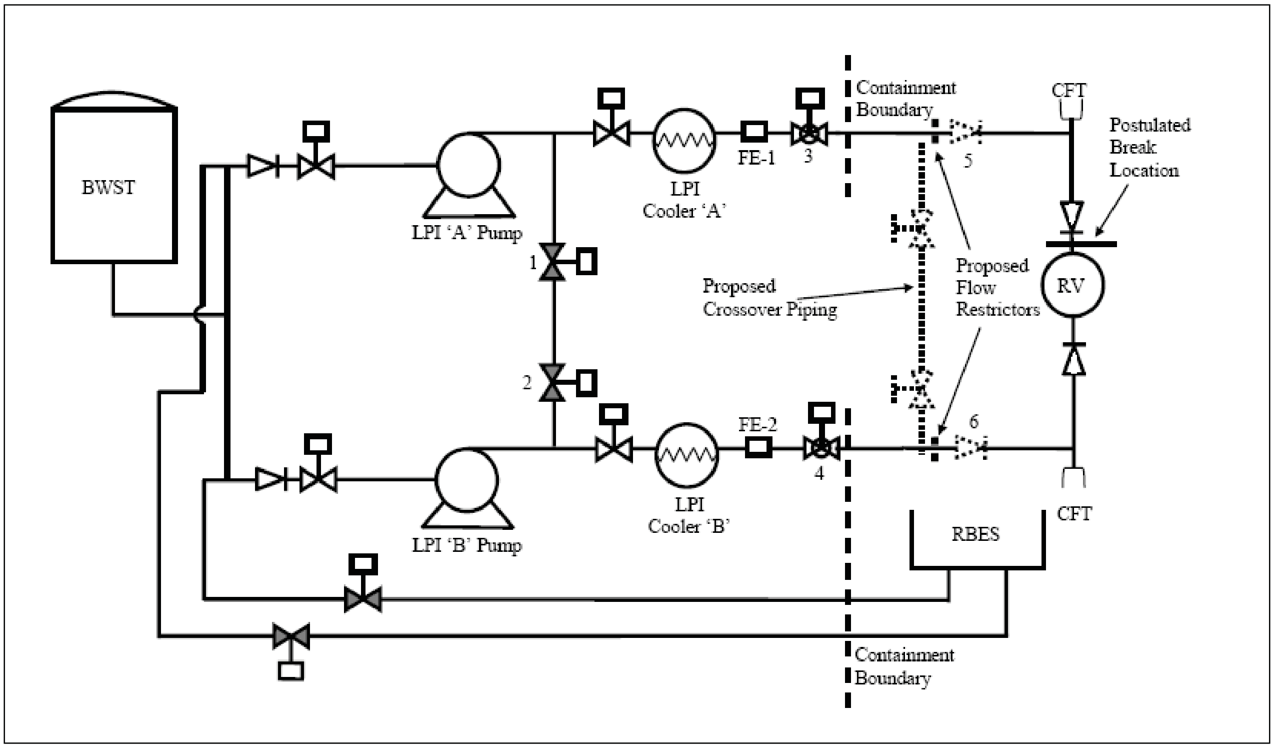


FIGURE 9 – O.E. SECOND PLANT



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CRYSTAL RIVER UNIT 3

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ENCLOSURE 2

ADV/FAST COOLDOWN SYSTEM MODIFICATION

ADV/FAST COOLDOWN SYSTEM MODIFICATION

1.0 PURPOSE

The CR-3 EPU conditions require larger atmospheric dump valves (ADVs) for a variety of transients. The ADVs will be installed as safety related to allow primary credit for this capability in mitigation of design basis accidents (i.e., Small Break LOCAs - SBLOCA). In addition to the larger ADVs, mitigation of SBLOCAs requires the addition of alternate controls referred to as the Fast Cooldown System.

At EPU power levels and without any modifications the SBLOCA Peak Clad Temperatures and peak local oxidation rates would be unacceptable due to insufficient High Pressure Injection Pump (HPI) flow. There is no practical means to sufficiently increase HPI flow. However, acceptable performance can be achieved coupling existing HPI capability with a more rapid main steam system depressurization. Depressurizing the secondary side increases primary-to-secondary heat transfer which reduces energy and thus pressure in the reactor coolant system (RCS). The RCS pressure decrease will result in additional safety injection from the HPI and earlier discharge from the Core Flood Tanks (CFTs). The combination of these effects leads to acceptable SBLOCA results (see Section 2.8.5.6.3, Emergency Core Cooling System and Loss-of-Coolant Accidents).

2.0 SCOPE OF DESIGN

2.1 FAST COOLDOWN SYSTEM (FCS)

The FCS utilizes both of the newly installed, safety-related ADVs for rapid depressurization of the main steam system to and in maintaining < 350 psig consistent with the updated safety analyses for EPU. The ADV venting will provide sufficient heat transfer through the OTSG to allow for the fast cooldown and corresponding depressurization of the RCS to provide sufficient flow from HPI and the CFTs, thereby ensuring adequate core cooling.

The overall scope of the design will:

- Install safety-related ADVs sufficiently sized to satisfy all functional requirements.
- Modify the piping and pipe supports as required to accommodate the new ADVs.
- Install safety-related instrumentation and controls to satisfy the performance requirements. When actuated by the Inadequate Core Cooling Mitigation System (ICCMS), the FCS will modulate the ADVs to control at the specified OTSG pressure (325 psig) to prevent excessive thermal cycling of the OTSG and associated components. The ICCMS is described in Enclosure 3 to this Appendix.
- Install pressure transmitters, pressure controllers, and DC backup power source for the pressure control such that ADV control can be maintained in the event of limiting concurrent single failures (including Loss of Offsite Power, Station Blackout or a DC battery or bus failure).
- Install safety-related backup air supply to support operation of the ADVs during a loss of normal instrument air over the four hour mission time.

2.2 OPERATOR ACTIONS FOR FAST COOLDOWN

During an SBO, upon recognition of a loss of subcooling margin current NRC approved operator actions are required to (1) select the SG level control setpoint to the ISCM level and establish maximum EFW flow to the SGs, and (2) commence a secondary plant depressurization using both ADVs. Under EPU, the FCS will be bypassed in order to depressurize the SGs below 350 psig. This is described in more detail in Sections 2.3.5 and 2.11. Operators provide a backup to the safety related actuation ICCMS described in Section 2.11 this Appendix.

For Appendix R fires in the Control Complex, the larger ADVs will involve a new operator action from inside the main control room and a new manual operator action from outside the main control room. These actions are described in more detail in Sections 2.4.3, 2.5.1.4, and 2.11.

2.3 SAFETY GRADE ADV CAPACITY

The ADVs are credited for operation during the following events:

- Appendix R (refer to Section 2.5.1.4, Fire Protection)
- SBO (refer to Section 2.3.5, Station Blackout)
- SBLOCA (refer to Section 2.8.5.6.3, Emergency Core Cooling System and Loss of Coolant Accidents)

The sizing for the ADVs utilized in the safety analysis for each event is summarized below:

TABLE 2-1: ADV FLOWRATE SUMMARY

	Units	Design	Appendix R	SBLOCA	SBO
Total ADV Flow	lbm/hr	1,240,000	1,239,920	1,178,000	602,592
Operable ADVs	Number of Valves	2	2	2	2
Flow Rate per valve	lbm/hr	620,000	619,960	589,000	301,296
Inlet Pressure	psia	962.7	962.7	962.7	962.7
Inlet Temperature	ºF	540	540	540	540

The ADVs will be sized for 620,000 lbm/hr at 962.7 psia and 540ºF, which bounds the required flow rate for the Appendix R event.

The ADVs are located within the ASME Section XI, ISI Code Class 2 boundary. Based on this, the valves fall under the In-service Inspection/Repair and Replacement Program (ISIRRPM). Design invokes ASME Section XI, 2001 Edition up to and including the 2003 Addenda. The requirements of ASME Section XI, IWA-4000 are also applicable.

The ADVs and their isolation valves have been specified to ASME III, 1986 Edition (or later up to the 2003 Addenda).

Locally, each ADV will include an electronic I/P and a pneumatic positioner to supply control air to the ADV in response to the 4-20 ma demand signal which is generated by either the EFIC control module or the new FCS pressure controller depending on which is in-service.

Based on Main Steam Safety Valve settings the maximum pressure during an over pressurization event is 1155 psig at 645^{o}F. The ADVs will be capable of operating or remaining closed during the over pressurization event.

2.4 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

The ADVs (MSV-25 and MSV-26) and new pressure transmitters are located inside the Intermediate Building, 119' elevation at the main steam line and are subject to harsh radiation environment conditions resulting from a Loss-of-Coolant Accident and Steam Generator Tube Rupture (SGTR). For EQ purposes, instrumentation will be qualified for LOCA and SGTR.

The ADVs and their electrical devices (I/P) are not credited (or relied upon) to operate or mitigate High Energy Line Break (HELB) inside the Intermediate Building. Since these valves and their support components are not required to function during or after a HELB scenario, they are not required to be environmentally qualified for HELB conditions per IEEE 323-1974.

Each ADV has two non-safety-related limit switches to indicate full closed position. The function of the limit switch is to provide closed indication for an ICS bias on TBV pressure control and also to provide an annunciator alarm when the valve is not closed. These functions will not change with new replacement ADV design. The limit switch functions are not required to mitigate SGTR or LOCA for the ADV operation and are not required for EQ qualification.

As noted below in Section 2.6 of this enclosure, the new pressure controller and new backup power source will be installed in the control complex mild environment. These components will have no EQ qualification requirements.

The ADVs and accessories shall be qualified per the requirements of IEEE 382, IEEE 323, and IEEE-344, as applicable.

