3.10.3.14 Inadvertent Loading of a Fuel Assembly into an Improper Position

This event is not affected by the initial power level, so no additional evaluation is required to support the power uprate.

3.10.3.15 Steam Generator Tube Rupture (SGTR) Accident

A Steam Generator Tube Rupture (SGTR) is a postulated double-ended rupture of a steam generator tube with unrestricted discharge from both ends of the tube. The acceptance criteria are related to offsite doses and further degradation of the primary-to-secondary pressure boundary beyond the affected tube.

A SGTR is a breach of the reactor coolant pressure boundary and results in a transfer of primary coolant to the secondary system. The core protection aspects of a SGTR are bounded by small break LOCA. The SGTR event is analyzed to determine the offsite doses resulting from the release of contaminated primary coolant into the steam generator and to the atmosphere via the MSSVs.

The SGTR analysis of record is based on a constant leak rate of 435 gpm. The leak flow rate is based on critical flow from each end of the rupture tube. This leak rate was assumed constant until the plant was cooled down to the decay heat removal cut-in temperature. This leak flow rate is conservative because it does not credit the decrease in the leakage rate with RCS depressurization or the secondary side pressurization following reactor trip and turbine trip. The leak rate calculation for the SGTR analysis is independent of power level based on the analytical method used. The reactor coolant activity is based on power operation at 102% of 2772 MWt. These conservative parameters bound the proposed power uprate.

3.10.3.16 Control Rod Assembly (CRA) Ejection Accident

The Rod Ejection from Full Power (FP) event is a postulated event involving a physical failure of a pressure barrier component in the Control Rod Drive assembly and subsequent ejection of the control rod. The event is classified as an infrequent event. The acceptance criteria for the Rod Ejection from FP event relate to peak RCS pressure and peak fuel enthalpy.

The ejection of a control rod with the reactor at full power causes a rapid positive reactivity insertion. Core power and fuel temperatures increase rapidly. The rapid fuel temperature rise produces negative Doppler reactivity feedback that terminates the power excursion. A reactor trip occurs on high flux and the reactor is returned subcritical by control rod insertion. The primary safety valves provide steam relief to limit the peak RCS pressure to less than the acceptance criterion. Limiting the reactivity worth of a given rod in the fuel design and the initial fuel enthalpy at full power will ensure that the peak fuel enthalpy does not exceed the maximum allowable limit.

The current analysis for the Control Rod Ejection Accident for the DBNPS was initiated from 102% of 2772 MWt using RELAP 5/MOD2-B&W. This bounds the power uprate.

3.10.3.17 Break in Instrument Lines or Lines from Primary System that Penetrate Containment

This transient is typically represented by a double-ended break of the letdown line and is considered a limiting fault event. While this transient is considered a small break loss of coolant accident, it is performed to assess the offsite dose release consequences from the release of primary coolant outside the reactor building. A new analysis was performed using an initial power level of 3025 MWt (102% of 2966 MWt). The results bound the power uprate.

3.10.3.18 Anticipated Transients Without Scram (ATWS)

The ATWS transients are considered beyond the original design basis of the B&W-designed plants. The acceptance criterion is that the peak RCS pressure would not exceed ASME Service Level C limits, i.e., 125% of the RCS design pressure.

In order to comply with 10CFR 50.62, the DBNPS installed a diverse scram system (DSS). The DSS initiates a redundant trip signal on high RCS pressure with an actuation setpoint corresponding to 2450 ± 25 psig. Consequently, the peak RCS pressure predicted for the ATWS events is significantly below the maximum pressure criterion allowed for the ATWS event, i.e. 3200 psig. Because the DSS setpoint, pressurizer safety valve setpoints, and pressurizer safety valve flow characteristics are not affected by the power uprate, a small increase in the initial core power will only result in a small increase in the rate of pressurization. There is significant margin in the analysis, however, to accommodate any small change in the peak pressure prediction. Therefore, the power uprate is bounded by the existing design.

3.10.4 Revised Power Calorimetric Uncertainties

The expression for core power in terms of a secondary side heat balance is shown below. This equation is used by the Core Thermal Power Analysis software for the plant computer.

$$Q_C = W_{FWA}(H_{SA} - H_{FWA}) + W_{FWB}(H_{SB} - H_{FWB}) + Q_{LD} - Q_{MU} - Q_{RCP} + Q_{LOSS}$$

Where W_{FWA}, W_{FWB}	Feedwater flows in Loop A & B
H_{SA}, H_{FWA}, H_{SB}, H_{FWB}	Steam & feedwater enthalpies for Loops A & B
$Q_{LD} = W_{LD} H_{LD}$	Heat loss due to primary side letdown flow
$Q_{MU} = W_{MU} H_{MU}$	Heat added due to makeup and net seal injection
Q_{RCP}	Heat added due to RC pumps
Q_{LOSS}	Ambient heat losses from the RCS
W_{LD}, W_{MU}	Letdown and Makeup Flow Rates
H_{LD}, H_{MU}	Letdown and Makeup Enthalpies

The ASME Performance Test Code 19.1 methodology was used to calculate the expected core thermal power uncertainty to be achieved using the Caldon CheckPlus™ System ultrasonic flow meter. The analysis concluded that the core thermal power uncertainty would be 0.37%, thus allowing a power uprate of 1.63% to be pursued.

3.10.5 RPS/SFAS/SFRCS/ARTS Setpoints

For the systems response analyses of the USAR Chapter 15 non-LOCA events, only the high flux, high reactor coolant system (RCS) pressure and low RCS pressure setpoints are credited. For the core response events, the flow-related trip setpoints (the flux-to-flow, power to pumps, power/imbalance/flow [P/I/F] and variable low pressure trip [VLPT] setpoints) are modeled. The P/I/F setpoint is evaluated for each fuel cycle, so no specific assessment of this setpoint is necessary for the power uprate. The flux-to-flow setpoint, which is part of the P/I/F setpoint, is also verified during the reload process. An evaluation of the other RPS setpoints listed above, with respect to the power uprate, has been performed, and the necessary changes are described below.

Several systems related analyses have recently been performed. The power level modeled in these analyses was 3025 MWt (102% of 2966 MWt). The analyses also assumed that the current RPS setpoints were applicable. The results of these calculations confirmed that the current high and low RCS pressure setpoints were acceptable for the 3025 MWt power level. These transients, however, were limited to inventory loss or over-heating events where only the pressure setpoints would be challenged. Since the power level modeled, with the current setpoints, was more than the planned power uprate and the current setpoints were modeled, it can be concluded that the existing high and low RCS pressure setpoints remain valid for the power uprate.

The RPS high flux trip setpoint will continue to be based on a maximum overpower limit of 112 % of 2772 MWt, or 3105 MWt. This limit will not be adjusted as a result of the power uprate and will effectively protect against 110.2 % of the uprated power level of 2817 MWt. The TS high flux trip setpoint Allowable Value will be revised accordingly, as described in the license amendment application.

The VLPT trip setpoint provides steady-state and transient DNBR protection. The pressure-temperature limits for four- and three-pump operation were recalculated for the power uprate. As a result, a revision to the VLPT setpoint is required, as described in the license amendment application.

The only SFAS setpoint that is credited in the LOCA calculation is the low RCS pressure actuation trip signal. LOCA analyses have been performed at a power level of 3025 MWt (102% of 2966 MWt). These calculations comply with the requirements of 10CFR 50.46 and Appendix K and were performed using the current plant SFAS low RCS pressure setpoint. Since these analyses bound the proposed power uprate, no changes will be required to the SFAS setpoints.

Regarding SFRCS setpoints, Table 3-1 notes that the power uprate does not change SG pressure. Therefore the low steam line pressure trip setpoint of the SFRCS will not be affected. Since the SG pressure does not change, there will not be a change in the check valve back pressure for loss of MFW, therefore, no setpoint change is required for the SFRCS loss of MFW dP switch. In addition, as stated in Section 3.6.7.1, SG level will not appreciably change due to the power uprate. Therefore, SFRCS SG level setpoint changes are not required. Reactor Coolant pump status is provided for all four RCPs for loss of all RCP detection using a signal proportional to RCP motor current. Power uprate will not impact this SFRCS setpoint. The SFRCS SG high level trip, which is non-TS, will not be impacted by the power uprate. Based on the above, it is concluded that the power uprate will have no impact on the safety or operational functions of the SFRCS. No design changes will be required to the SFRCS as a result of the power uprate.

The function of the Anticipatory Reactor Trip System (ARTS) is to initiate a reactor trip upon detection of parameters that are an indication of nuclear plant secondary system upsets and an anticipatory signal of potential unsafe conditions in the reactor primary system. ARTS will trip the reactor upon a turbine trip, when reactor power is above an arming setpoint. ARTS will also trip the reactor upon a loss of both main feedwater pumps, or upon an SFRCS actuation. The setpoint for the reactor power level bistable is based on the capacity of the turbine bypass valves and the first bank of main steam safety valves to handle a trip. The combined steaming capacity of these valves provides sufficient margin, taking into account the proposed power uprate. Therefore, no change is required for the ARTS setpoint.

3.11 CONTAINMENT/BOP ACCIDENT EVALUATIONS

3.11.1 Mass and Energy Release Data

3.11.1.1 Subcompartment Analysis

The containment subcompartments in which a major loss-of-coolant accident could occur are the reactor cavity and the steam generator compartments. The walls of these chambers are designed to bear the combined loads of differential pressure and jet impingement resulting from breaks in the Reactor Coolant System (RCS) piping. These loads are primarily dynamic loads, which may be excluded from consideration using the using the Leak-Before-Break (LBB) evaluation procedures. Per USAR Section 3.6.2.2.1, "Pipe Restraint Design Criteria to Prevent Pipe Whip Impact Within the Containment Vessel," the RCS has been evaluated using the criteria of Standard Review Plan 3.6.3, Leak-Before-Break evaluation procedures. This criterion, in conjunction with General Design Criterion (GDC-4) of 10CFR50 Appendix A, allows the exclusion of the dynamic effects of a postulated pipe rupture. These subcompartments do not provide a containment function. Consequently, consideration of differential pressure and jet impingement loads on these compartments is no longer required.

3.11.1.2 Main Steam Line Mass and Energy Release Data

A main steam line break (MSLB) reanalysis was performed for the DBNPS. The purpose of the analysis was to generate mass and energy release data to be used in a containment peak pressure analysis. It was performed with bounding plant conditions to maximize heat generated in the RCS, heat transfer from primary to secondary, and maximum inventory in the OTSGs. Each of these conditions maximizes the mass and energy release through the break.

The MSLB reanalysis was performed using the RELAP5/MOD2 B&W computer code using an updated DBNPS-specific model that was initialized at 102% of 2772 MWt. A double-ended guillotine rupture of the main steam line in the line upstream of the Main Steam Isolation Valve (MSIV) was initiated at start of the run. The main feedwater system was included in the model to account for the transient effects of feedwater flow when the OTSG depressurizes. No credit was taken for control system action during the transient. Termination of main feedwater was initiated on an SFRCS low steam line pressure signal, including delay, and accounted for the stroke time of the motor-operated feedwater isolation valves. Air-operated feedwater control valves were conservatively not closed during the analysis because the motor-operated feedwater isolation valves have a longer stroke time. AFW was initiated on the SFRCS low steam line pressure signal, including delay, to the depressurized OTSG throughout the transient. No credit was taken for the ARTS trip on SFRCS actuation, which occurred within the first second of the transient. Delaying the reactor trip maximizes energy addition to the RCS.

3.11.1.3 LOCA Mass and Energy Release Data

A Large Break Loss-of-Coolant Accident (LBLOCA) mass and energy release analysis was performed in support of a future planned 7% core power uprate for the DBNPS. The analysis assumed a power level of 3025 MWt (102% of 2966 MWt). Although the 7% uprate has not been applied, this bounding analysis is the current LOCA analysis of record. The RELAP5/MOD2-B&W code was used to perform the entire blowdown and refill portions of the transient. Input model modifications were made to maximize mass and energy releases for the same postulated spectrum of breaks evaluated for peak containment pressure in Section 6.2 of the USAR.

3.11.2 Containment Analysis

3.11.2.1 MSLB and LOCA

The mass and energy releases derived using the methodology described above were used in evaluating the short term and long term containment performance. The short term (<300 seconds) mass and energy releases were taken from the RELAP5 Evaluation Model output, utilizing worst case break locations for effects on containment.

In order to evaluate the effect of increased power on the containment performance beyond 300 seconds following a LOCA, the energy release rate to the containment was estimated based on the core decay heat and the stored energy in the primary and secondary system metal and fluids.

The core decay heat for >300 seconds was calculated using equations given in Branch Technical Position ASB 9-2. In calculating the decay heat due to fission products, 20% uncertainty was applied for decay times less than 1000 seconds. For decay times greater than 1000 seconds, 10% uncertainty was applied. A core thermal power level of 3026 MW and an operating time of 17,000 hours were used in the calculation of decay heat. This corresponds to 102% of 1.07 x 2772 MW, to allow for a future planned 7% power uprate.