2.5 ADV AIR SUPPLY

Each ADV will be provided with a safety-related backup air supply using a bottled air system. This safety related back-up air supply will be separated from the existing instrument air supply with check valves and will be the source of motive air for the ADV actuators upon loss of instrument air during a LOOP event. The safety-related air supply will be sized according to SBLOCA and SBO accident mitigation time requirements and valve air usage.

A conceptual schematic of safety-related Instrument Air Supply with high pressure cylinders of breathing quality air is in Figure 3.

2.6 INSTRUMENTATION & CONTROLS

Presently the ADV demand signal is produced from the EFIC Cabinet A and EFIC Cabinet B control modules with a setpoint of 1025 psig or from manual control from the ADV hand/auto stations in the control room. The manual control from the ADV hand/auto station is also routed through the EFIC Cabinet A and B control modules. This circuitry will be retained.

For the FCS, with a selected pressure control of 325 psig circuit modifications will require a new fast cooldown switch in the control room for each ADV. With the selection of the “Actuated” position of the FCS, the ADV demand signal will be switched from the EFIC control module and replaced with the FCS control setpoint demand signal. A selection of the “Auto” position will enable the FCS control actuation from the Inadequate Core Cooling Mitigation System, which will also replace the EFIC control module signal.

A postulated loss of an existing 1E Battery Bus during a LOOP would result in a loss of an HPI pump; inability of the Emergency Diesel Generator (EGDG) to supply power to an Engineered Safeguards Bus; and a loss of two EFIC Cabinets. Since the ADV demand signals for MSV-25 and MSV-26 are presently supplied from EFIC Cabinets A and B respectively, one of the ADVs would also fail closed due to demand signal failed low. To ensure that both ADVs are operable with any loss of a safety-related existing 1E DC battery bus coincident with a LOOP, the FCS/ADV modification will also provide separate stand alone DC power backup sources, and associated battery chargers, that can provide control circuit power for FCS valve position demand signals to the ADVs.

Existing pressure transmitters MS-106, 107, 110, 111-PT, which are Regulatory Guide 1.97, Category 1 instruments, will be used for Regulatory Guide 1.97 requirements to monitor and verify that the ADVs are performing their required system and accident mitigation function of cooldown of RCS using secondary system main steam pressure reduction.

The additional control and indication circuitry for the ADVs that are being installed will include the following 1E safety-related, seismically qualified and mounted components for each ADV:

1. New pressure controller with a 325 psig setpoint.

2. New OTSG pressure transmitter with calibration span from 0-1200 psig.

3. New transfer relays that will transfer control from the normal EFIC control module ADV demand signal based on 1025 psig setpoint, to the new FCS pressure controller on automatic control actuation from the ICCMS.

4. New fast cooldown selector switch with status light indication on the main control board.

5. New OTSG pressure indication for pressure signal being used as input to the pressure controller will be provided to RECALL and SPDS.

6. New independent stand alone DC backup power source, including batteries and battery chargers (chargers are not safety-related), for the new pressure transmitter, pressure controller, transfer relay, status light indication, and pressure indication. Each ADV has two redundant battery backed power supplies.

7. An I/P and positioner for converting the 4-20 ma demand signal from the new pressure control and the existing EFIC control modules to a linear direct acting air pressure supplied to the new ADV actuator.

8. A double pole, double throw (DPDT) closed end limit switch (non-safety related, non-1E).

A new stand alone DC power supplies will be independent and separate from the CR-3 1E batteries. Normal power will be supplied from a non-safety-related, non 1E battery charger. The new batteries will be 1E qualified, installed in the mild environment of the control complex and sized to supply power for a period in excess of the FCS mission time. Alarms will be included to alert operators to degrading conditions.

The ADV design includes an override circuit, actuated by operators from inside the control room that will prevent or terminate ADV spurious opening during a fire in the Control Complex. The circuit interrupts the control signal to the ADVs with two lockout relays. The relays are in series so tripping of either relay would isolate the control signal to the ADVs. Power for the lockout relays is supplied by the FCS batteries. The lockout relays may be reset from outside the main control room to allow control either from the remote shutdown panel or the control room. An alarm will alert the operators to the actuation of the lockout circuit.

2.7 STRUCTURAL EVALUATION OF ADV AND MAIN STEAM PIPING

For both ADV locations, the pipe runs from the connection at the 24 inch main steam header to the discharge of the ADVs will remain Seismic Class I, B31.1, Essential Piping. To accommodate the different valve configurations being installed, the piping will be rerouted. The new piping will be routed to minimize pipe stress in the existing 24" main steam header pipe and the new piping while avoiding surrounding SSCs. The piping and supports will be analyzed per plant procedures and guidelines, consistent with the piping design criteria in FSAR Section 5.4.4. Air tubing will be routed in accordance with plant procedures.

The new bottled back-up air supply system will support operation of the ADVs during a loss of normal instrument air. All of the above changes will be designed and installed as safety-related.

2.8 SINGLE FAILURE ANALYSIS

A single failure review of the ADV and FCS system has been performed. The FCS system is specifically designed to address the single failure of one train of HPI during SBLOCAs. Thus additional concurrent failures of the ADVs and FCS are not required to be assumed as part of the single failure design criteria within the context of SBLOCAs. For failures of the ADV and FCS, two trains of HPI are adequate to mitigate SBLOCAs without reliance on ADVs or FCS. For other events (i.e., SBO and Appendix R), single failure criteria is not applicable. The single failure review did not identify any single failure vulnerabilities.