At 300 seconds the stored energy in the primary and secondary system metals and fluids was estimated to be approximately 330 million BTU. Approximately 50% of this energy is stored in the secondary system and 40% is in the RCS metal. The remaining 10% is in the RCS fluid. Since the stored energy release rate is dependent on the temperature difference between the emergency core cooling water and the RCS metal or secondary side fluid temperature, the energy addition to the containment will be much lower in the ECCS recirculation phase than in the injection phase. It is conservatively assumed that 80% of the stored energy will be released to the containment within 4500 seconds (start of ECCS recirculation) and the remaining 20% will be released over a 24 hour period following the accident.

The mass energy release data for a hot leg break at the steam generator, a hot leg break at the reactor vessel, and a cold leg break at the pump discharge were used to determine the break that results in a peak containment pressure and temperature. The containment analyses were performed using a revision of the Bechtel computer program COPATTA. This code is essentially identical to the version of COPATTA that was used in the original licensing of DBNPS except it is PC-based rather than mainframe-based. This code version was benchmarked to the original version. With identical input, the results were the same within machine precision. These analyses showed that the hot leg break at the steam generator resulted in a peak pressure of 36.8 psig and a peak containment temperature of less than 260 °F.

DBNPS Technical Specification 3/4.6.1.4, "Containment Systems – Internal Pressure," permits a normal operating containment pressure of 25 inches of water (0.9 psi) higher than outside atmospheric pressure. In order to prevent the station approaching this limit during low atmospheric pressure conditions, the containment pressure is administratively maintained well below 0.9 psi. However, assuming an initial positive pressure of 0.9 psi inside the containment prior to the LOCA, a peak LOCA pressure of 37.8 psig was estimated. This is less than the containment accident pressure of 40 psig per the containment design specification. Per the containment leak rate testing program, the containment is required to be leak tested at 38 psig. Normal test pressure is typically slightly higher than 38 psig. The last Integrated Leak Rate test was done at 38.6 psig. Therefore the power uprate does not have any impact on containment integrity.

A comparison of long term containment pressures and temperatures compared to previous licensing submittals is presented in Figures 3-1 and 3-2. Although the shape of the long term containment vapor temperature curve is different than previous submittals due to revised blowdown input, it can be seen that the peak temperatures are lower and the area under the curve is lower than the previous analyses. Therefore, neither the long term containment cooling nor qualification of the equipment is impacted by the power uprate. The peak sump water temperature and the long term sump water temperatures are similar to the previous licensing submittals, and it has been determined that there is no adverse impact on ECCS room cooling, on emergency diesel generator cooling, or on other required loads.

The containment performance following a MSLB was also evaluated using the COPATTA program. The steam generator blowdown calculation conservatively assumed a continued feed of 800 gpm auxiliary feedwater to the faulted generator. A comparison of the containment pressure and temperatures with previous licensing submittals is presented in Figures 3-3 and 3-4. The calculated temperatures are bounded by the previous analyses. Therefore, there is no impact on equipment qualification.

3.11.2.2 Combustible Gas Control

An increase in the power level will increase the hydrogen generation rate due to radiolytic decomposition of water. This mechanism is responsible for a majority of the hydrogen generation. Current calculations identify the theoretical radiolysis production rate would increase by 0.8% for the power uprate, however the hydrogen production rate increase is bounded by conservatisms inherent in the radiolysis production term as cited in Standard Review Plan (SRP) 6.2.5, Appendix A.

3.11.3 Equipment Qualification Environments

The analysis of accident environments for equipment qualification is evaluated in two parts: LOCA and main steam line break inside containment; and high-energy lines outside containment.

3.11.3.1 LOCA and Main Steam Line Break Inside Containment

As stated in Section 3.11.2, the current containment LOCA and main steam line break analyses will not be affected by uprate conditions. The current equipment qualification accident environmental conditions inside containment bound the environmental conditions resulting from the power uprate.

3.11.3.2 High-Energy Line Breaks Outside Containment

The post-accident thermal environmental parameters were generated from computer models of the building structures that calculate the environment created by mass and energy releases during postulated pipe breaks. The mass-energy release is dependent on line pressure, enthalpy, and system inventory. The USAR identifies room heat and pressurization for HELBs in the Main Steam, Main Steam Supply to the Auxiliary Feed Pump Turbine, Main Feedwater, Steam Generator Blowdown, and Auxiliary Steam Systems. A review of the line pressure, enthalpy and system inventories used in these calculations shows that the current calculated environments bound power uprate.

3.11.3.3 Normal Environment Outside Containment

The normal environments for the plant buildings were assessed. The power uprate has an insignificant effect on process fluid temperatures in the auxiliary buildings. With the exception of the main feedwater, the increase in the heat loads is caused by the increase in the decay heat load as it is transferred to the Component Cooling Water system and Service Water system. The increase in these system temperatures has been evaluated and found to have an insignificant impact. Small changes in fluid temperatures have an insignificant affect on the area temperatures. Similar conclusions were reached following the evaluations of the normal environmental conditions in the containment building.

3.12 Radiological Consequences

3.12.1 Normal Operation Analyses

3.12.1.1 Radiation Source Terms

The impact of the power uprate on the normal radiological source terms is addressed in Sections 3.12.1.2, 3.12.1.3, 3.12.1.4 and 3.12.1.5 below.

3.12.1.2 Gaseous and Liquid Releases

The assumed offsite doses (10CFR50, Appendix I) resulting from the liquid and gaseous effluent releases, are conservatively based on 0.1% failed fuel. DBNPS Technical Specification Sections 3/4.4.8, Specific Activity, and 3/4.4.6.2, Operational Leakage, limit the primary activity and primary to secondary leakage, respectively. DBNPS Technical Specification 6.8.4.d, Administrative Controls – Radioactive Effluent Controls Program, also provides requirements for maintaining the doses to the members of the public from radioactive effluents as low as reasonably achievable. These requirements are implemented in the Offsite Dose Calculation Manual (ODCM) and station procedures. Therefore, for the power uprate, the offsite doses from normal effluent releases will remain significantly below bounding limits of 10CFR50 Appendix I.

Previous Annual Radioactive Effluent Discharge Reports demonstrate that the actual releases from the plant are historically a very small percentage of the allowable limits. Based on the most recent report, which evaluated airborne, aquatic, and terrestrial samples to determine radiological impacts from operation of the DBNPS, no significant radiological consequences can be attributed to the DBNPS.

3.12.1.3 Shielding

The DBNPS gamma and neutron shielding designs and radiation area locations are based on the source terms in USAR Chapters 11 & 12. These source terms have been determined to bound the source terms for the power uprate. Radiation exposures to in-plant personnel will continue to be controlled under the site ALARA program.

The post-accident radiation doses within the plant currently evaluated in the USAR remain bounding. Thus, no changes are required for the post-accident vital area doses and the Post-Accident Sampling System.

3.12.1.4 Gaseous, Liquid, and Solid Radwaste Systems

Gaseous waste is processed in the Waste Gas Decay Tanks prior to being discharged. Each of the Waste Gas Decay Tanks is sized to accommodate an entire fuel cycle of waste production with 10% margin. Since the Waste Gas Decay Tanks normally accumulate 0-3 months radioactive waste gas and the maximum increase in waste gas effluents is approximately proportional to the power uprate, sufficient margin exists in the Waste Gas Decay Tank capacity to accommodate the increase in power.

The solid waste management and liquid waste processing systems are designed to control, collect, process, store, and dispose of radioactive wastes due to normal operation including anticipated operational transients. Operation of these systems is primarily influenced by the volume of waste processed, which is not expected to change as a result of the uprate condition.

In summary, the power uprate has no significant effect on any of the waste subsystems or components of these subsystems. Because these systems are typically operated in batch mode, the only potential effect is a slight increase in the frequency at which the batches may be processed. These systems continue to meet the current design basis.

3.12.1.5 Normal Operation Analyses - Summary

Based on the discussions provided above, the proposed power uprate will not cause radiological exposure in excess of the dose criteria (for restricted and unrestricted access) provided in the current 10CFR20. From an operations perspective, radiation levels in most areas of the plant are expected to increase no more than the percentage increase in power level. Individual worker exposures will be maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Gaseous and Liquid Effluent releases are also expected to increase by no more than the percentage increase in power level. Offsite release concentrations and doses will be maintained within the limits of the current 10CFR20 and 10CFR50, Appendix I by the site radioactive effluent control program.

3.12.2 Accident Analyses

The radiological accident analyses are based on 102% of 2772 MWt (2827 MWt). Thus, the power uprate, in combination with more accurate measurement of the thermal power level, yields the same maximum thermal power that is currently the basis for source term and accident analyses. Consequently, the current source term and radiological analyses will remain applicable to for the power uprate in conjunction with the more accurate Caldon LEFM flow meters.

3.12.3 Equipment Qualification (EQ)

The environmental radiation levels for both normal operation and accident conditions were originally developed using assumed power levels that envelope the uprated condition.

For the accident contribution, margins were incorporated into the equipment specifications that met or exceeded the requirements of IEEE-323-1974. Generally, postulated radiation doses impacting equipment qualification depend primarily on post-accident contributions. However, normal operating dose rate contributions are included in the design basis calculations. These normal operating contributions are, in all cases, based on source terms, which are bounding for the power uprate. Therefore, regarding cases where normal operating equipment qualification dose rate contributions may be significant, it can safely be concluded that the power uprate would not cause dose rates or integrated doses to exceed design basis values.

The effects of post-accident radiological consequences on equipment qualification were also evaluated. The source term used in the 24-month cycle analyses was based on 2827 MWt (102% of 2772 MWt). It is concluded that this source term remains bounding.

Therefore on the basis of these considerations, it is acceptable from an EQ standpoint to operate at the uprated power.

3.13 NUCLEAR FUEL

This section summarizes the evaluations performed to determine the effect of the power uprate on nuclear fuel performance. The core design and reload safety evaluations are performed for each specific fuel cycle and vary according to the needs and specifications for each cycle. The nuclear fuel review for the power uprate evaluated the fuel and core design, core thermal-hydraulic design, and fuel rod mechanical performance.

3.13.1 Fuel and Core Design

The installation of the Caldon LEFM CheckPlus™ System for the power uprate was evaluated. A DBNPS fuel cycle, typical of current designs and fuel management, was modeled at the uprated power level to evaluate the effects of the power uprate conditions on the fuel and core design key parameters. The results were compared to a DBNPS design without the uprated power level. Since the power uprate is relatively small, the representative cycle is adequate to demonstrate the sensitivity of reload parameters to the power uprate conditions.

The methods and core models used in the uprate analyses are consistent with those presented in the DBNPS USAR. No changes to the nuclear design philosophy, methods, or models are necessary due to the uprate. The core analyses for the uprate were performed primarily to determine if the values previously used for the key safety parameters remain applicable prior to the cycle-specific reload design.

The core analyses show that the implementation of the power uprate will continue to meet the current nuclear design basis documented in the USAR. The impact of the uprate on peaking factors, rod worths, reactivity coefficients, shutdown margin, and kinetics parameters is expected to be either well within normal cycle-to-cycle variation of these values or controlled by the core design, and will be addressed on a cycle-specific basis consistent with current approved reload methodology.

Additionally, evaluations of Critical Boron Concentrations, Shutdown Boron Concentrations, and Refueling Boron Concentrations were performed. The boric acid storage volume for both the Boric Acid Addition System and the Borated Water Storage Tank provides sufficient shutdown boration capability at the uprated power level. Based on the evaluations performed, the power level uprate will have a minimal impact on nuclear licensing core physics parameters.

3.13.2 Core Thermal-Hydraulic Design

The core thermal-hydraulic analyses and evaluations were performed based on a cycle 14 design at an uprated core power level of 2820 MWt, which bounds the expected uprate of 2817 MWt. The analyses assumed that the uprated core design would be composed of a mixed loading of Mark-B10 and Mark-B10K fuel assemblies. The Mark-B10K fuel assemblies feature the M5™ advanced, low corrosion cladding and the Trapper™ debris-resistant lower end fitting. All fuel assemblies in the cycle 14 core have compatible thermal-hydraulic characteristics. These analyses also remain applicable for the DBNPS mixed cores containing Mark-B12 fuel assemblies. The Mark-B12 fuel design, which will be utilized for the Cycle 14 fresh fuel, incorporates an improved instrument guide tube design and a slightly shorter fuel rod length.

The thermal-hydraulic design methods and computer codes used for the power uprate to meet the DNB design basis are consistent with those presented in the DBNPS USAR. The BWC CHF correlation for Mark-B fuel assemblies with Zircaloy-4 (Zr-4) or M5™ grid spacers, BAW-10143P-A, "BWC Correlation of Critical Heat Flux," is used for the power uprated thermal-hydraulic core protection evaluations. No changes to the thermal-hydraulic design philosophy, methods, or models are necessary due to the power uprate. The results show that the uprated core will meet all required thermal-hydraulic core protection requirements.