FIGURE 1: CURRENT TYPICAL ADV CONTROL SCHEMATIC

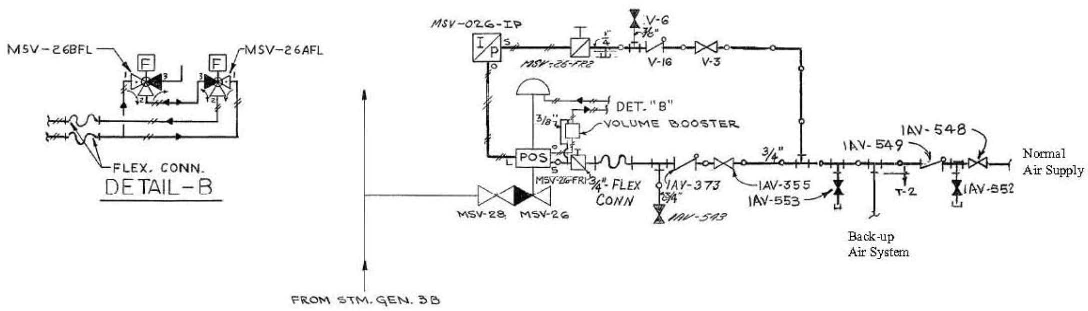


FIGURE 2: CURRENT ADV BACK UP AIR SUPPLY

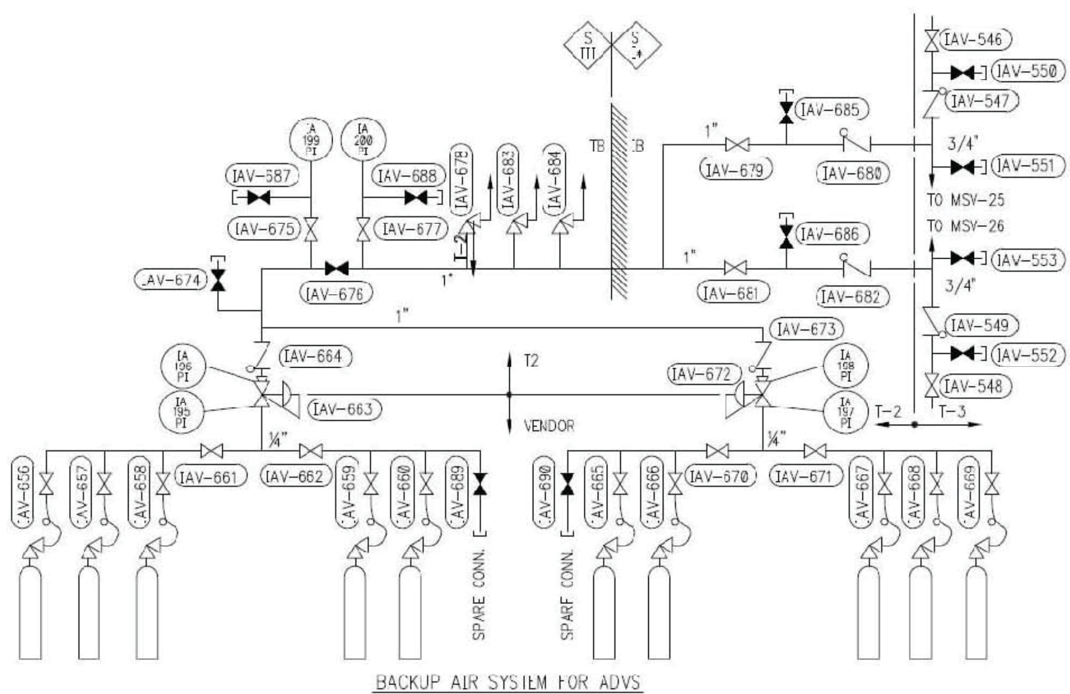


FIGURE 3: CONCEPTUAL SCHEMATIC OF SAFETY-RELATED INSTRUMENT AIR SUPPLY

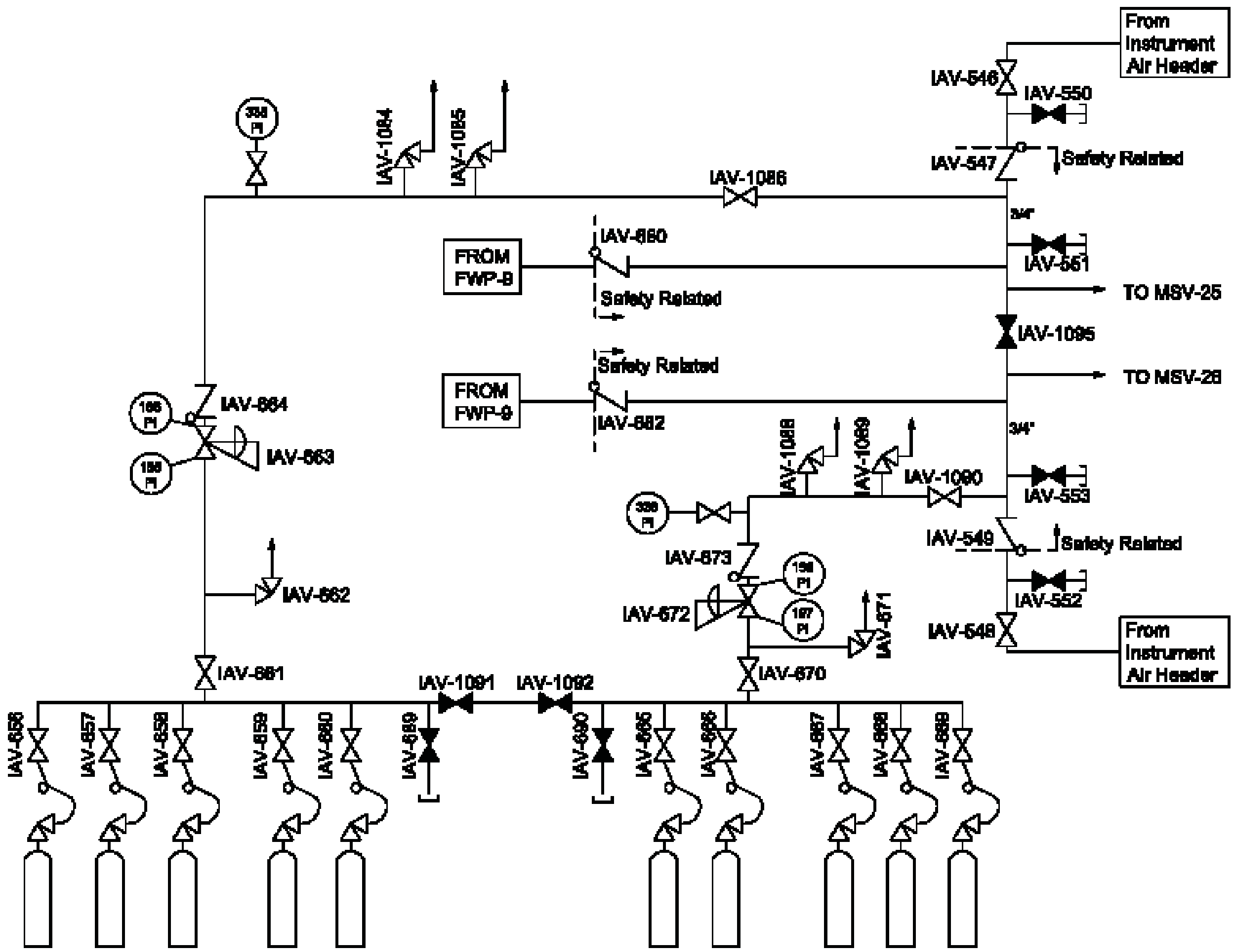


FIGURE 4: OTSB 3A – ATMOSPHERIC DUMP VALVE (CURRENT)

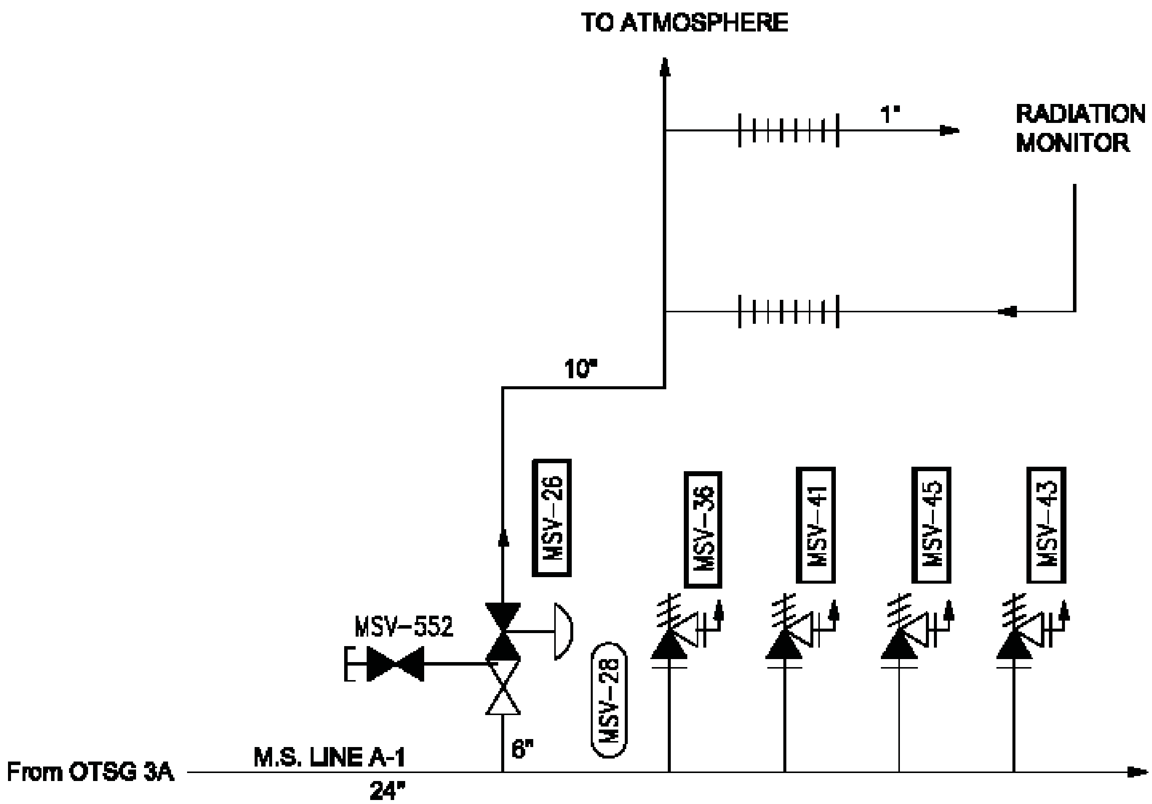


FIGURE 5: OTSB 3B – ATMOSPHERIC DUMP VALVE (CURRENT)

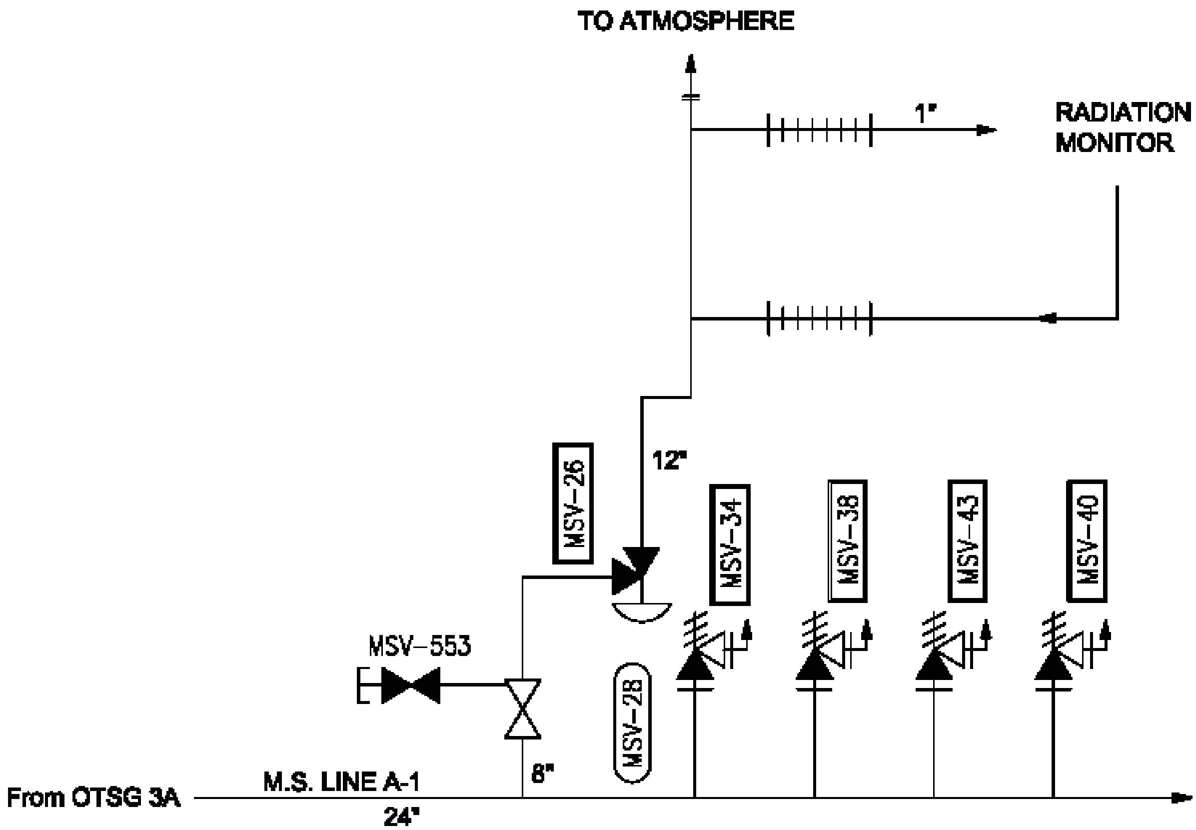


FIGURE 6: OTSG 3A – ATMOSPHERIC DUMP VALVE (NEW)