3.13.3 Fuel Rod Mechanical Performance

The design analysis for fuel rod cladding corrosion was reviewed to assess the impact of the power uprate. Fuel rod cladding corrosion is adversely affected by increases in coolant temperature. Limited corrosion margin exists in the current DBNPS operating analyses for Zr-4 cladding. Use of the M5™ fuel rod cladding in the core limits the Zr-4 clad fuel to third-burn fuel. A preliminary evaluation of the fuel rod cladding corrosion was performed under the uprated conditions. The results show that all the fuel in the core will continue to meet the acceptance criteria of less than or equal to 100 microns. The fuel rod cladding corrosion and all other fuel rod design evaluations will continue to be performed on a cycle-by-cycle basis using actual rod power histories. The results from these evaluations are expected to be within normal cycle-to-cycle variation of these parameters or controlled by the core design, and will be addressed on a cycle-specific basis consistent with current reload methodology.

Figure 3-1
LOCA Containment Pressure vs. Time

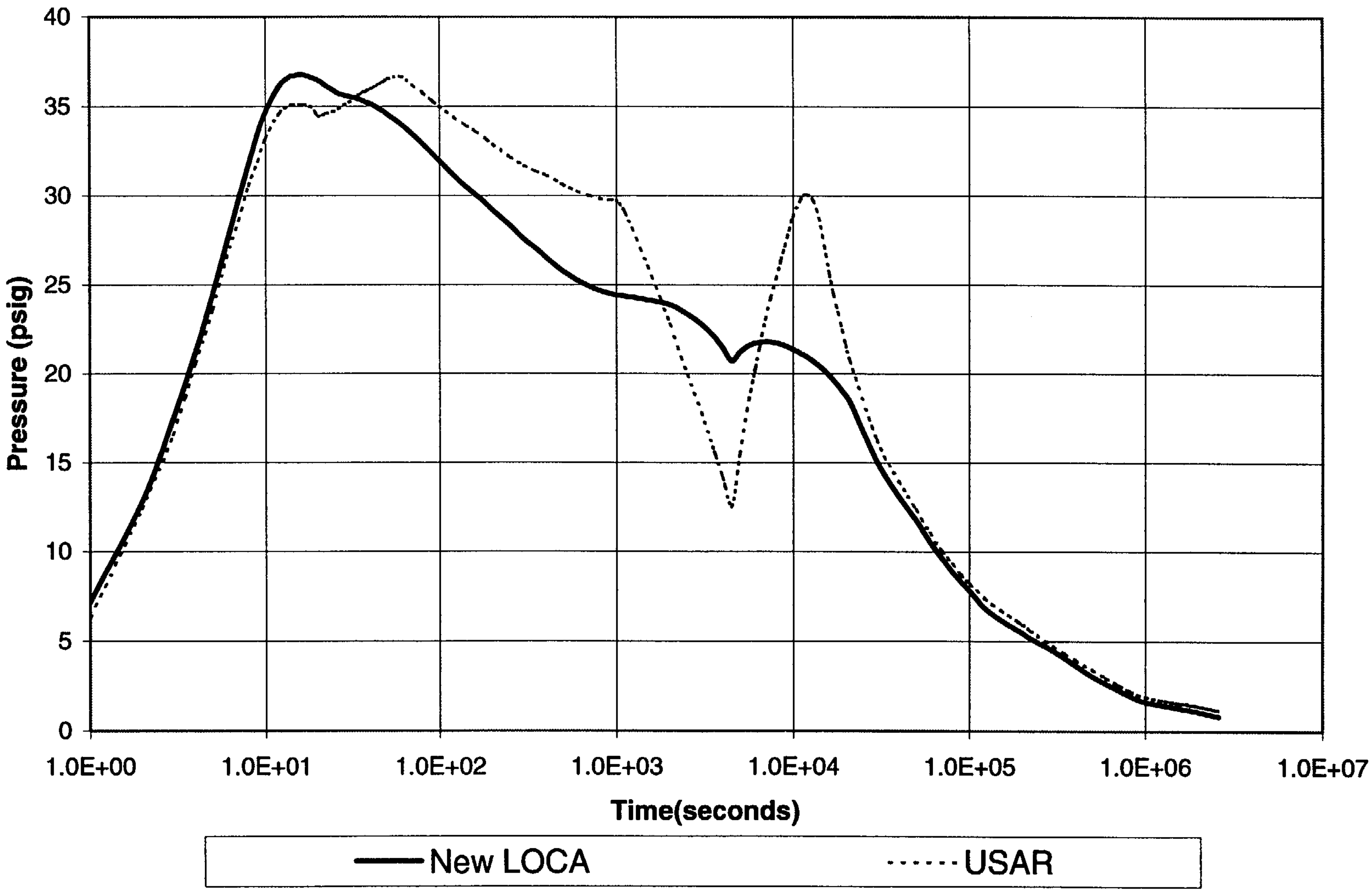
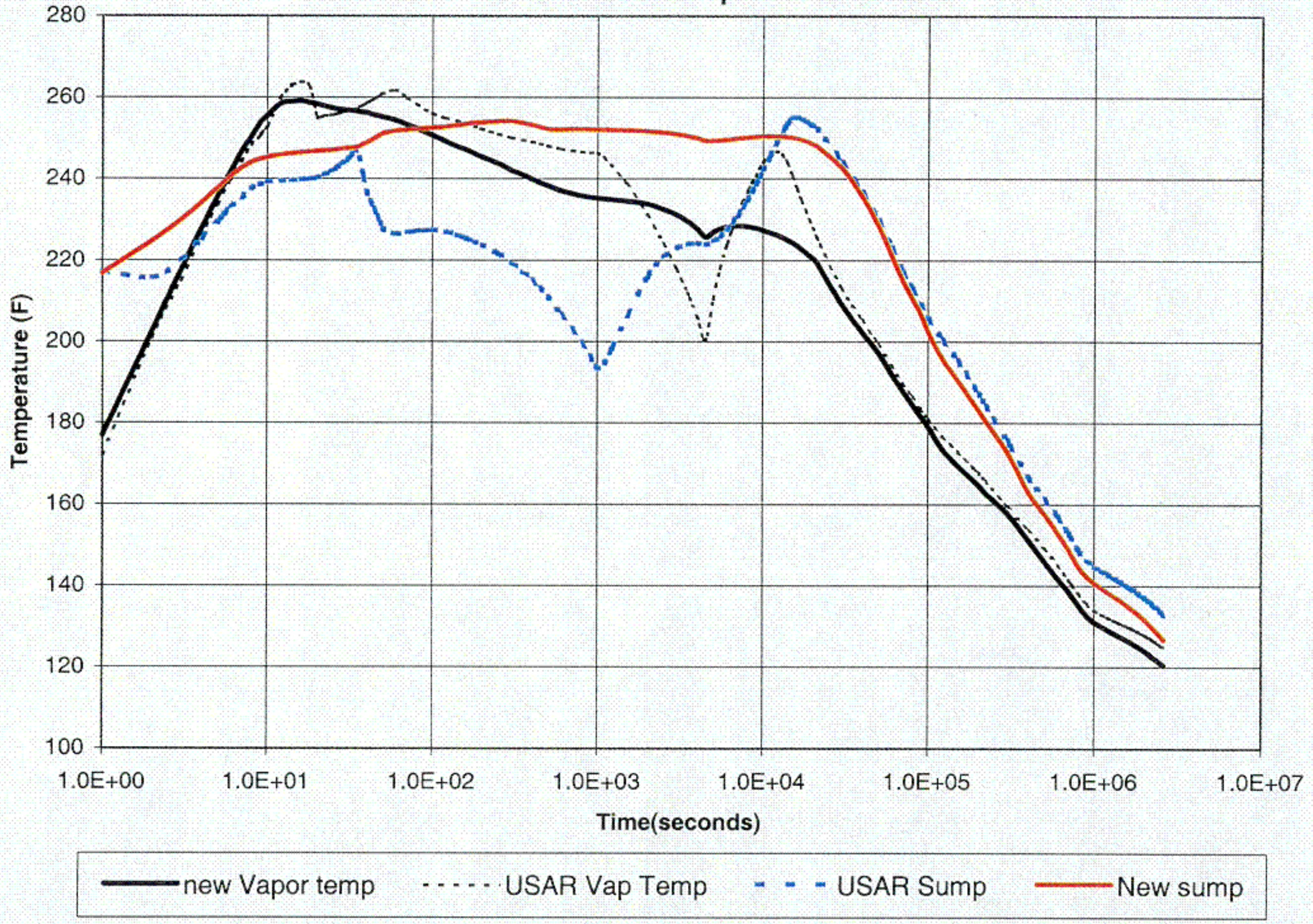


Figure 3-2
LOCA Containment Temperature vs. Time



C01

Figure 3-3
MSLB Containment Pressure vs. Time

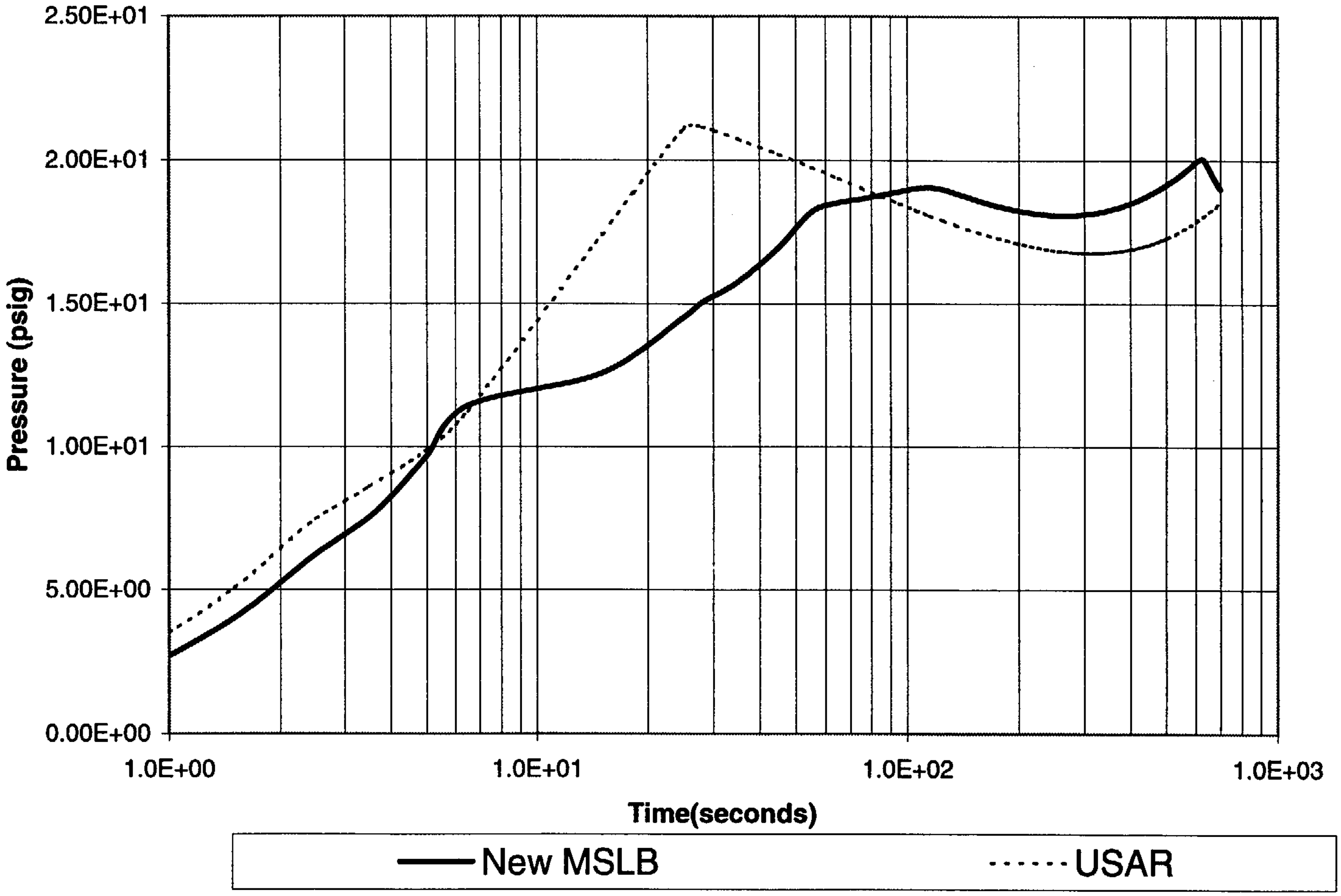
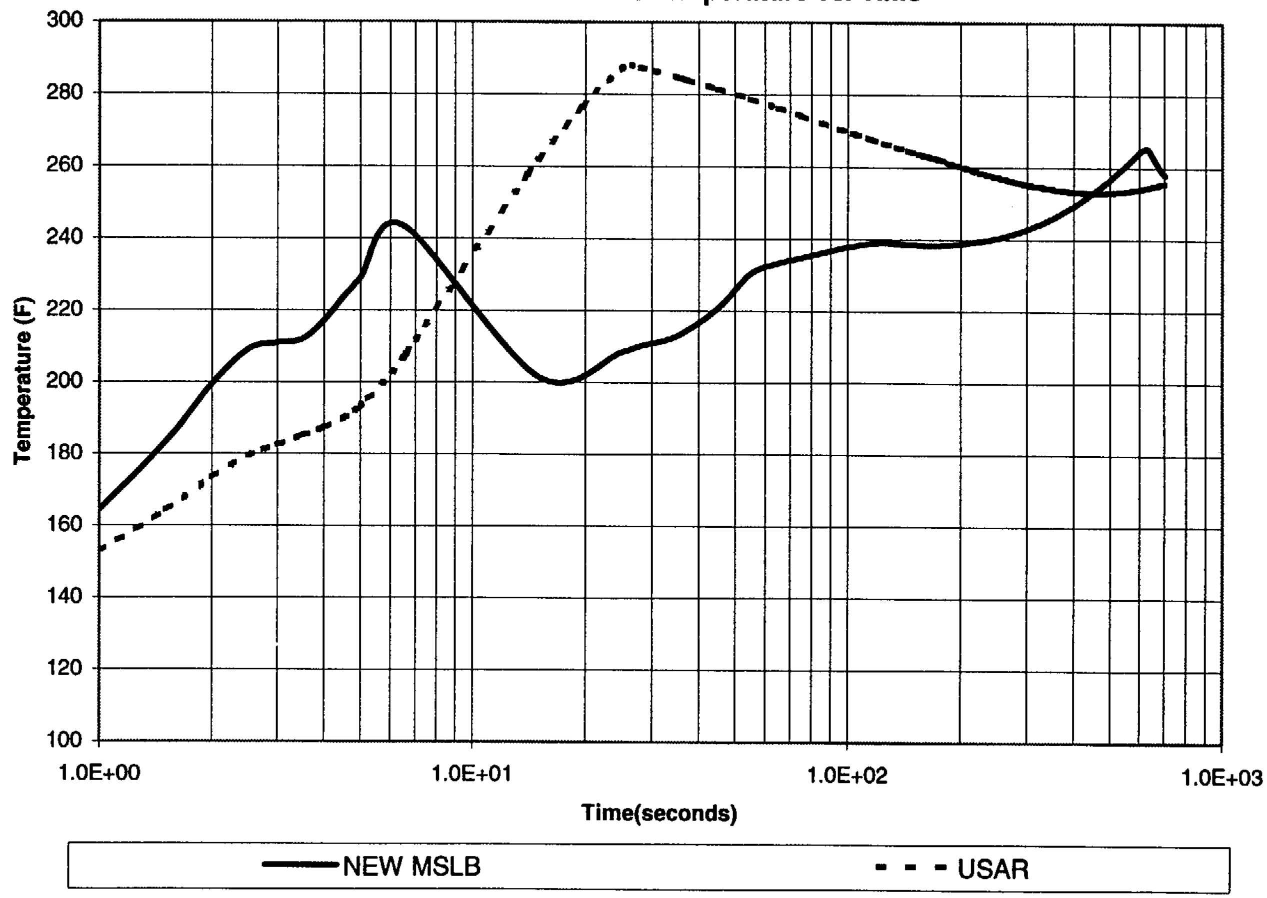