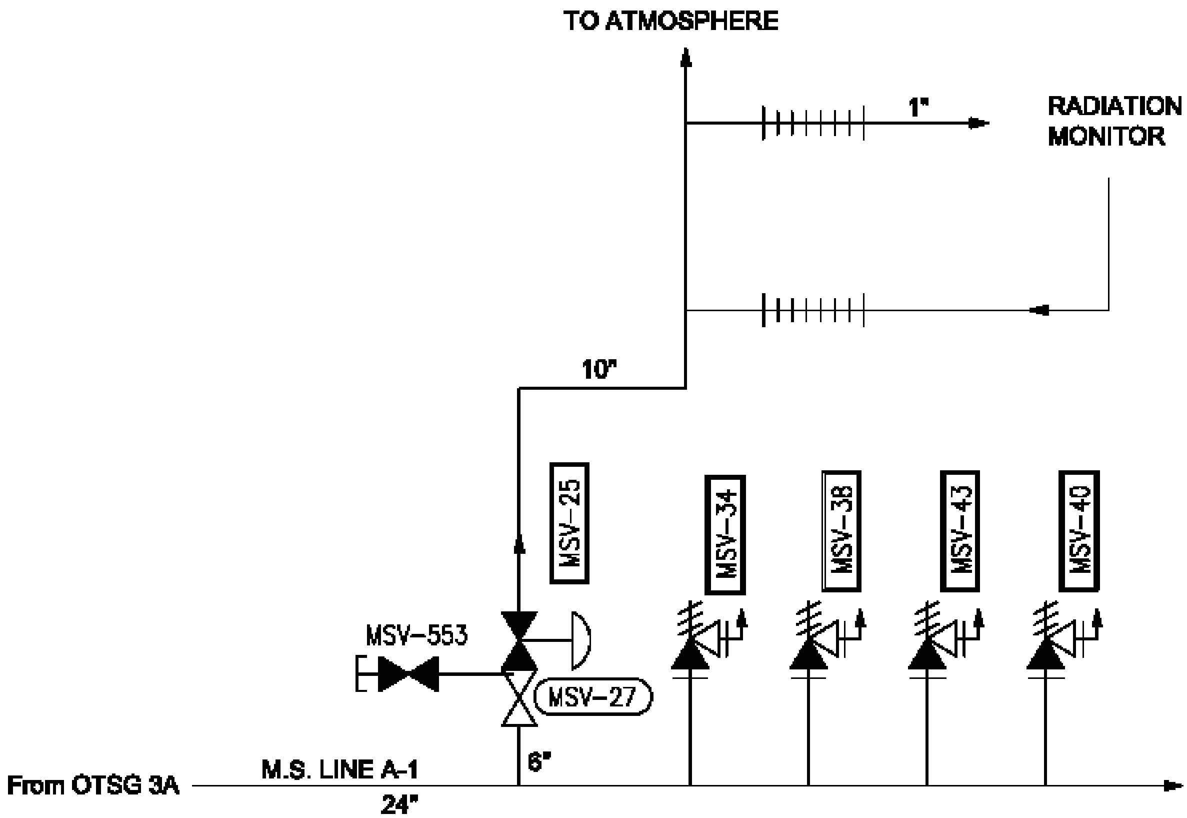
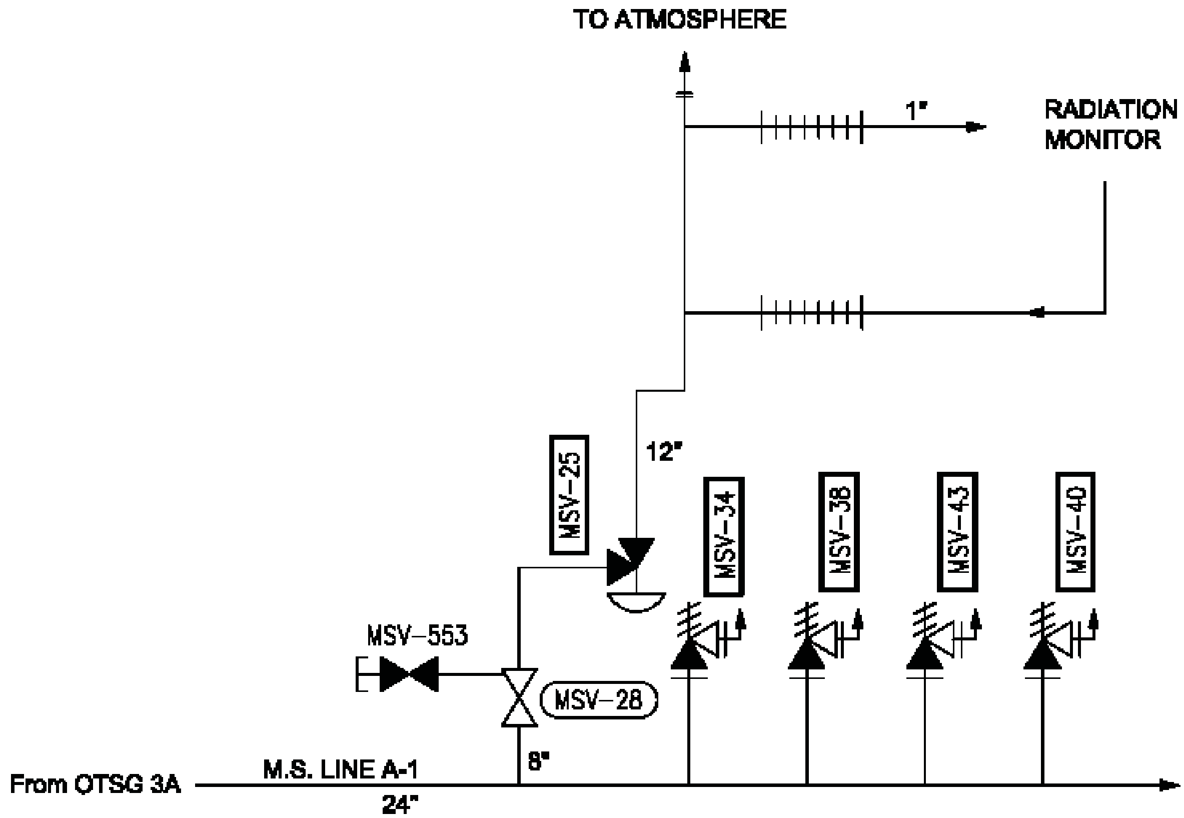


FIGURE 7: OTSG 3B – ATMOSPHERIC DUMP VALVE (NEW)



PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

LICENSE AMENDMENT REQUEST #309, REVISION 0

ENCLOSURE 3

ANALOG INADEQUATE CORE COOLING MITIGATION SYSTEM

ANALOG INADEQUATE CORE COOLING MITIGATION SYSTEM

Crystal River Unit 3 (CR-3) is applying for an extended power uprate. As part of the EPU, a new accident mitigation system is being installed – an Inadequate Core Cooling Mitigation System (ICCMS). The ICCMS has three mitigation functions and two post-accident monitoring indications.

- Function 1 – on a sustained loss of subcooling coincident with a reactor trip, the ICCMS shall trip the reactor coolant pumps to protect core inventory
- Function 2 – on a sustained loss of subcooling coincident with a reactor trip, the ICCMS shall raise the OTSG level setpoint to the Inadequate Subcooling Margin (ISCM) Setpoint
- Function 3 – on a sustained loss of subcooling coincident with a reactor trip and a sustained indication of inadequate HPI flow, the ICCMS shall send an actuation signal to the Fast Cooldown System. (Reference Enclosure 2 of Attachment E)
- Indication 1 – degrees of subcooling/superheat
- Indication 2 – HPI flow margin

The ICCMS proposed by CR-3 is an analog system. The functions required for the ICCMS are summing, subtraction, high select, square root, and curve generation. These are all simple functions for which analog circuitry has existed for decades. Much of the circuitry for the ICCMS will be similar to the CR-3 Reactor Protection System (RPS) and Non-Nuclear Instrumentation / Integrated Control System (NNI/ICS) designs. This circuitry has proved reliable and robust throughout the industry.

The ICCMS also includes a separate isolated non-safety digital online monitoring subsystem. The online monitoring system receives isolated outputs from the safety system and performs continuous checks on the ICCMS performance. The digital monitoring subsystem is fully isolated and separate from all safety functions and has no ability to affect the function of any of the safety-related portions of the ICCMS.

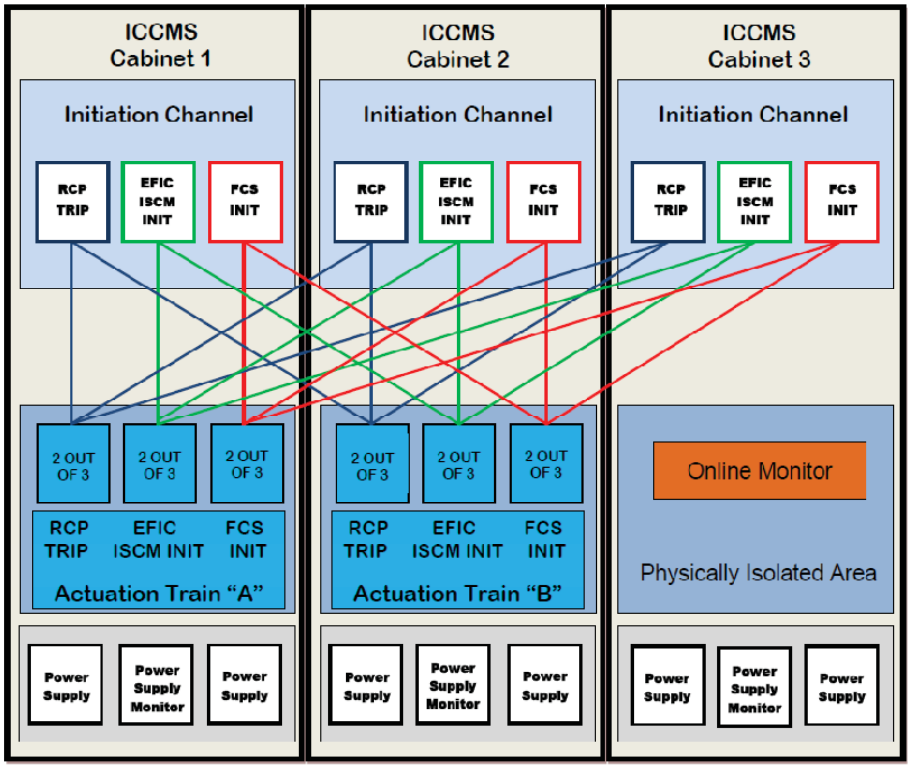
The proposed ICCMS is composed of three Initiation Channels and two Actuation Trains. The first two cabinets house an initiation channel and an actuation train. The third cabinet houses an initiation channel and includes an electrically and physically isolated compartment that houses a non-safety-related online monitor.