Figure 3-4
MSLB Containment Temperature vs. Time



4.0 OTHER ISSUES

4.1 MISCELLANEOUS PROGRAMS

The power uprate has the potential to affect programs that are developed and implemented by station personnel to demonstrate that topical areas comply with various design and licensing requirements. The plant programs and/or issues listed in Table 4-1 were reviewed to determine the impact due to the power uprate. In addition to the programs, plant Technical Specifications address specific requirements for a number of programs. These programs are identified in Table 4-2.

For the programs listed in Table 4-1, the controlling procedures and processes for the programs and key reference items within the procedures were reviewed. Program sponsors, implementing organization personnel, and other cognizant individuals were interviewed for those issues and programs that would be impacted by the uprate. Based upon the review of this information, the extent of impact by the implementation of the power uprate was determined for the various issues and programs.

For the programs listed in Table 4-2, the Technical Specifications and Technical Requirements Manual Sections associated with the programs were reviewed to identify any areas affected by power uprate.

The review process resulted in two groupings: not affected; and affected but changes would be captured by in-place processes and procedures such that the power uprate information would be incorporated into the affected programs. The results of the review identified three programs that would be impacted by the uprate. However, changes to these programs will be captured by in-place change procedures as identified below.

4.1.1 Simulator

The DBNPS-specific simulator mimics the actual control room and is primarily used for training of operations personnel. In addition to the overall physical likeness between the actual control room and the simulator, computer systems provide simulator responses that are intended to match actual plant conditions for the simulation of accidents and transients, to the greatest extent possible. Simulator changes resulting from the power uprate will mimic the control room changes by adding an annunciator window and LEFM panel. Simulator changes will be implemented as part of the plant modification.

A review of the training simulator fidelity with the new power rating will be included at the next regularly scheduled review following the uprate. Simulator revalidation is performed in accordance with ANSI/ANS 3.5-1985.

4.1.2 Fire Protection/Appendix R

The impact on the Appendix R evaluations consists of slight change in the time estimates of operator response times for initiation of AFW, MU&P/HPI, CAC, and closing of the PORV.

These slight changes do not impact the conclusion of the evaluation regarding operator response capability or system availability. The Appendix R calculations to determine the time constrains for AFW loss and MU/HPI loss are based on initial reactor power levels. These calculations were performed assuming a power level of 102% of licensed power, which bounds the proposed power uprate.

The calculation for safe shutdown with one CAC cooler is based on reactor coolant temperature and pressure. The temperature assumed, 608 °F, bounds the T_{hot} for the power uprate, and the power uprate does not result in a change to reactor pressure.

The calculation to estimate the minimum time to close the PORV after an Appendix R accident (control room fire with LOFW) to ensure that the MU pump does not run out used the 1979 ANS infinite irradiation decay heat term and compared it an actual Cycle 2 EOC calculation. A comparison of these two decay heat curves show that the infinite irradiation curve has a margin of the order of 15% at 0.2 hour and 33% at 2.0 hours. Therefore, the increase in power is bounded.

An estimate of the time and water volume required for RCS cooldown to DHR cut-in temperature (280°F) was performed. The assumed power level used for determining the decay heat could not be identified. Assuming that the decay heat is based on 100% of licensed thermal power, the required water volume would be increased by 1.7% and the time to reach DHR cut-in temperature increased accordingly. The increase in required water volume would slightly decrease the time when the CST is depleted. When the CST is depleted, the Service Water system would be utilized for makeup water. The current time frames for CST depletion is 1.5 to 4 days depending on the assumed quantity of water and relief valve capacity. Even with this reduction, sufficient time is available for action to establish the service water as makeup.

4.1.3 Corrosion/Erosion Monitoring And Analysis Program (CEMAP)

The main feedwater systems, as well as other power conversion systems, are important to safe operation. Failures of passive components in these systems, such as piping can result in undesirable challenges to plant safety systems required for safe shutdown and accident mitigation. Failure of high-energy piping, such as feedwater system piping, can result in complex challenges to operating staff and the plant because of potential system interactions of high-energy steam and water with other systems, such as electrical distribution, fire protection, and security. The DBNPS adheres to criteria, codes and standards for high-energy piping systems as described in the licensing basis. Piping will be maintained within allowable thickness values.

Flow-Accelerated Corrosion (FAC), in the piping systems at the DBNPS, is modeled using the CHECWORKS computer program. CHECWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current Corrosion/Erosion Monitoring and Analysis Program (CEMAP).

4.2 OPERATING PROCEDURES (ABNORMAL/NORMAL) AND OPERATOR ACTIONS

The power uprate is not expected to have any significant effect on the manner in which the operators control the plant, including operator response times described in the USAR and modeled in the individual plant examination (IPE), either during normal operations or transient conditions. The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Steam Generator Pressure and Main Feedwater and Steam flows. Changes associated with the power uprate will be treated in the same manner consistent with any other plant modification, and will be included in Operator Training accordingly.

A new annunciator will be installed to indicate Caldon flowmeter system trouble. No other changes to control room annunciators, controls and displays are required as a direct result of the power uprate. When the power uprate is implemented, the Nuclear Instrumentation System will simply be adjusted to indicate the new 100% RTP in accordance with Technical Specification requirements and plant administrative controls.

The plant computer system will provide an audible alarm for LEFM CheckPlus™ failure or if maintenance is required. As discussed in Section 1.0, the DBNPS USAR TRM will be updated to address the requirements to be followed should the LEFM CheckPlus™ system become unavailable.

There are no new operator tasks required for safe shutdown due to the proposed uprate.

The only change in the alarms for the Safety Parameter Display System (SPDS) is the high reactor power level. The current alarm is set at 107% of 2772 MWt (2966 MWt), which is above the current RPS trip setpoint of 105.1%. The new alarm will be set at 106% of 2817 MWt (2986 MWt), which remains above the proposed new RPS trip setpoint of 104.9%. The power uprate does not impact other alarms in the SPDS. The remaining inputs in the SPDS are provided for trending purposes only. Ranges of the measuring instruments provided for SPDS are not exceeded due to power uprate.

4.3 STATION BLACKOUT EVENT

A separate and dedicated alternate AC power source, the Station Blackout Diesel Generator (SBODG) is available to supply systems required for coping with a station blackout as defined by 10CFR50.2. The full-size SBODG was installed in response to 10CFR 50.63. However, as described in the USAR, the SBODG is not specifically credited for event mitigation. With an alternate source of power available, the intent of the analysis is to demonstrate that the plant will evolve to a stable condition on safety grade equipment and without reliance on operator action.

The USAR analyses assume that the RCS pressure boundary remains intact. A reactor trip occurs, initiated by SFRCS. The steam-driven auxiliary feedwater pump is automatically started. There is very little increase in residual decay heat associated with the small increase in rated power. This conclusion is based on the loss of main feedwater accident analysis, which was initiated from 102% power and included the additional heat generated by operating the reactor coolant pumps. The analysis demonstrates that the plant will evolve to stable single-phase natural circulation and core cooling will be ensured. The RCS temperature will be maintained within a few degrees of the SG sink temperature, at the saturation temperature corresponding to the lift pressure of the lowest set of main steam safety valves.

Once AC power is restored, operator action will be sufficient to reduce the RCS temperature to the DHR cut-in conditions, where the cooldown can continue to ambient conditions. Fuel damage will not occur and the RCS pressure will be maintained within acceptable limits.

4.4 GENERIC LETTERS 89-10/96-05, 95-07 AND 96-06

4.4.1 Generic Letters 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance," and 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves"

As a result of the power uprate, there are no required changes to the DBNPS GL 89-10/96-05 MOV Program. Design basis differential pressures developed from conservative assumptions are used for MOV sizing requirements. These conditions bound uprate conditions and do not compromise margin of safety.

4.4.2 Generic Letter 95-07 "Pressure Locking and Thermal Binding of Safety Related Operated Gate Valves"

The pressurizer spray isolation valve, pressurizer pilot-operated relief valve, letdown cooler stop valves, feedwater block valves, RCS to Decay Heat Removal isolation valves, and decay heat pump to HPI pump suction isolation valves, were previously modified to address pressure locking or thermal binding, or both. These modifications were performed to address valve and plant reliability concerns.

A review of the evaluation of the GL 95-07 issue was performed to determine if the proposed power increase would adversely affect any conclusion related to pressure locking or thermal binding. The ambient conditions during normal operation are not impacted in the Containment, Auxiliary Building, or Turbine Building. The current post-LOCA and HELB conditions are bounding for the power uprate.

The proposed power uprate does not introduce any increased challenge for thermal binding and/or pressure locking, and the responses and conclusions of GL 95-07 remain bounding.

4.4.3 Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions"

Generic Letter GL 96-06 addresses the overpressurization of isolated piping segments.

The isolated segments of pipe have been previously evaluated and have been upgraded, where required, to meet the criteria of the GL 96-06. Isolated segments of piping that are susceptible to overpressurization, due to thermal stresses imposed by the environment, have been provided with thermal safety-relief valves, provided with bypass check valves, determined to have inherent relief capability, partially drained after use to prevent overpressurization, or have been determined to be structurally adequate to withstand the stresses imposed by the thermal loading. LOCA analyses that affect the containment side of the isolated segment of piping have been determined to be bounding for the proposed power uprate.

The environmental conditions, imposed by LOCAs or secondary side line breaks, or normal operating conditions, have not been affected in an adverse manner that would compromise the piping that is capable of being isolated by segments.

There is no increase in the possibility of overpressurization of isolated segments of piping and the responses and conclusions of G L96-06 remain valid.

4.5 INDEPENDENT PLANT EXAMINATION (IPE)

The DBNPS Probabilistic Risk Assessment (PRA) models both a Level 1 Core Damage Frequency (CDF) analysis, a Level 2 Large Early Release Frequency (LERF) analysis, and a Level 3 Offsite Consequences analysis.