The channel initiation signals are sent to the train actuation logics through a simple on-off fiber optic link, similar to that currently used in the CR-3 Reactor Protection System. No other digital communications are sent or received (or exist) in the safety-related portion of the proposed ICCMS.

The system is classified as a Class 1E, safety-related protection system, meeting the requirements of IEEE-279 and IEEE-603. It will be qualified for mild environment per IEEE-323, for operation during and after a seismic event per IEEE-344, and tested to the EMI/RFI levels in RG 1.180 and EPRI TR-102323. Qualified isolation per IEEE 384 will be used between the non-safety-related Reactor Coolant Pump trip circuits and ICCMS.

Each of the three cabinets is independently powered from safety-related battery backed power sources. If 120 Vac power is lost to an initiation channel, all of the channel initiation outputs fail to the tripped state. If 120 Vac power is lost to an actuation train, both train actuation outputs fail to the untripped state.

Low level DC voltage signals shall be utilized for all analog variables and for all contact and status signals to provide greater EMI/RFI immunity.



The system shall be testable online without affecting plant operation or violating the single failure criterion. The initiation channels are tested by exercising the inputs to verify they create the correct outputs. Tripping an initiation channel puts the system in a 1-out-of-2 logic state, during which the system is still able to perform its safety function concurrent with a single failure. The train actuations are tested using a test signal that verifies continuity through the combination logic, but does not actuate the final system actuation relay. If a valid trip signal occurs during train actuation testing, the valid signal still actuates the final system actuation relay.

Reactor Trip Processing Circuitry

Each channel monitors a set of independent auxiliary contacts for each of the six safety-related Control Rod Drive breakers to confirm that a reactor trip has occurred.

HPI Flow Processing Circuitry

Each channel receives four flow signals from dedicated safety-related flow transmitters (one from each of the four HPI lines). The four signals are added together to produce a total HPI flow signal for each channel.

RCS Pressure Processing Circuitry

Each channel monitors two independent pressure sensors – a low range sensor (0-600 psig) and a wide range sensor (0-2500 psig). For RCS pressures of 500 psig and below, the low range signal will be used; above 500 psig, the wide range signal will be used.

The selected RCS pressure is an input to function generators used to derive the degrees of subcooling, degrees of superheat, and required HPI flow.

Core Exit Thermocouple (CET) Processing Circuitry

Each channel monitors eight Type K core exit thermocouples. Since core exit thermocouples have a history of occasional failure, and since they often can not be repaired until the next refueling outage, provisions for bypassing a failed CET are included. The circuitry then selects the highest of the remaining CET signals for use in the subcooling margin and superheat calculations. Automatic actuations always use CET input.

Hotleg Resistance Temperature Detector (RTD) Processing Circuitry

Channels 1 and 2 each monitor one hot leg RTD. The RTD input can be selected for the degrees of subcooling indication. On a reactor trip, the CETs are automatically selected for indication of degrees of subcooling and superheat.

Subcooling / Superheat / HPI Flow Margin Processing

Subcooling margin is the difference between saturation temperature for a given pressure as compared to the actual temperature corrected for instrument uncertainty. Superheat is defined as the difference between actual temperature and the saturation temperature for a given pressure. Subcooling margin, superheat and inadequate HPI flow, defined below, are used in initiating system actuation.

Function generators will provide reference curves for subcooling margin, superheat and HPI flow margin.

Subcooling Margin and HPI Flow Margin curves will be considered as generated Limited Safety System Setting (LSSS) setpoints and will be used in the channel actuation circuitry. The inputs to these curves (RCS temperature, RCS pressure, and HPI flow) will have their instrument uncertainties treated as LSSS instrument uncertainties and will have the footnotes of TSTF-493, Revision 4, “Clarify Application of Setpoint Methodology for LSSS Functions,” Option A, applied in the ICCMS instrumentation Improved Technical Specification (3.3.19). A Sample setpoint calculation for these parameters are provided in Attachment 8.

RG 1.97 Post Accident Monitoring Displays

Safety-related Regulatory Guide 1.97 Category 1 indicators shall be provided for the degrees of subcooling / superheat and HPI flow margin. Channel 1 and 2 shall display degrees of subcooling / superheat on two digital displays mounted on the main control board. Normally the displays indicates the degrees of subcooling / superheat based on the highest CET however, the operator can switch to the indication based on hot leg RTD temperature. Indicator lights shall indicate whether the display is showing degrees of subcooling or superheat.

HPI flow margin is calculated as the difference between actual HPI flow and required HPI flow. Channel 1 and Channel 2 HPI flow margin are displayed on two additional main control board digital displays. When HPI flow is not required both indicators will read zero.

The digital displays utilized will undergo the required commercial grade dedication procedure. Similar digital displays are currently in use at CR-3.

Train Actuation Circuitry

There are two actuation trains. Each train monitors all outputs of all three initiation channels and inititates a train actuation on two out of three initiation channels actuated. Each train independently actuates all required mitigation functions.

Manual Actuation

The operator has the ability to manually perform all ICCMS mitigation functions from the main control board independent of the ICCMS. The four Reactor Coolant Pumps may be tripped by operating the main control board switches, the OTSG level setpoint may be selected by depressing the main control board push buttons, and the Fast Cooldown System has manual actuation switches on the main control board.

Online Monitor

A non-safety-related Online Monitor shall be provided to alert the plant staff to possible malfunctions. The ICCMS analog safety system shall have Class 1E isolated analog monitoring to the Online Monitor.

The signals to be monitored will be sent to an isolator on the originating card. The isolator performs no safety function, but guarantees that no down stream failure can affect the safety function of the card. The signals then go to a multiplexer in each cabinet. The multiplexer performs no safety function, and because of the isolators on each card, multiplexer failures can not affect the ICCMS ability to perform any of its safety functions.

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

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APPENDIX F

GRID STABILITY

Appendix F Grid Stability

Introduction

Progress Energy Florida (PEF) Crystal River Unit 3 (CR-3) is completing a series of uprates, in three separate steps, for a total net uprate of 180 MW. These uprates are scheduled to be completed by Winter 2011. At the end of the uprate, CR-3 will have a maximum net summer output of 1020 MW and a maximum net winter output of 1070 MW. To accommodate the additional generation at CR-3, specific modifications have been or will be completed to assure safe reliable generation at the uprated power level. These modifications include modification of the main generator, replacement of the low and high pressure turbines, and upgrades to multiple non-safety related components that will support operation at higher power levels. Refer to Appendix E for additional discussion of the EPU related modifications.

In addition, as part of the grid stability evaluation, the CR-3 switchyard components and offsite transmission network were studied to confirm that the effects of the uprate would not create unstable conditions on the grid. This Appendix provides the results of those studies.

Study Methodology

A transient stability analysis, a steady state load flow analysis (i.e., analyzing for the adequacy of thermal and voltage conditions), and a short circuit analysis were performed to investigate the CR-3 uprated power level impact on the grid.

Base cases represented the generation at CR-3 prior to any of the uprates with the 962 MVA Generator Step-Up Transformer (GSU) in-service, and the transfer cases represented the uprate and the new 1200 MVA GSU.

All 69 kV and above facilities within the Florida Reliability Coordinating Council (FRCC) Region were evaluated for any overloads.

Short Circuit Study

The short circuit study did not identify any over-dutied breakers within the FRCC Region due to the increased generation at CR-3. All transmission system breakers have sufficient capacity to absorb the CR-3 uprate and maintain the capability of performing their function.

Results

The Transmission Working Group and the Stability Working Group from the FRCC have determined that the proposed interconnection and integration of CR-3 to serve PEF’s load is reliable, adequate, and does not adversely impact the FRCC transmission system.

Enclosures:

FRCC Evaluation of PEF’s Crystal River Unit 3, Generator Interconnection Service Request

APPENDIX F

Enclosure 1

FRCC Evaluation of PEF's Crystal River Unit 3, Generator Interconnection Service Request

FRCC Evaluation of PEF's Crystal River Unit 3

Generator Interconnection Service Request

	Rev 1
Prepared by TWG	November/2008
Accepted by PC	

PEF's Crystal River Unit 3 Generator Interconnect Study Request

FRCC's TWG and SWG Review

Summary

Florida Reliability Coordinating Council's (FRCC) Transmission Working Group (TWG) and the Stability Working Group (SWG) have evaluated the proposed interconnection and integration of the Crystal River Unit 3 uprate (CR3) to serve Progress Energy Florida's (PEF) load.

Based upon the information provided by PEF, the TWG and SWG have determined that the proposed interconnection and integration of the uprated CR3 to serve PEF's load is reliable, adequate and does not adversely impact the FRCC transmission system.

Project and Study Assumptions

PEF's CR3 will undergo a series of up-rates, to be completed in three separate steps fir a total net up-rate of 180MW, scheduled to be completed by 12/2011. At the end of the up-rate, CR3 will have a net summer output of 1020MW and a net winter output of 1070 MW. To accommodate the additional generation at CR3, the 500/22 kV GSU has been upgrade from 962 MVA to 1200 MVA (completed 12/2007).

CR3 scheduled up-rates

- 12/1/2007 – 14 MW
- 12/1/2009 – 26 MW
- 12/1/2011 – 140 MW

Study Methodology

The TWG reviewed the proposed interconnection and integration of CR3 under a wide range of conditions as described below:

- Steady State Analysis
    - The steady state analysis was based on 2007 FRCC Final Base Cases (Rev 4.1) representing summer peak conditions for 2009 through 2013 and winter peak conditions for 2009/10 through 2013/14. These base cases represent firm commitments. The base cases represent the generation at CR3 prior to any up-rates with the 962 MVA GSU in-service.
    -
    - The study cases were developed from the base cases by modeling CR3 at full output with the upgrade to the GSU to 1200 MVA. Code 30 generation at Bartow was removed to accommodate the output of CR3. Under normal operating conditions (Category A), all facilities remained within applicable ratings.
    -
    - The Category B contingency analysis preformed on this study included all facilities 69 kV and greater in the FRCC region.

- Short Circuit Analysis
  - The 2007 FRCC short circuit case (Final) representing summer 2009 was used as the Base Case. The study cases were developed from the base case by modeling CR3s with the transmission expansion facilities discussed above in the Project and Study Assumptions section.

- Stability Analysis
  - Dynamic simulations performed by PEF were based on a 2006 FRCC dynamics case representing summer 2012 peak load conditions. The study cases were developed from the base case by modeling the CR3 Unit with the GSU upgrade discussed above in the Project and Study Assumptions section.

Results

1. Steady State Analysis

The single contingency analysis was comprehensive and complete. The study evaluated facilities 69 kV and above. The table shown in Attachment A summarizes the results of the single contingency analysis for each of the study years showing the impact of CR3 on the Bulk Electric System. For each facility loading shown in Attachment A, a mitigation plan was developed.

2. Short Circuit Analysis

PEF provided short circuit cases to the TWG for any potentially impacted utilities to review and evaluate. No impacts were identified by any of the TWG members.

3. Stability Analysis

The PEF stability analysis was comprehensive and complete. The SWG reviewed the stability analysis results provided by PEF and concluded that there are no significant issues for delayed clearing Category D events in the vicinity of CR3.

Conclusion

Based upon the information provided by PEF, the TWG has determined that the proposed interconnection and integration of the CR3 to serve PEF's load is reliable, adequate and does not adversely impact the FRCC transmission system.

CR3 MUST SCREENING		SUMMER 2012 -2014						WINTER 2011 - 2014							
Contingency	**Facility**	S2012-B	S2012-T	S2013-B	S2013-T	S2014-B	S2014-T	W2011-B	W2011-T	W2012-T	W2012-B	W2013-B	W2013-T	W2014-B	W2014-T
3551 CENT FLA 500 3555 CRYST RV 500 1	3518 BRKRIDGE 230 3548 BRDG-DUM 500 1								103.6				102.1		102.5
Remedy: PEF will decrease generation at CR-5 and increase generation elsewhere															
7431 STC EAST 230 7438 STC EAST 69.0 1	7439 STC CENT 69.0 7441 STC SOU 69.0 1		100		100.5										
3519 BRKSVL W 230 3520 BRKSVWTP 230 1	3450 BRKRIDGE 115 3518 BRKRIDGE 230 2								100.1						
Remedy: PEF will open the following line segments Brooksville - Neff Lk Tp (3350-3419), Brooksville - Consolidated Rock (3350-3348) and Brooksville - Nobleton Tp (3350-3403)															
348 LAURELWD 230 1781 LAURLDIS 230 1	337 VENICE 138 1711 CENTER 138 1												100		
REMEDY: Planned Laurelwood distribution station (LAURLDIS) now planned to be fed radially from 138 kV at Laurelwood. Eliminates this overload.															
2437 DEBARY 230 2585 N LONGWD 230 1	705 SYLVAN 230 742 SANFORDO 230 1	...	...	...	...	...	...	...	...	...	...	...	...	...	100.3
REMEDY: Open Sylvan (705)-Sanford (742) ckt 1-230kV line section. Redispatch generation in PEF															
3521 CENT FLA 230 3552 CENT-DM2 500 1	3521 CENT FLA 230 3553 CENT-DUM 500 1	...	...	...	...	...	...	103.5	108.2	...	...	...	...	...	...
Err: PEF's recent TX study of the 500/230 kV TX at Central Florida determined that C.FL 500/230 kV TX #1 has a B rating of 840 MVA, which is not exceeded under this contingency															
3521 CENT FLA 230 3553 CENT-DUM 500 1	3521 CENT FLA 230 3552 CENT-DM2 500 1	...	...	...	...	...	...	103.5	108.2	...	...	...	...	...	...
Err: PEF's recent TX study of the 500/230 kV TX at Central Florida determined that C.FL 500/230 kV TX #1 has a B rating of 840 MVA, which is not exceeded under this contingency															
3551 CENT FLA 500 3552 CENT-DM2 500 1	3521 CENT FLA 230 3553 CENT-DUM 500 1	...	...	...	...	...	...	103.5	108.2	...	...	...	...	...	...
Err: PEF's recent TX study of the 500/230 kV TX at Central Florida determined that C.FL 500/230 kV TX #1 has a B rating of 840 MVA, which is not exceeded under this contingency															
3551 CENT FLA 500 3553 CENT-DUM 500 1	3521 CENT FLA 230 3552 CENT-DM2 500 1	...	...	...	...	...	...	103.5	108.2	...	...	...	...	...	...
Err: PEF's recent TX study of the 500/230 kV TX at Central Florida determined that C.FL 500/230 kV TX #1 has a B rating of 840 MVA, which is not exceeded under this contingency															
2416 DLTONA E 115 2419 TURNR 115 1	2415 DELTONA 115 2419 TURNR 115 1	...	...	...	...	...	...	...	100.2	...	...	...	...	...	...
Remedy: PEF will decrease generation at Turner and increase generation elsewhere															
8610 FISHHAWK 230 9050 PEBB 230 1	8602 HAMPTN 69.0 8654 HOPEWL T 69.0 1	...	...	...	...	...	...	...	100.3	...	...	...	...	...	...
Remedy: Open the Willow Oak transformer (8680 - 8682)															
8610 FISHHAWK 230 9050 PEBB 230 1	8652 HOPEWELL 69.0 8654 HOPEWL T 69.0 1	...	...	...	...	...	...	...	101	...	...	...	...	...	...
Remedy: Open the Willow Oak transformer (8680 - 8682)															
8600 HAMPTN 230 8602 HAMPTN 69.0 1	8652 HOPEWELL 69.0 8654 HOPEWL T 69.0 1	...	...	...	...	...	...	...	100.5	...	...	...	...	...	...
Remedy: Open the Willow Oak transformer (8680 - 8682)															
8600 HAMPTN 230 8610 FISHHAWK 230 1	8652 HOPEWELL 69.0 8654 HOPEWL T 69.0 1	...	...	...	...	...	...	...	100.5	...	...	...	...	...	...
Remedy: Open the Willow Oak transformer (8680 - 8682)															
3550 BRKRIDGE 500 3555 CRYST RV 500 1	3521 CENT FLA 230 3552 CENT-DM2 500 1	...	...	...	...	...	...	...	104.6	...	...	...	...	...	...
Err: PEF's recent TX study of the 500/230 kV TX at Central Florida determined that C.FL 500/230 kV TX #2 has a B rating of 802 MVA, which is not exceeded under this contingency															
3550 BRKRIDGE 500 3555 CRYST RV 500 1	3521 CENT FLA 230 3553 CENT-DUM 500 1	...	...	...	...	...	...	...	104.6	...	...	...	...	...	...
Err: PEF's recent TX study of the 500/230 kV TX at Central Florida determined that C.FL 500/230 kV TX #2 has a B rating of 802 MVA, which is not exceeded under this contingency															