The success criteria for the Level 1 analyses were derived primarily from the USAR analyses, and as such were already analyzed using a 102-percent core power level (2827 MWt). Since the proposed uprate is based on reducing the 2-percent margin for power measurement uncertainty that has been used typically in the USAR, the Level 1 analysis remain bounded by the uprated power conditions.

Some success criteria derivations used for the Level 1 and Level 3 analysis were performed using the Modular Accident Analysis Program (MAAP) with a nominal core power level of 2772 MWt. However, the Level 2 analyses based on MAAP are only expected to have minor timing impacts from the power uprate on the Level 2 containment release analyses and not any significant changes on the release magnitudes. Therefore, the power uprate is not expected to have significant impact on the PRA results.

Table 4-1 Program/Issues

Issues and Programs	Requires Update
Plant Simulator	YES
Fire Protection/Appendix R	NO
Check Valves	NO
Motor-Operated Valve Administrative Program (GL 89-10)	NO
Air-Operated Valves	NO
Service Water System Control and Monitoring (GL 89-13)	NO
Inservice Inspection Program	NO
Inservice Test Program	NO
Containment Integrity (Appendix J)	NO
Equipment Qualification	NO
Human Factors	NO
Station Blackout	NO
Anticipated Transient Without Scram	NO
Flow-Accelerated Corrosion Program	YES

No - Programs not impacted by uprate change or are bounded by existing analysis.
Yes - Programs impacted and changes to be addressed in uprate implementation.

Table 4-2 Technical Specification Programs

Program	Requires Update
Primary Coolant Sources Outside Containment Program (TS 6.8.4.a)	NO
In Plant Radiation Monitoring Program (TS 6.8.4.b)	NO
Post-Accident Sampling Program (TS 6.8.4.c)	NO
Radioactive Effluent Controls Program (TS 6.8.4.d)	NO
Radiological Environmental Monitoring Program (TS 6.8.4.e)	NO
Ventilation Filter Testing Program (TS 6.8.4.f)	NO
Process Control Program (TS 6.14)	NO
Containment Leakage Rate Testing Program (TS 6.16)	NO

Docket Number 50-346
License Number NPF-3
Serial Number 2692
Enclosure 1
Attachment 4

Not Used

Docket Number 50-346
License Number NPF-3
Serial Number 2692
Enclosure 1
Attachment 5

Not Used

Docket Number 50-346
License Number NPF-3
Serial Number 2692
Enclosure 1
Attachment 6

Not Used

Docket Number 50-346
License Number NPF-3
Serial Number 2692
Enclosure 1
Attachment 7

FRAMATOME ANP AFFIDAVIT PURSUANT TO 10 CFR 2.790

(3 pages follow)

AFFIDAVIT

STATE VIRGINIA

1. My name is James F. Mallay. I am Director, Regulatory Affairs, for Framatome ANP ("FRA-ANP"), and as such I am authorized to execute this Affidavit.

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

3. I am familiar with the FRA-ANP information included in the report, "Davis Besse Heat Balance Uncertainty Calculation Appendix K," 32-5012428-00, and referred to herein as "Document." Information contained in this Document has been classified by FRA-ANP as proprietary in accordance with the policies established by FRA-ANP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by FRA-ANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in the Document be withheld from public disclosure.

6. The following criteria are customarily applied by FRA-ANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FRA-ANP'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 FRA-ANP.

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

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

7. In accordance with FRA-ANP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside FRA-ANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FRA-ANP 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.

James Malloy

SUBSCRIBED before me this 3rd

day of August, 2001.

Brenda C. Maddox

NOTARY PUBLIC, STATE OF VIRGINIA

My commission expires July 31, 2003.
I was commissioned a notary public
as Brenda C. Cardona.

APPLICABILITY OF COMANCHE PEAK RAI QUESTIONS TO PROPOSED DAVIS-BESSE POWER UPRATE

For the proposed power uprate, the Davis-Besse Nuclear Power Station (DBNPS) has taken into consideration the Requests for Additional Information (RAIs) made by the NRC staff in their review of a similar power uprate license amendment application for the Comanche Peak Steam Electric Station (CPSES) Units 1 and 2 (Amendment No. 72 to Facility Operating License NPF-87, and Amendment No. 72 to Facility Operating License NPF-89, dated September 30, 1999). The following CPSES RAI responses were evaluated:

CPSES Log #	Date
TXX-99105	April 23, 1999
TXX-99115	May 14, 1999
TXX-99164	July 9, 1999
TXX-99195	August 13, 1999
TXX-99203	August 25, 1999

In addition, the DBNPS has taken into consideration the RAI made by the NRC staff in their review of Caldon Inc. Topical Report ER-80P. Selected questions from the following CPSES RAI response were evaluated:

CPSES Log #	Date
TXX-98274	December 17, 1998

The following includes the questions that were addressed by the CPSES in the above-listed letters, and a DBNPS-specific response for each question.

Question 1 (TXX-99105):

Provide a discussion that addresses the impact of the proposed power uprate on the load, voltage, and short circuit values for all levels of the station auxiliary electrical distribution system. Include in this discussion any impact on the direct current power systems.

Response:

Refer to Section 3.9 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2 (TXX-99105):

For the power uprated conditions, discuss environmental qualification for the safety related electrical equipment located in harsh environmental areas. For this safety-related electrical equipment, address the continued environmental qualification and the process for establishing qualification for any increased temperature, pressure, humidity, and radiation values.

Response:

Refer to Sections 3.11.3 and 3.12.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 3 (TXX-99105):

Discuss and verify the assumptions for the station blackout analysis are valid for the power uprate conditions, particularly as they relate to issues such as the heat-up analysis, equipment operability, and battery capacity.

Response:

Refer to Section 4.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 4 (TXX-99105):

Provide a discussion addressing the impact of the CPSES Unit 2 power uprate on the turbine/generator, isophase bus, main transformers, and switchyards. Address in detail any non-hardware changes for these items as a result of the CPSES Unit 2 power uprate.

Response:

Refer to Section 3.9 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 5 (TXX-99105):

Discuss the impact of the CPSES Unit 2 power uprate electrical conditions on the current grid stability and reliability analysis. Describe in this discussion, how the station continues to be in conformance with General Design Criterion 17 with CPSES Unit 2 at the power uprated electrical conditions.

Response:

Refer to Section 3.9.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 6 (TXX-99105):

Provide a pressurized thermal shock evaluation for the CPSES Unit 2 reactor vessel before implementing the power uprate and after implementing the power uprate.

Response:

Refer to Section 3.6.2.5 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 7 (TXX-99105):

What is the calculated end-of-life fluence in the current vessel design of CPSES Unit 2? What is the expected fluence for pressurized thermal shock with the revised design conditions/power uprate for CPSES Unit 2?

Response:

Refer to Section 3.6.2.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 8 (TXX-99105):

Does the power uprate for CPSES Unit 2 change the cold leg temperature? If so, please provide details.

Response:

The power uprate will result in a 0.4°F decrease in the cold leg temperature. The new cold leg temperature will remain bounded by the design cold leg temperature as specified in the

RCS Functional Specification. The expected cold leg temperatures remain within the range assumed in the development of the equations and tables which form the bases for evaluating the neutron irradiation effects on vessel integrity. Refer to Section 3.3 and Table 3-1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 9 (TXX-99105):

Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

Response:

Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 10 (TXX-99105):

Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

Response:

The power uprate represents a small increase in the rated core thermal power level. A review of the USAR accident analyses confirms that the acceptance criteria for each event will not be violated. Furthermore, no additional operator actions will be required for mitigation of the accidents. The power uprate is not expected to have any significant effect on the manner in which the operators control the plant, including operator response times described in the USAR and modeled in the individual plant examination(IPE), either during normal operations or transient conditions. There are no operator actions that are being automated as a result of the uprate. Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 11 (TXX-99105):

Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-of-tolerance range)? If changes will occur, discuss how they will be addressed.

Response:

Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 12 (TXX-99105):

Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

Response:

Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 13 (TXX-99105):

Describe all changes the power uprate will have on the operator training program and the plant simulator. Provide a copy of the post-modification test report (or test abstracts) to document and support the effectiveness of simulator changes as required by American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.5-1985, Section 5.4.1.

Specifically, please propose a license condition and/or commitment that stipulates the following:

(a) Provide classroom and simulator training on all changes that effect operator performance caused by the power uprate modification.

(b) Complete simulator changes that are consistent with ANSI/ANS 3.5-1985. Simulator fidelity will be re-validated in accordance with ANSI/ANS 3.5-1985, Section 5.4.1, "Simulator Performance Testing." Simulator revalidation will include comparison of individual simulated systems and components, and simulated integrated plant steady state and transient performance with reference plant responses using similar startup test procedures.

(c) Complete all control room and plant process computer system changes as a result of the power uprate.

(d) Modify operator training and the plant simulator, as required, to address all related issues and discrepancies that are identified during the startup testing program.

Response:

Refer to Sections 4.1.1 and 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 14 (TXX-99105):

The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurement and plant operation.

Response :

As described in Section 1.0 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, new procedures for maintenance and calibration of the LEFM CheckPlus™ system will be developed for the DBNPS based on the vendor's recommendations. In addition, the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM) will be updated to address the requirements to be followed should the LEFM Check Plus system become unavailable. Additional detail is provided below in the later Response to Question 2 (TXX-99203).

Question 15 (TXX-99105):

For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the installed installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

Response:

The LEFM system has not yet been installed at the DBNPS, therefore this question is not applicable to the DBNPS.

Question 16 (TXX-99105):

The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

Response:

The methodology used to calculate the combined feedwater mass flow and feedwater temperature uncertainty for the improved LEFM is exactly the same as the methodology presented in Caldon Topical Report ER-80P. This value is then utilized to calculate the total power measurement uncertainty described in and further elaborated for the LEFM CheckPlus™ system configuration in Caldon Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System."

Question 17 (TXX-99105):

Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter factors not representative of the plant specific installation), should provide additional justification for use. This justification should show that the meter installation is either independent of the plant specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response:

As described in Section 3.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, the LEFM CheckPlus™ system to be installed at the DBNPS will be calibrated to a site-specific piping configuration prior to installation. The results of the calibration will provide a meter factor representative of the plant specific configuration. In addition, the accuracy with which the meter factor is determined will be incorporated into the uncertainty analysis of record. Therefore, additional justification for use will not be required.

Question 18 (TXX-99105):

Based on the above, the staff finds that feedwater flow measurement using the LEFM can provide a thermal power measurement that will remain bounding within an uncertainty of 1% of rated thermal power. This is premised on the assumption that no additional uncertainties beyond those included in Topical Report ER-80P are assumed to be included in the 10 CFR Part 50, Appendix K 102% thermal power margin requirement.

Response:

Refer to Framatome ANP Document 32-5012428-00, "Davis-Besse Heat Balance Uncertainty Calc. – App. K Uprate," (Enclosure 1 Attachment 8 of the DBNPS license amendment application) for a discussion of the power measurement uncertainty calculation.

Question 19 (TXX-99105):

The amendment request proposes to reduce the margin for assumed power level for non-LOCA accident and transient analysis on the same basis as the proposed exemption to the Appendix K ECCS evaluation requirement. Staff consideration of the related Appendix K exemption request was in part based on the premise that the power level requirement is one of several conservative features that, taken together, provide substantial conservatism in ECCS analyses.

Justify the proposed margin reduction for non-LOCA analyses that currently assume 102% power. The justification should include a quantitative or qualitative discussion of conservative analysis assumptions for the non-LOCA accidents and transients and the safety margin they provide relative to the power level margin assumption.

Response:

An evaluation of accident analyses is provided in Sections 3.10 and 3.11 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. The typical non-LOCA analyses contain several other conservatisms in addition to the heat balance error. These may include a conservative selection of the reactivity parameters, but they also consider conservative trip setpoints, valve actuation setpoints lift tolerances, time delays, and flow reductions for the equipment used for mitigation. The non-LOCA analyses impose a penalty of the highest worth control rod being stuck out of the core. The minimum tripped rod worth is also reduced to preserve a 1% shutdown margin at hot zero power, even though additional worth is available. Even with a higher accuracy measurement of the feedwater flow, the analyses will still impose a heat balance error on the calculations. Coupled with conservative initial and boundary conditions, the method of analysis may provide additional conservatism. While the effect of one conservative component of the analysis is reduced, the combination of all other constraints on the analyses will still produce a conservative calculation.

Question 20 (TXX-99105):

Increasing licensed power level would result in an increased heat source that could affect the progression of certain accidents. Discuss the potential impact of plant operation at the higher proposed power level on ATWS progression, containment integrity analyses, and on overall IPE results.

Response:

The proposed small power increase is not sufficient to materially affect the progression of any event. Refer to Sections 3.10.3.18, 3.11.2, and 4.5 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, respectively, for a discussion of the impact of the proposed uprate on ATWS, containment integrity analysis, and overall IPE results.