Contingency	Facility	S2012-B	S2012-T	S2013-B	S2013-T	S2014-B	S2014-T	W2011-B	W2011-T	W2012-T	W2012-B	W2013-B	W2013-T	W2014-B	W2014-T
3039 JASPER 69.0 3113 JASPER 115 1	3027 FT WHT A 69.0 3043 LIVE OAK 69.0 1						100.1								
Remedy: PEF will utilize DSM as needed															
474 VOLUSIA 230 718 LPGA 230 1	378 BUNNELL 115 911 KORONA 115 1				100.5										
REMEDY: Not due to CR3. Less than 3% impact, due to displacement of 186 MW all at Intercession. Ampacity upgrade proposed for Bunnell-Korona line section S14 (267MVA)															
2757 FTGRN3TP 69.0 2828 VANDOLAH 69.0 1	2698 FTGN11TP 69.0 2828 VANDOLAH 69.0 1		100.1												
Remedy: PEF will utilize DSM as needed															
3335 ANDERSEN 69.0 3336 WLDWD TP 69.0 1	3366 DALLAS 69.0 3525 DALLAS 230 1														100.4
Remedy: PEF has a project to add a 2nd 230/69 kV TX at Dallas Substation by W-2013															
8110 OHIO-N 230 8500 11TH AVE 230 1	8502 ELEVEN-E 69.0 8506 FOURTEEN 69.0 1														100.8
Remedy: This branch is scheduled for rebuild (147.5 MVA) by 06/01/2012 (project in FY08 cases)															
8350 DAVIS ROAD 230 8720 WHEELER ROAD 230	8604 ALEX-E 69.0 8634 WILSN T1 69.0 1														101.4
Remedy: This branch is scheduled for rebuild (160 MVA) by 12/01/2010 (project in FY08 cases)															
8518 PLANT 69.0 8528 WASH 69.0 1	8526 HKRSPT-S 69.0 8532 HARBOUR 69.0 1														101.6
Remedy: Open Hookers Point 69 kV bus-tie (8524 - 8526)															
3162 FT WHT N 230 3163 FT WHT S 230 1	3085 FT WHT B 69.0 3162 FT WHT N 230 1														106.3
Remedy: PEF will open the Ft. White Tp - High Springs 69 kV line (check with Bart -- not a real contingency -- line modeled to show the state split?)**															
3818 HUDSON2 115 3836 HUDSN 230 2	3806 HUDSN 115 3836 HUDSN 230 1														101.9
Remedy: PEF will open Hudson - Heritage Tp 115 kV line															
469 SANFORDI 230 2437 DEBARY 230 1	705 SYLVAN 230 742 SANFORDO 230 1											104.1	107.9		
REMEDY: Open Sylvan (705)-Sanford (742) ckt 1-230kV line section. Redispatch generation in PEF															
2769 INTERCSN 69.0 2883 INTERCSN 230 1	2769 INTERCSN 69.0 2883 INTERCSN 230 2							110.1	120.4						
Remedy: PEF will increase generation at the Intercession City 69 kV peakers to relieve the overload															
2769 INTERCSN 69.0 2883 INTERCSN 230 2	2769 INTERCSN 69.0 2883 INTERCSN 230 1							110.1	120.4						
Remedy: PEF will increase generation at the Intercession City 69 kV peakers to relieve the overload															
2073 SPG LAKE 230 2580 ALTAMONT 230 1	2509 ALTAMONT 69.0 2580 ALTAMONT 230 1									115.8	119.3				
Remedy: PEF has a project to increase the size of the 230/69 kV Altamonte TX to a 300/336 MVA by S-2012															
2884 KATHLEEN 230 3838 ZEPHYR N 230 1	8604 ALEX-E 69.0 8634 WILSN T1 69.0 1										100				
Remedy: This branch is scheduled for rebuild by 12/01/2012 (project in FY08 cases)															
469 SANFORDI 230 2580 ALTAMONT 230 1	705 SYLVAN 230 742 SANFORDO 230 1										101.9				
REMEDY: Open Sylvan (705)-Sanford (742) ckt 1-230kV line section. Redispatch generation in PEF															
1056 KORONA 230 1705 EAGLE 230 1	415 ORMOND 115 590 COQUINA 115 1				100.9										
REMEDY: Open Ormond-Coquina 15kV line															
7431 STC EAST 230 7438 STC EAST 69.0 1	7439 STC CENT 69.0 7441 STC SOU 69.0 1		100		100.4										
5565 SO WOOD 115 5701 SO WOOD 230 1	5564 HOLDEN 115 5568 MICHIGAN 115 1		101.2												

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APPENDIX G

ACRONYMS IN ADDITION TO THOSE IN RS-001

Appendix G: Acronyms in Addition to Those in RS-001

%FP	Percent Full Power
%WD	Percent Withdrawn
95/95	95% probability at the 95% confidence level
A/E	Architect/Engineer
a/o	Atom Percent
AAC	Alternate AC (Source)
AC	Alternating Current
ACIS	Automatic Closure and Interlock System
ADV	Atmospheric Dump Valve
AEC	Atomic Energy Commission
AFW	Auxiliary Feedwater
AH-XC	Reactor Building Purge
AH-XD	Auxiliary Building Supply System
AH-XE	Fuel Handling Area Supply System
AH-XF	Decay Heat Closed Cycle Pump Cooling System
AH-XG	Spent Fuel Coolant Pumps Air Handling System
AH-XH	Spent Fuel Pit Supply System
AH-XJ	Auxiliary and Fuel Building Exhaust System
AH-XK	Control Complex Ventilation System
AH-XL	Emergency Diesel Generator Air Handling System
AH-XM	Miscellaneous Area HVAC System
AH-XN	Turbine Building Non 1E Battery Room Ventilation System
AH-XN	Turbine Building Ventilation System
AH-XS	Emergency Feedwater Initiation and Control
AH-XT	Control Complex Dedicated Cooling Supply
AH-XU	Emergency Feedwater Pump Building HVAC System
AL	Analytical Limits
ALARA	As Low As Reasonably Achievable
AMCA	Air Movement Control Association
AMSAC	ATWS Mitigating System Actuation Circuitry
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
API	Axial Power Imbalance
APS	Auxiliary Pressurizer Spray
APSR	Axial Power Shaping Rod
APSRA	Axial Power Shaping Rod Assembly
AR	Condenser Air Removal System
AR	Air Removal
ARI	All Rods In
ARI-2	All Rods In with Two Maximum Combined Worth Rods Out
ARO	All Rods Out
ARP	Air Removal Pump
ART	Anticipatory Reactor Trip

ARTS	Anticipatory Reactor Trip System
ASME	American Society of Mechanical Engineers
ASME B&PV	American Society of Mechanical Engineers Boiler and Pressure Vessel
AST	Alternate Source Term
AT	Anticipated Trip Without SCRAM
ATWS	Anticipated Transients Without Scram
AULD	Automatic Unit Load Demand
B&PV	Boiler and Pressure Vessel
B&W	Babcock and Wilcox
BAST	Boric Acid Storage Tank
BCM	Boiler Condenser Mode
bhp	Brake Horsepower
BL	Bulletin
BOC	Beginning of Cycle
BOCR	Bottom of Core Recovery
BOL	Beginning of Life
BOP	Balance of Plant
BPRA	Burnable Poison Rod Assembly
BS	Reactor Building Spray System
BTP	Branch Technical Position
BWR	Boiling Water Reactor
BWST	Borated Water Storage Tank
CA	Chemical Addition
CASS	Cast Austenitic Stainless Steel
CB	Core Barrel
CBC	Critical Boron Concentration
CCHE	Control Complex Habitability Envelope
CD	Condensate System
CDF	Core Damage Frequency
CDHE	Condensate/Feedwater Heat Exchanger
CDP	Condensate Pump
CDPO	Condensate Pump Discharge Oxygen Level
CE	Combustion Engineering
CF	Chemistry Factor
CF	Clad Failure
CF	Core Flood
CFLB	Core Flood Line Break
CFM	Centerline Fuel Melt
CFR	Code of Federal Regulations
CFT	Core Flood Tank
CH	Chilled Water System
CHF	Critical Heat Flux
CLB	Current Licensing Basis
CLPD	Cold Leg Pump Discharge
CLPS	Cold Leg Pump Suction

COLR	Core Operating Limits Report
CPB	Condensate Booster Pump
CPCS	Containment Emergency Sump pH Control
CPDO	Condensate Pump Discharge Oxygen
CR	Control Room
CR-3	Crystal River Unit 3
CRA	Control Rod Assembly
CRAVS	Control Room Area Ventilation System
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CRDS	Control Rod Drive System
CREA	Control Rod Ejection Accident
CREVS	Control Room Emergency Ventilation System
CRGT	Control Rod Guide Tube
CRH	Control Room Habitability
CRHVAC	Control Room Heating, Ventilation, and Air Conditioning System
CSA	Core Support Assembly
CSS	Core Support Shield
CST	Condensate Storage Tank
CDF	Core Damage Frequency
CTF	Core Tank Flow
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
CvUSE	Charpy Upper-Shelf Energy
CWA	Cold Water Accident
CWS	Circulating Water System
DAW	Dry Active Waste
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBLOCA	Design Basis Loss of Coolant Accident
DC	Decay Heat Closed Cycle Cooling
DC	Design Criteria
DC	Direct Current
DCF	Dose Conversion Factor
DCHE	DC Heat Exchanger
DE	Dose Equivalent
DEG	Double-Ended Guillotine
DEHL	Double-Ended Hot Leg
DF	Decontamination Factor
DG	Draft Guide
DH	Decay Heat Removal System
DHDL	Decay Heat Drop Line
DHHE	DH Heat Exchanger
DHR	Decay Heat Removal
DNB	Departure from Nucleate Boiling

DNBR	Departure from Nucleate Boiling Ratio
DOT	Department of Transportation
DPA	Displacement Per Atom
DRA	Dropped Rod Accident
DRW	Dropped Rod Worth
DSS	Diverse SCRAM System
DTC	Doppler Temperature Coefficient
DW	Deadweight
EAB	Exclusion Area Boundary
EC	Engineering Change
ECCS	Emergency Core Cooling System
ECT	Eddy Current Technique
EDBD	Enhanced Design Basis Document
EDG	Emergency Diesel Generator
EDY	Effective Degradation Years
EFDS	Equipment and Floor Drainage System
EFIC	Emergency Feedwater Initiation and Control
EFP	Emergency Feedwater Pump
EFPB	Emergency Feedwater Pump Building
EFPD	Effective Full Power Day
EFPY	Effective Full Power Year
EFT	Emergency Feedwater Tank
EFW	Emergency Feedwater
EHCS	Electro-hydraulic Control System
EM	Evaluation Models
EMA	Equivalent Margin Analysis
EMS	Excore Monitoring System
EOC	End of Cycle
EOL	End of Life
EOLL	End of Licensed Length
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Environmental Qualification
EQML	EQ Equipment Master List
EQPPD	Environmental Qualification Plant Profile Document
EQXE	Equilibrium Xenon
ES	Engineered Safeguards
ESAS	Engineered Safeguards Actuation System
ESF	Engineered Safety Feature
FA	Fuel Assembly
FAC	Flow Accelerated Corrosion
FB	Former Bolt
FCS	Fast Cooldown System
FD	Flow Distributor