Question 21 (TXX-99105):

Discuss the impact on LOCA and non-LOCA analysis results (e.g., main steam line break) of the revised values for RCP heat addition and RCS flow rate included in the amendment request.

Response:

For the DBNPS license amendment application, no change is required for either RCP heat addition or minimum RCS flow. The slight change in RCS flow rate is solely due to temperature (density) changes caused by the power uprate. The minimum required RCS flow rate to protect against DNB is not changing.

Question 22 (TXX-99105):

Provide the detailed calculational basis to substantiate the statement made in the amendment request that a 10-percent SG tube plugging level supports a peak plugging level of 15% in any one SG, provided that the average level of plugging of all four SGs is no greater than 10 percent. Explain the difference between the plugging level used in the analysis discussed in the amendment request and the plugging level assumed in the current LOCA analysis.

Response:

Steam generator tube plugging has been taken into account by the safety, LOCA, and fuel design analyses. For the safety analyses, 10% tube plugging has been analyzed. For LOCA analyses, 20% tube plugging has been analyzed. For fuel design analyses, an RCS flow that corresponds to 8.4% tube plugging has been used as input.

As discussed in Section 3.3 and Table 3-1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, for the power uprate, operating conditions for 0% and 20% tube plugging were defined. The limiting conditions were then used to structurally qualify the equipment for the uprate. The approved tube plugging limit for the DBNPS is 1300 equivalent tubes per steam generator (or 8.4% of the tube population). Additional analyses and evaluations would be required to increase this limit.

No change to the allowed SG plugging limit is included in the proposed license amendment application. For information, the LOCA analyses that have been performed for the DBNPS support an overall plugging level of up to 20 percent in either steam generator. The analyses were also performed to support an operating power level of 2966 MWt, which bounds the proposed uprate.

Question 23 (TXX-99105):

Plant response to SGTR and other events depends on SG atmospheric relief valve operation. Reactor operation at higher power levels may cause these valves to operate more often in the event of certain events, thereby affecting their reliability. Discuss the effects of operation at the proposed new power level on the possible increased challenge to these valves and their expected failure frequency during a SGTR event (and other events requiring their operation).

Response:

Refer to Section 3.7.1.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 24 (TXX-99105):

When considered in terms of core power, the proposed changes in power range neutron flux, and overpower N-16 nominal and allowable reactor power trip levels appear slightly non-conservative. Explain the basis for the proposed revision to the N-16 overpower and power range neutron flux trip set points given in the amendment request. Provide justification for the apparently non-conservative set point changes.

Response:

The N-16 overpower trip setpoint is not applicable to the DBNPS. However, as described in the license amendment application, the high flux setpoint will be reduced for the proposed power uprate.

Question 25 (TXX-99105):

The N-16 overtemperature trip setpoint was not changed in the amendment request, based on the statement that it was previously analyzed at the power level requested in the proposed amendment. Confirm that the other proposed changes to plant parameters such as RCS flow and coolant temperatures do not result in a change to the N-16 overtemperature trip setpoint. Explain how the proposed changes in core flow rate and coolant temperatures affect the calculation of the N-16 overtemperature trip setpoint.

Response:

The N-16 overtemperature trip setpoint is not applicable to the DBNPS.

Question 1 (TXX-99115 - Attachment 3):

a. In Attachment 2 of the submittal, the licensee states that the Balance of Plant (BOP) fluid systems were reviewed for compliance with the Westinghouse Nuclear Steam Supply System (NSSS)/BOP Interface guidelines. How does the power uprate affect the design basis of the following systems: main steam, steam dump system, feedwater and condensate system, and auxiliary feedwater system?

b. In Section C of Attachment 2, the licensee states that design documentation and instrumentation and control setpoint changes are required. Which, if any, of the following systems and items would exceed the design basis: circulating water, turbine plant cooling, spent fuel pool cooling, component cooling, station service water, station blackout, spent fuel storage, HVAC systems, turbine/generator? If any, provide the new limits and explain why the new design basis is acceptable.

c. In Table IV-1 of Attachment 2, "NSSS Revised Design Parameters," the licensee describes three limiting cases. Explain which case(s) was (were) used in the evaluation of the above listed BOP systems and the NSSS/BOP interfaces. If only one was used, explain why it provides conservative results.

Response to Part a:

Refer to Section 3.7 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Response to Part b:

Refer to Sections 3.5.5, 3.8, 3.8.6, 3.8.7, 3.8.8, 3.8.9, 3.8.10, 3.8.12, and 4.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Response to Part c:

Refer to Section 3.8.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2 (TXX-99115 - Attachment 3):

Solid, liquid, and gaseous radioactive waste activity are influenced by the reactor coolant activity, which is a function of the reactor core power. What is the impact of these systems by the increase in power?

Response:

Refer to Section 3.12 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 3 (TXX-99115 - Attachment 3):

Discuss why the current containment analysis remains appropriate for use at power uprate conditions.

Response:

Refer to Section 3.11.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 1 (TXX-99115 - Attachment 6)

In regard to Section B.4 of Attachment 2 to the reference transmittal, provide the maximum-calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of the RPV and internals (such as RPV nozzles, lower and core plates, core barrel, baffle/barrel, control rod drive mechanism, and fuel assembly, etc.), the allowable code limits, the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification. Also, provide an assessment of flow-induced vibration of the reactor internal components due to power uprate.

Response:

The design of the reactor vessel internals components was addressed for flow-induced vibration in Topical Report BAW-10051. A comparative analysis was based on the new operational conditions after the power uprate has been performed. It is concluded that the uprated power operational conditions are bounded by Topical Report BAW-10051. Therefore, the RV internals are structurally adequate for flow-induced vibration.

An evaluation has been performed of the impact of the power uprate on the RV, RV internals, and control rod drive mechanisms (refer to Section 3.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application). This evaluation showed that the temperature changes for the uprated power condition are bounded by those in the RCS Functional Specification and thus are bounded by those used in existing analyses. Therefore, the existing loads remain valid and the stresses and fatigue values remain valid. Thus, existing stress reports remain valid for the uprated power conditions.

The mechanical design of the fuel assemblies is analyzed for fatigue and is verified to meet all current design criteria. The fuel assembly mechanical performance evaluation is documented on a cycle specific basis consistent with current approved reload methodology.

Question 2 (TXX-99115 - Attachment 6):

On page 22 of Attachment 2 to the reference transmittal, provide the methodology and assumptions used for evaluating the reactor coolant piping systems, equipment nozzles, and supports for the increased hot leg and cold leg temperatures, increased dynamic hydraulic forcing functions, and the affected design transients due to the power uprate, as stated in the transmittal. Also, provide the calculated maximum stress, critical locations, allowable stress limits, and the Code and Code edition used in the evaluation for the power uprate.

Response:

For the reactor coolant piping systems, which include nozzles and supports, an evaluation was performed that showed that the new operational conditions after the power uprate are bounded by those used in the existing analyses (refer to Section 3.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application). In addition, the evaluation showed that the hydraulic forcing functions are bounded by those values used in existing analyses. It was not necessary to recalculate stresses and usage factors since the existing stresses and usage factors remain applicable for the uprated power conditions.

Question 3 (TXX-99115 - Attachment 6):

Were the analytical computer codes used in the power uprate evaluation different from those used in the original design-basis analyses? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response:

The mass and energy release rates for the loss of coolant accident and the main steam line break analyses were recalculated using RELAP5/MOD2-B&W. The system responses for the LOCAs and MSLB accidents have historically been generated using CRAFT2 and TRAP2, respectively. The mass and energy release rates were also generated using these NRC-approved computer codes. Framatome ANP has performed numerous benchmarks of RELAP5/MOD2-B&W to plant transient data, scaled and integral systems test data, and by comparing results to existing USAR analyses. The comparisons showed that RELAP5/MOD2 predicts the key phenomena of interest and, given the same conservative initial and boundary conditions, will calculate results similar to those produced by the previous analysis codes.

The NRC has reviewed and approved RELAP5/MOD2 for both LOCA (BAW-10192P-A) and MSLB applications (BAW-10193P-A). Since the analysis of record for the DBNPS is based on RELAP5/MOD2, the mass and energy release rates for the power uprate were also generated with RELAP5/MOD2-B&W.

Question 4 (TXX-99115 - Attachment 6):

In reference to the reactor coolant pump (RCP) structural analysis on page 23 of Attachment 2 to the reference transmittal, you stated that "an analysis was performed to determine the impact of the revised design conditions on the stresses and fatigue usage of the RCP ("CRDM" stated in your report should be "RCP") components and the results indicated that the stress and fatigue usage remain within ASME Code limits." Describe the analysis methodology and assumptions (if any), used for evaluating the RCP. Also provide the maximum-calculated stress and CUF for the RCP, the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

Response:

For the reactor coolant pump, an evaluation (refer to Section 3.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application) was performed that showed that the new operational conditions after the power uprate are bounded by those used in the existing analyses. It was not necessary to recalculate stresses and usage factors since the existing stresses and usage factors remain applicable for the uprated power conditions.

Question 5 (TXX-99115 - Attachment 6):

On page 23 of Attachment 2 to the reference transmittal, provide a comparison of the design parameters (i.e., steam pressure, temperature, primary-to-secondary pressure differential, etc.) and transients for the steam generators (SGs) Model D5 against the power uprate condition. Also, provide the maximum calculated stress and CUF for the critical locations (such as the vessel shell, secondary manway bolts, and nozzles), the allowable code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification. Also, provide an evaluation on the flow-induced vibration of the SG U-bends tubes due to power uprate regarding the analysis methodology, vibration level, computer codes used in the analysis and the calculated cross flow velocity.

Response:

For the once-through steam generators, an evaluation was performed that showed that the new operational conditions after the power uprate are bounded by those used in the existing analyses. It was not necessary to recalculate stresses and usage factors since the existing stresses and usage factors remain applicable for the uprated power conditions. The impact on the OTSG FIV tube analyses is discussed in Section 3.6.7.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 6 (TXX-99115 - Attachment 6):

On page 25 of Attachment 2 to the reference transmittal, you stated that the pressurizer structural evaluation was performed by comparing the key inputs in the current pressurizer stress report with the revised design conditions in Table IV-1, and that the results indicated that the design condition used in the current analysis remain bounding for the revised design conditions. Provide a comparison of the design parameters (i.e., RCS pressure, hot leg temperature, cold leg temperature, temperature differential, etc.), the stratification and cyclic design transients for the CPSES pressurizer against the power uprate condition. Also, provide the maximum calculated stress and CUF at the critical locations (such as surge nozzle, skirt support, spray nozzle, safety and relief nozzle, upper head/upper shell and instrument nozzle) of the pressurizer, the allowable code limits, and the Code and code edition used in the evaluation for the power uprate. If different from the Code of record, provide a justification.

Response:

Refer to Section 3.6.8 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 7 (TXX-99115 - Attachment 6):

Discuss the operability of safety-related mechanical components (i.e., valves and pumps) affected by the power uprate to ensure that the performance specifications and technical specification requirements (e.g., flow rate, close and open times) will be met for the proposed power uprate. Confirm that safety-related motor-operated valves (MOVs) will be capable of performing their intended function(s) following the power uprate including such affected parameters as fluid flow, temperature, pressure and differential pressure, and ambient temperature conditions. Identify mechanical components for which operability at the uprated power level could not be confirmed.

Response:

The 10CFR50 Appendix K-required power level (102%) was included in the design of plant safety-related pumps. Refer to Section 3.5.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for a discussion regarding the power uprate effect on systems for decay heat removal and safety injection.

The flow requirements of the Auxiliary Feedwater Pumps (motor and turbine driven) are not affected by the modest power uprate. The steam generator MSSV setpoints remain the same; therefore, the steam generator pressures at which the equipment is required to pump against are unchanged.

The small increase in power results in minor changes to feedwater/condensate and main steam conditions. The small increase in power does not impact air-operated valves (AOVs) included in the AOV Program due to the fact that the design maximum differential pressures and temperatures bound system conditions after power uprate.

As a result of the proposed power uprate, as discussed in Section 4.4.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, there are no required changes to the DBNPS MOV Program. Design basis differential pressures developed from conservative assumptions are used for MOV sizing requirements. These conditions bound uprate conditions and do not compromise margin of safety.

Question 8 (TXX-99115 - Attachment 6):

(This question has been subdivided in order to provide clearer responses.)

a) In reference to Section C on page 26 of Attachment 2 to the reference transmittal, list the balance-of-plant (BOP) piping systems that were evaluated for the power uprate.

b) Discuss the methodology and assumptions used for evaluating BOP piping, components, and pipe supports, nozzles, penetrations, guides, valves, pumps, heat exchangers and anchorage for pipe supports.

c) Provide the calculated maximum stresses for the critical BOP piping systems, the allowable limits, the Code of record and Code edition used for the power uprate conditions. If different from the Code of record, justify and reconcile the differences.

d) Were the analytical computer codes used in the evaluation different from those used in the original design-basis analysis? If so, identify the new codes and provide justification for using the new codes and state how the codes were qualified for such applications.

Response:

The piping systems evaluated are the power conversion systems identified in Table 3-3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application which experience design changes to normal or maximum service temperatures, and the containment spray and decay heat removal piping from the containment emergency sump.

For all pipe segments the small temperature increase is judged to have an insignificant effect on thermal expansion pipe stresses, pipe displacements, and pipe support, anchor and equipment nozzle loads. The temperature increase will not result in significant reduction of stress allowable or appreciable increase in pipe expansion that would translate into increased pipe stresses and increased support, anchor and equipment nozzle loads. Increase in pipe thermal expansion is judged to be offset by conservatism in pipe modeling techniques, supports gaps and support flexibility.

Pipe stresses will remain within the allowables of the Section III ASME Boiler and Pressure Vessel Code, 1971 edition and ANSI B31.1 Power Piping Code, 1967 edition and its addenda. Refer to Section 3.8.11 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

No analytical computer codes different from those used in original design were used.

Question 9 (TXX-99115 - Attachment 6)

Discuss the potential for flow-induced vibration in the heat exchangers following the power uprate. Provide a summary of evaluation for power uprate effects on the high energy line break analysis, jet impingement and pipe whip loads for the power uprate conditions.

Response:

Flow-induced vibration potential is a function of the shell side flow rates (i.e., flow velocities) in the various NSSS heat exchangers (Letdown Coolers and DHR Coolers). Shell side flow rates in these heat exchangers are not significantly affected by the power uprate. In addition, all of these heat exchangers have been designed to withstand up to 2 times the shell side design flow without encountering damaging tube vibrations. Therefore, flow-induced vibration is not a concern following the power uprate.

The OTSGs were evaluated for flow induced vibration and it was concluded a 15% margin exists in the mass flow rates for the OTSG tubes. Refer to Section 3.6.7.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for further details. Other heat exchangers (i.e., feedwater heaters, main condenser, etc.) on the secondary side are bounded by their current design conditions.

Since, the new operating conditions after the power uprate are bounded by the RCS Functional Specification, there is no effect on HELB locations, jet impingement forces, or pipe whip loads since they are based on the RCS Functional Specification parameters.

BOP heat exchanger flows (feedwater heaters and moisture separator reheaters) are bounded by the design conditions, except as identified in Section 3.8.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

The main condenser tubes have been previously staked to address tube bundle vibration. An approximately 2% increase in steam flow will not impact the main condenser vibration.

The high energy line breaks on the secondary side of the plant are discussed in Sections 3.8.11 and 3.11.3.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 1 (TXX-99115 - Attachment 7)

Provide a description, references and standards to describe CPSES configuration management/procedures including software.

Response:

The LEFM CheckPlus™ system is designed as a non-safety related system. Configuration management of the LEFM CheckPlus™ system is maintained by DBNPS Procedure NG-EN-00307, "Configuration Management". Software control for the LEFM CheckPlus™ system is in accordance with DBNPS Procedure NG-EN-00332, "Computer Software/Hardware Administrative Control". The Software and Firmware Verification and

Validation Report by Caldon is described in Topical Report ER-80P, Section 6.4, "Quality Measures in Design, Fabrication and Factory Acceptance Testing of the LEFM."

Question 2 (TXX-99115 - Attachment 7):

In response to Question 16 the methodology used to calculate calorimetric uncertainty is referenced as ASME PTC 19.1 - 1985, Measurement Uncertainty and is the same methodology as used to determine the uncertainty using the LEFM√ system.

A review of the CPSES FSAR and TS shows the following Information:

* Chapter 15 Page 15.0-16. Section 15.0.7, "Instrumentation Drift and Calorimetric errors - Power range neutron Flux" is deleted but references Section 15.0.6, "Trip Setpoints and Time Delays to Trip Assumed in Accident Analysis" references Section 7.1.2.1.9 and the CPSES Technical Specifications. This references Westinghouse setpoint methodology. PTC 19 is not referenced.

* The CPSES FSAR references RG 1.105 and the Westinghouse setpoint methodology not PTC 19.

* The CPSES Bases B 3/4 2-11 DNB parameters references the RCS total flow uncertainty as 1.8%. The uncertainty is stated to be based on Westinghouse Revised Thermal Design Procedure which includes measurements of reactor power. The methodology used to develop the associated uncertainties and includes specific treatment of feedwater flow uncertainties. PTC 19 is not referenced.

* FSAR Page 4.4-37 Reference 85 lists "Improved Thermal Design Procedure" as the methodology used. PTC 19 is not referenced.

Response:

Refer to the above DBNPS-specific Response to Question 16 (TXX-99105).

Question 3 (TXX-99115 - Attachment 7):

For Question 17, provide a calibration report from a calibration lab with accuracy traceable to NIST that indicates the accuracy of the LEFM in fully conditioned flow. Additionally, provide a test report from a calibration facility that shows the LEFM accuracy is unaffected by velocity profile changes including those based on piping geometry changes (reducers, header, elbows, etc.) such that it can be confirmed the LEFM is not sensitive to plant specific piping installation effects and that the calibration facility results are directly applicable to a plant specific installation.

Response:

The LEFM CheckPlus™ system will be calibrated in hydraulically similar piping at Alden Research Laboratories prior to installation. The results from the calibration laboratory report will be directly applicable to the plant-specific installation and will be incorporated in the site-specific uncertainty analysis for the LEFM CheckPlus™ system. This analysis can be made available for NRC review at the NRC's request.

Additionally no fewer than six LEFM CheckPlus™ spool pieces have been tested for other licensees, in a wide variety of geometry. The LEFM CheckPlus™ configuration has proven very insensitive to the upstream piping configuration.

Question 1 (TXX-99164):

The licensee needs to evaluate the effects of the power uprate on the tube degradation mechanisms (present and potential) including wear.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2 (TXX-99164):

Discuss how steam generator tube inspection plan will be assessed to monitor potential tube degradation including wear. Will additional inspections be necessary? How will TXU Electric assess their inspection plans should new degradation mechanisms be discovered?

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 3 (TXX-99164):

The licensee needs to evaluate if the Technical Specification plugging limit of 40 percent through wall degradation is still adequate.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 1 (TXX-99195):

Provide a comparison of the relevant acceptance criterion to the appropriate design limit (e.g., DNBR, RCS pressure) for each of the following safety analyses:

15.4.2	Uncontrolled RCCA withdrawal from power
15.4.7	Misloaded fuel assembly
15.4.8	Rod Ejection
15.4.3	Dropped RCCA

Response:

Refer to Sections 3.10.3.2, 3.10.3.3, 3.10.3.14, and 3.10.3.16 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2 (TXX-99195):

The topical report detailing the analysis of an inadvertent boron dilution event (RXE-91-002-A) indicates that the analysis assumed a power level of 100 percent. Discuss the sensitivity of the analysis results to initial power level. Summarize the methods and results of any supporting sensitivity analysis and provide references.

Response:

Topical Report RXE-91-002-A is not applicable to the DBNPS. Refer to Section 3.10.3.4 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for a description of the impact of the power uprate on the moderator dilution accident.

Question 3 (TXX-99195):

Discuss the sensitivity of the analysis results to initial power level for the SG tube rupture event. Summarize the methods and results of any supporting sensitivity analysis and provide references.

Response:

The SGTR analysis of record is independent of power level, therefore this question is not applicable to the DBNPS license amendment application. Refer to Section 3.10.3.15 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for further details.

Question 4 (TXX-99195):

CPSES technical specifications contain a surveillance requirement (3.3.1.2) requiring that power levels measured by nuclear instruments and by the N-16 monitoring system be checked to within 2% of the daily calorimetric. Explain why this surveillance requirement is not being modified to require that the readings be within 1% of the calorimetric.

Response:

The DBNPS does not have an N-16 monitoring system, therefore this question is not applicable.

Question 5 (TXX-99195):

In response to a previous request for additional information, the revised overpower N-16 allowable value of 113.5% of rated thermal power was defended as having been derived based on WCAP-12123 methods. Provide the detailed calculation showing how the allowable value for the N-16 overpower trip was determined.

Response:

As described in the above DBNPS-specific Response to Question 24 (TXX-99105), the N-16 overpower trip setpoint is not applicable to the DBNPS.

Question 1 (TXX-99203):

In section 6 of the Caldon Topical, reference is made to use of the LEFM to calibrate the NIs. How does CPSES plan to use the LEFM and explain the relation of the LEFM as M&TE with regards to Appendix B?

Response :

The power range neutron detector channels are checked at least every 24 hours using the heat balance when power is above 15% per Technical Specification Table 4.3-1. As stated in USAR Section 7.8.1.1, "Neutron Detectors," the sum of the outputs from the two sections of each power range detector is calibrated within +/-2% of heat balance at 100% of rated thermal power (RTP). Using these guidelines, it is more correct to state that the neutron detector indication of reactor power is normalized, rather than calibrated, against the reactor power calculated with the LEFM CheckPlus™ system-based secondary plant power calorimetric measurement. The LEFM CheckPlus™ will be an installed constantly operated instrument; and will not be test equipment. It will replace the current venturi-based instrumentation that provides the feedwater flow input to the heat balance calculation. As such, the application of M&TE is not strictly appropriate.

As discussed in the above DBNPS-specific Response to Question 1 (TXX-99115 – Attachment 7), the LEFM CheckPlus™ is non-safety related (non-10CFR50 Appendix B).

Question 2 (TXX-99203):

Page 5.5 of the Caldon Topical discusses the use of the LEFM to correct the Venturi measurement. Page 8 of the TXU license amendment request also discusses the use of the LEFM for providing correction for venturi. What are CPSES plans when the LEFM is unavailable and the venturis are used for normalizing the NIs?

Response:

Through the use of the LEFM CheckPlus™ system, the power calorimetric uncertainty is shown to be less than 0.37% RTP. However, this uncertainty calculation is not applicable to the case where the power calorimetric is based on venturi-based feedwater flow indication, even if the LEFM CheckPlus™ system is used to correct the venturi-based feedwater flow indications for effects such as fouling.

The DBNPS will be operated in accordance with the safety analyses and the applicable power calorimetric uncertainty analysis. When the LEFM CheckPlus™ system-based calorimetric measurement is available, the plant will be operated at a nominal core power of up to 2817 MWt. The reactor operators will be provided with procedural guidance for those occasions when the LEFM CheckPlus™ is not available. As summarized below, for those instances a new section of the DBNPS Technical Requirements Manual (TRM) will specify the appropriate actions to be taken when the LEFM CheckPlus™ system is unavailable.

The DBNPS TRM and other appropriate plant procedures will specify that if the LEFM CheckPlus™ becomes unavailable during the interval between daily performances of the heat balance comparison with the neutron detector (Technical Specification Table 4.3-1), plant operations may remain at a thermal power of 2817 MWt while continuing to use the power indications from the neutron detector power range channels. However, in order to remain in compliance with the bases for operation at a Rated Thermal Power of 2817 MWt, the LEFM CheckPlus™ system must be returned to service prior to the next performance of the heat balance comparison required by Technical Specification Table 4.3-1. If the LEFM CheckPlus™ system has not been returned to service prior to the next performance of the heat balance comparison, the procedural guidance/TRM would require that the reactor power be reduced to, or maintained at, a power level of no greater than 2772 MWt.

This power level is consistent with the uncertainty previously assumed for the venturi-based indication of feedwater flow. This power reduction is intended to be performed prior to the Technical Specification Table 4.3-1 calibration being performed. The calibration would then be performed using the venturi-based feedwater flow indications in the case where the LEFM CheckPlus™ is unavailable. Once the calibration is performed using the corrected venturi-based feedwater flow indications, the assumed power uncertainty is 2% RTP even though the actual uncertainty is much better than this. In order to maintain compliance with the safety analyses, it would be necessary to operate the plant at a maximum core thermal power of 2772 MWt until the LEFM CheckPlus™ is restored. Once LEFM CheckPlus™ is restored, performance of the Technical Specification Table 4.3-1 calibration is required using the LEFM CheckPlus™ indication of feedwater flow. Upon completion of this calibration, the plant could again be operated at 2817 MWt.

Question 3 (TXX-98274):

Describe how the LEFM√ is used in calorimetric power determinations.

Response:

Refer to Section 3.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application and the above DBNPS-specific Responses to Questions 1 and 2 (TXX-99203).

Question 5 (TXX-98274):

Who is responsible and how are Calibration, Maintenance, and Training performed and achieved?

Response:

The Verification Test of the LEFM CheckPlus™ spool pieces is contracted by Caldon and will be performed at Alden Research Laboratories before the installation of the spool pieces into the main feedwater headers at the DBNPS. The installation requirements will be in accordance with the DBNPS plant modification process. The LEFM CheckPlus™ software has provisions for on-line monitoring and diagnostics and will alert the operator if the system has failed, or the performance of the system indicates a maintenance/alert condition. In that event, it may become necessary for maintenance to be performed. This necessary maintenance would be controlled in accordance with the DBNPS work control process.