FD&A	Fuel Design and Analysis
FEI	Fluid-Elastic Instability
FERC	Federal Energy Regulatory Commission
FF	Fluence Factor
FFCD	Final Fuel Cycle Design
FGR	Federal Guidance Report
FHA	Fuel Handling Accident
FIV	Flow Induced Vibration
FOA	Forced Oil and Air
FP	Full Power
FPC	Florida Power Corporation
FPP	Fire Protection Program
FRCC	Florida Reliability Coordinating Council
FSAR	Final Safety Analysis Report
FTC	Fuel Transfer Canal
FW	Feedwater
FWBP	Feedwater Booster Pump
FWLB	Feedwater Line Break
GALL	Generic Aging Lessons Learned
GDC	General Design Criteria
GL	Generic Letter
GMS	Gamma Metrics System
GOTHIC	Generation of Thermal Hydraulic Information for Containment
GPM	Gallons per Minute
GSU	Generator Step-up Transformer
GWD	Giga Watt Days
GWd/mtU	Gigawatt Day per Metric Ton of Uranium
GWDS	Gas Waste Disposal System
HEI	Heat Exchange Institute
HELB	High Energy Line Break
HELBA	High Energy Line Break Accident
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air
HFE	Human Failure Event
HFP	Hot Full Power
HLI	Hot Leg Injection
HLSG	Hot Leg Near Steam Generator
HMP	High Mechanical Performance
HPI	High Pressure Injection
HRA	Human Reliability Analysis
HTP	High Thermal Performance
HX	Heat Exchanger
HZP	Hot Zero Power
I&C	Instrumentation and Controls
IA	Instrument Air

IASCC	Irradiation Assisted Stress Corrosion Cracking
IB	Intermediate Building
IC	Integrated Control
ICC	Inadequate Core Cooling
ICCMS	Inadequate Core Cooling Mitigation System
ICCMS	Inadequate Core Cooling Mitigation System
ICS	Integrated Control System
IE	Irradiation Embrittlement
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	Intergranular Stress Corrosion Cracking
IMI	Incore Monitoring Instrumentation
IMS	Incore Monitoring System
IN	Information Notice
INBW	Inverse Boron Worth
IPBD	Isolated Phase Bus Duct
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPSAR	Integrated Plant Safety Assessment Systematic Evaluation Program
ISCM	Inadequate Subcooling Margin
ISCM	Inadequate Subcooling Margin
ISG	Intact Steam Generator
ISLOCA	Interfacing System Loss of Coolant Accident
ITS	Improved Technical Specifications
ISI	Inservice Inspection
JIT	Just in Time
LAR	License Amendment Request
LB	Large Break
lbm/hr	Mass Flow Rate
LBB	Leak-Before-Break
LBLOCA	Large Break Loss of Coolant Accident
LCB	Lower Core Barrel
LCO	Limiting Condition of Operation
LERF	Large Early Release Frequency
LEFM	Leading Edge Flow Meter
LHR	Linear Heat Rate
LHRTM	Linear Heat Rate to Melt
LLHS	Light Load Handling System
LLR	Letdown Line Rupture
LOCA	Loss of Coolant Accident
LOCF	Loss of Coolant Flow
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LP	Low Pressure
LPI	Low Pressure Injection
LPZ	Low Population Zone

LRA	License Renewal Application
LR	Locked Rotor
LRA	Locked Rotor Accident
LOSCM	Loss of Subcooling Margin
LTC	Long Term Cooling
LTOP	Low Temperature Overpressure Protection
LTS	Lower Thermal Shield
LWDS	Liquid Waste Disposal System
LWR	Light Water Reactor
M&E	Mass and Energy
MAP	Maximum Allowable Peaking
MARP	Maximum Allowable Radial Peaking
MCB	Main Control Board
MCC	Motor Control Center
MCES	Main Condenser Evacuation System
MCR	Main Control Room
MDA	Moderator Dilution Accident
MDNBR	Minimum Departure from Nucleate Boiling Ratio
MELB	Moderate Energy Line Break
MFW	Main Feedwater
MFWI	Main Feedwater Isolation
MFWP	Main Feedwater Pump
MHE	Maximum Hypothetical Earthquake
MIRVSP	Master Integrated Reactor Vessel Surveillance Program
MOC	Middle of Cycle
MOL	Middle of Life
MOV	Motor Operated Valve
MRP	Material Reliability Project
MS	Main Steam
MSIV	Main Steam Isolation Valve
MSLI	Main Steam Line Isolation
MSLB	Main Steam Line Break
MSR	Moisture Separator Reheater
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
MTU	Metric Tonne Uranium
MUP	Makeup and Purification
MUR	Measurement Uncertainty Recapture
MUT	Makeup Tank
MUV	Makeup Valve
MWd/mtU	Megawatt Day per Metric Ton of Uranium
MWt	Megawatts Thermal
MWth	Megawatts Thermal
NDE	Non-Destructive Examination
NDTT	Nil Ductility Transition Temperature

NEI	Nuclear Energy Institute
NERC	North American Electric Reliability Corporation
NNI	Non-Nuclear Instrumentation
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
O&M	Operations and Maintenance
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OE	Operating Experience
OEM	Original Equipment Manufacturer
OL	Operating Limit
OL	Operating License
OMA	Operator Manual Action
OPC	Over-Speed Protection Circuit
OTSG	Once-Through Steam Generators
P_b	Primary Bending Stress Intensity
P_L	Primary Local Stress Intensity
P_m	Primary Membrane Stress Intensity
PCD	Pump Coast Down
PCP	Process Control Program
PCS	Power Conversion System
PCSV	Pressurizer Code Safety Valve
PCT	Peak Cladding Temperature
PIE	Post-Irradiation Examination
PMH	Possible Maximum Hurricane
POD	Point of Discharge
PORV	Power Operated Relief Valve
PPC	Plant Process Computer
ppmB	Parts per Million Boron
PRA	Probabilistic Risk Analysis
PRT	Pressurizer Relief Tank
PSA	Probabilistic Safety Analysis
PSC	Preliminary Safety Concern
psia	pound per square inch absolute
PSV	Pressurizer Safety Valve
P/T	Pressure/Temperature
PTLR	Pressure Temperature Limit Report
PTS	Pressurized Thermal Shock
p.u.	per unit
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
PZR	Pressurizer
Q	Secondary Stress Intensity

QA	Quality Assurance
QPT	Quadrant Power Tilt
RADTRAD	Radionuclide and Transport and Removal and Dose Estimation
RAI	Request for Additional Information
RB	Reactor Building
RBCU	Reactor Building Cooling Unit
RBES	Reactor Building Emergency Sump
RBFA	Reactor Building Fan Assembly
RBIC	Reactor Building Isolation and Cooling
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCDT	Reactor Coolant Drain Tank
RCITS	Reactor Coolant Inventory Tracking System
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
REA	Rod Ejection Analysis
REA	Rod Ejection Accident
RG	Regulatory Guide
RHR	Residual Heat Removal
RIR	Reactivity Insertion Rate
RIS	Regulatory Issue Summary
RMS	Radiation Monitoring System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RS	Review Standard
RSP	Remote Shutdown Panel
RSS	Remote Shutdown System
RSWPS	Radioactive Solid Waste Packaging System
RT	Random Turbulence
RT_{NDT}	Reference Temperature for Nil-Ductility
RT_{PTS}	Reference Temperature for Pressurized Thermal Shock
RTP	Rated Thermal Power
RV	Reactor Vessel
RVCH	Reactor Vessel Closure Head
RW	Raw Water System
RW	Nuclear Services Seawater and the Decay Heat Seawater System
S_m	Allowable Membrane Stress Intensity
SAFDL	Specified Acceptable Fuel Design Limit
SAG	Severe Accident Guideline
SAR	Safety Analysis Report
SAT	Systematic Approach to Training
SLOCA	Small Loss-of-Coolant Accident
SBLOCA	Small Break Loss-of-Coolant Accident

SBO	Station Blackout
SC	Secondary Service Closed Cycle Cooling
SCC	Stress Corrosion Cracking
SCD	Statistical Core Design
SCHE	Secondary Cooling Heat Exchanger
SCM	Subcooling Margin
SCP	Secondary Cooling Pump
SCRAM	Safety Control Rod Axe Man
SDBC	Shutdown Boron Concentration
SDL	Statistical Design Limit
SDM	Shutdown Margin
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SF	Spend Fuel Cooling System
SG	Steam Generator
SGBD	Steam Generator Blowdown
SGTP	Steam Generator Tube Plug
SGTR	Steam Generator Tube Rupture
SL	Safety Limits
SLB	Steam Line Break
SP	Surveillance Procedure
SPDS	Safety Parameter Display System
SRP	Standard Review Plan
SRSS	Square Root of Sum of Squares
SRV	Steam Relief Valve
SRV	Safety Relief Valve
SRW	Stuck Rod Worth
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSHT	Surveillance Specimen Holder Tube
SW	Nuclear Services Closed Cycle Cooling
SWG	Stability Working Group
SWHE	SW Heat Exchangers
SWP	Service Water Pump
T_{AVG}	Average Moderator Temperature
T_{fw}	Feedwater Temperature
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TDL	Thermal Design Limit
TE	Thermal Aging Embrittlement
TEDE	Total Effective Dose Equivalent
TGSCC	Transgranular Stress Corrosion Cracking
THRM	Thermal
TID	Total Integrated Dose

TIL	Time In Life
TLD	Thermoluminescence Dosimeter
TPD	Total Power Deficit
TRIP	Rapid Rod Insertion
TSC	Technical Support Center
TSP-C	Trisodium Phosphate Dodecahydrate
TSV	Turbine Stop Valve
TT	Thermally Treated
UCB	Upper Core Barrel
UCB	Upper Core Limit
UHS	Ultimate Heat Sink
ULD	Unit Load Demand Control
US	United States
USE	Upper-Shelf Energy
UT	Ultrasonic Testing
UTS	Upper Thermal Shield
VE	Visual Examination
VQP	Vendor Qualification Package
VS	Void Swelling
VS	Vortex Shedding
WGDT	Waste Gas Decay Tank
WGDTR	Waste Gas Decay Tank Rupture
wt%	Weight Percent
χ/Q	Atmospheric Dispersion Factor