Training on the operation and maintenance of the LEFM CheckPlus™ system is contractually provided by Caldon. Maintenance is planned to be performed by DBNPS plant personnel per vendor recommendations contained in vendor supplied instructions, and does not require any special skills that would be beyond that encompassed in the DBNPS I&C technician training program.

Question 6 (TXX-98274):

How will monitoring, verification, and error reporting be handled? Provide clarification (list) of Quality Control standards used by Caldon in the design and manufacturing of the LEFM. Provide clarification (list) as to the standards followed under Caldon's verification and validation program.

Response:

The DBNPS will include the LEFM CheckPlus™ system in the calibration and maintenance program, including the preventive maintenance program. The system will be monitored by the System Engineer for reliability. As a plant instrument, all equipment problems fall under the site work control process. All adverse conditions that are identified will be documented on a Condition Report (CR) in accordance with the DBNPS Corrective Action Program. The DBNPS has required Caldon to maintain the LEFM CheckPlus™ software under their V & V Program with requirements that Caldon notify the DBNPS of any deficiencies that could affect the design basis accuracy.

Although the LEFM system is not safety-related, it is designed and manufactured under Caldon's Quality Control Program, which provides for configuration control, deficiency reporting and correction, and maintenance. Specific examples of quality measures undertaken in the design, fabrication and testing of the LEFM system are provided in the Caldon Topical Report.

Question 10 (TXX-98274):

How does the LEFM√ uncertainty compare to the venturi uncertainty at Comanche Peak, in measuring reactor thermal power?

Response:

Refer to the above DBNPS-specific Response to Question 16 (TXX-99105).

Question 29 (TXX-98274):

How is the LEFM used currently to provide correction factors to the venturis? Is the correction determined on the basis of the absolute accuracy or the repeatability of the LEFM?

Response:

This question is not applicable to the DBNPS, as there is no LEFM presently installed. The LEFM, when installed, will not be used to provide correction factors to the venturis.

Question 30 (TXX-98274):

What action is taken when the LEFM fails?

Response:

Refer to the above DBNPS-specific Response to Question 2 (TXX-99203).

Question 34 (TXX-98274):

Provide a figure analogous to Figure 5-2 in the Topical using the Comanche Peak site-specific uncertainty values for the venturi and LEFM instruments.

Response:

This question is addressed in Figure 6 and the accompanying text in Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System."

APPLICABILITY OF WATTS BAR RAI QUESTIONS TO PROPOSED DAVIS-BESSE POWER UPRATE

For the proposed power uprate, the Davis-Besse Nuclear Power Station (DBNPS) has taken into consideration the Request for Additional Information (RAI) made by the NRC staff in their review of a similar power uprate license amendment application for the Watts Bar Nuclear Plant (WBN) Unit 1 (Amendment No. 31 to Facility Operating License NPF-90, dated January 19, 2001). The August 24, 2000 WBN RAI response was evaluated.

The following includes the questions that were addressed by the WBN in the August 24, 2000 letter, and a DBNPS-specific response for each question.

I. MECHANICAL ENGINEERING BRANCH

Question 1:

In section III.5.1.1 of Enclosure 1 and Page E6-16 of Enclosure 6, you stated that in most cases (but not all), revised fatigue usage and stress intensities of the reactor vessel components did not need to be calculated for the power uprate. Please identify components that are impacted by the power uprate and require further calculation. For these components evaluated for the uprated conditions, provide the maximum calculated stress and cumulative fatigue usage factor (CUF) at the critical locations of these components. Also, provide the allowable Code limits, and the Code and Code edition used in the evaluation for the power uprate. If different from the Code of record, provide the necessary justification.

Response:

For the reactor vessel, an evaluation was performed that showed that the new operational conditions after the power uprate are bounded by those used in the existing analyses. It was not necessary to recalculate stresses and usage factors since the existing stresses and usage factors remain applicable for the uprated power conditions. Refer to Section 3.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. (Note: the following response to Question 2 addresses reactor vessel internal components.)

Question 2:

In regard to Section III.5.2.3 of Enclosure 1, provide the maximum calculated stress and CUF at the critical locations of the reactor internal components (such as lower and upper core plates, core barrel, baffle/barrel, and fuel assembly) for the power uprate condition. If codes are used in the evaluation for the power uprate, provide the allowable Code limits, and the Code and Code edition. Confirm that methodology, assumptions and allowable limits used for the power uprate evaluation are the same as those in the current licensing basis of record.

Response:

An evaluation has been performed for impact on the RV and internals. This evaluation showed that the temperature changes for the uprated power condition are bounded by those in the RCS Functional Specification and thus are bounded by those used in existing analyses. Therefore, the existing loads remain valid and the stresses and fatigue values remain valid. Thus, existing stress reports remain valid for the uprated power conditions. Refer to Section 3.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 3:

In regard to section III.5.2.2 of Enclosure 1, provide an assessment of flow-induced vibration of the reactor internal components due to the changes of T_{hot} and T_{cold} for the power uprate.

Response:

The design of the reactor vessel internals components was addressed for flow-induced vibration in Topical Report BAW-10051. A comparative analysis based on the new operational conditions after the power uprate has been performed. It is concluded that the uprated power operational conditions are bounded by Topical Report BAW-10051. Therefore, the RV internals are structurally adequate for flow-induced vibration. Refer to Section 3.6.3.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 4:

In reference to section III.5.3 of Enclosure 1, provide an evaluation of the control rod drive mechanism with regard to the stress and fatigue usage as a result of the power uprate. Also, provide the allowable Code limits for the critical components evaluated, and the Code and Code edition used for the evaluation. If different from the Code of record, justify and reconcile the differences.

Response:

For the control rod drive mechanisms, an evaluation was performed that showed that the new operational conditions after the power uprate are bounded by those used in the existing analyses. It was not necessary to recalculate stresses and usage factors since the existing stresses and usage factors remain applicable for the uprated power conditions. Refer to Section 3.6.4 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 5:

In reference to Section III.6.4 of Enclosure 1, you stated that the 2-percent increase in forces (loop forces increase due to a reduction of T_{cold}) was offset by a more representative characterization of the loop at the break location. Explain more about the approach using "the more representative characterization of the loop," which was claimed to result in 17-percent reduction in loop force at the break location. Is this approach currently used by WBN for a licensing basis documented in the UFSAR?

Response:

This question does not apply to the DBNPS power uprate because there is no increase in force since the T_{cold} temperature after power uprate is bounded by the current RCS Functional Specification. Thus the current design analysis remains applicable. Refer to Section 3.6.5 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 6:

Provide evaluation of the potential of flow induced vibration for the steam generator U-Bend tubes quantitatively based on the increase in feedwater flow and the increase in pressure difference between the primary system pressure (unchanged at 2250 psi) and the decreased steam pressure for the proposed power uprate.

Response:

The DBNPS steam generators are of OTSG design and therefore do not have a U-Bend tube configuration.

Refer to Section 3.6.7.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for a discussion of the FIV effects on the OTSG tubes.

Question 7:

In Section III.7, "Balance of Plant," you stated that as part of design change process for the power uprate, additional heat balance studies will be performed at higher ambient conditions to assess potential impact on individual BOP components. Please provide such an evaluation and identify systems and components that will be affected by the higher ambient conditions for the power uprate.

Response:

As described in Section 3.8.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, heat balance studies have been performed at bounding OTSG and Circulating Water conditions. The bounding conditions were used to assess the equipment. Based on this, no major impacts on balance of plant equipment are expected.

Using the revised NSSS parameters, the DBNPS has performed a heat balance for the proposed uprate. As reported in Table 3-3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, a comparison of the uprate heat balance with the current 100% heat balance revealed no significant differences in pressures, temperatures, or flows for the secondary side plant systems.

Based on the DBNPS evaluations, the Balance of Plant systems are deemed adequate for the increase in thermal loads produced by the power uprate. Areas of consideration that were explored further in the Watts Bar submittal included the main condenser backpressure, condensate polishing inlet temperature, main feed pump turbine and associated condenser, high pressure turbine impulse pressure, flow instrumentation range limitations, and the high pressure reheater operating vent line. The following paragraphs discuss each of these items with respect to the DBNPS:

<u>Main Condenser Backpressure</u>

The DBNPS low vacuum pressure alarm setpoint is 7.5 in Hg. As shown in Table 3-3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, the bounding predicted power uprate high pressure condenser vacuum (20% OTSG tube plugging and 95°F circulating water) is 4.55 in Hg.

<u>Condensate Polishing Inlet Temperature</u>

No steam generator blowdown flow directly enters the polisher during normal operation. The condensate polishing system will experience a small increase in temperature (approximately 2.6°F). This is well within the maximum allowable temperatures of the resin and the filtering elements. The condensate flow will remain within the design flow of the polishers during normal operation, assuming 3 out of the 4 polishers are in service. The purity of the system is not expected to be significantly different during normal operation, and a review of the design documents for the condensate demineralizers indicates that the power uprate is acceptable for this system.

Main Feedwater Pump Turbine (MFPT) and Associated Condenser

The Main Feedwater Pump Turbine discharges to the Main Condenser. For the Main Condenser evaluation, refer to Section 3.8.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

High Pressure (HP) Turbine Impulse Pressure

As shown in Table 3-3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, the turbine throttle valve inlet pressure is maintained unchanged for power uprate. The impact of the power uprate on the Integrated Control System is addressed in Section 3.7.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Flow Instrumentation Range Limitations

The Balance of Plant instrumentation have been reviewed and deemed sufficient for the proposed increase in power. The ranges of several flow measuring devices are insufficient based on maximum conditions after the proposed power uprate. However, these instruments provide no control or safety function and are not being modified at this time. These instruments only provide non-essential monitoring capability for secondary-side parameters. All the BOP systems have been evaluated to be capable of operating under the power uprate conditions. Therefore, no changes to the control logic are proposed. Refer to Section 3.8.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. However, the total power calorimetric uncertainty using LEFM was evaluated and resulted in the uncertainties for several BOP instrument channels having to be re-calculated using current methodology. This required several BOP instrument loop accuracies to be revised.

The impact of power uprate on the Integrated Control System is addressed in Section 3.7.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

HP Reheater (Second Stage) Operating Vent Line

The HP reheaters were evaluated. Refer to Sections 3.8.1 and 3.8.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 8:

On Page E6-22 of the reference, you indicated that the licensing basis conditions for the motor-operated valves (MOVs) program by TVA bound the uprated conditions and therefore, the safety-related MOVs at WBN will be capable of performing their intended function(s) following the power uprate. Please discuss effects of the proposed power uprate on the pressure locking and thermal binding of the safety-related power-operated gate valves for Generic Letter (GL) 95-07, and on the evaluation of overpressurization of isolated sections of piping segment for GL 96-06. Identify mechanical components for which functionality at the uprated conditions could not be confirmed.

Response:

GL 95-07:

Refer to Section 4.4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

GL 96-06:

Refer to Section 4.4.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 9:

Describe superscripts "a" and "c" which are not defined in Tables 1 and 2 on Pages E6-20 and E6-21.

Response:

This clarification question is specific to the Watts Bar submittal and is not applicable to the DBNPS license amendment application.

Question 10:

Do you project modifications to piping or equipment supports for the proposed power uprate? If any, provide examples of pipe supports requiring modification and discuss the nature of these modifications.

Response:

As stated in Section 3.8.11 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, no piping or pipe support modifications are required as a result of the increased power level.

II. REACTOR SYSTEMS ENGINEERING BRANCH

Question 1:

The SG Atmospheric Relief Valves (ARVs) are discussed on Page El-16 of TVA's application. Provide additional information to justify the adequacy of the ARVs' design relief capacity for the 1.4% uprate.

Response:

Refer to Section 3.7.1.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2:

With respect to the discussion of BELBLOCA, Page E1-35 of TVA's application, discuss the relationship between the MONTEC computer Code and WCOBRA/TRAC and whether it may be used separately from WCOBRA/TRAC.

Response:

This question is not applicable to the DBNPS license amendment application.

Question 3:

Section 6.5.1, beginning on page E1-37 of TVA's application provides a discussion of the affects on the Non-LOCA/Transient Analyses for the 1.4% power uprate. Please provide additional information to justify the conclusion that DNBR margins remain acceptable.

Response:

Refer to Section 3.13.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 4:

TVA's application discusses the Rod Ejection Event, on Page E1-44. Please discuss the acceptance criteria for the fuel pellets with respect to 10 CFR 50, Appendix-A, General Design Criteria 28.

Response:

Refer to Section 3.10.3.16 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 5:

Please provide additional information to justify TVA's conclusion on Page E1-45, that Reactor Trip and ESFAS Setpoints remain acceptable for the 1.4% Power Uprate.

Response:

As described in the DBNPS license amendment application, including Section 3.10.5 of Enclosure 1 Attachment 3, an evaluation of the plant setpoints has been performed relative to the power uprate request. This evaluation concluded that the RPS high flux related setpoints must be reduced in order to preserve the existing accident analyses. Furthermore, the variable low pressure trip setpoint must also be revised.

III. MATERIALS AND CHEMICAL ENGINEERING BRANCH

Question 1:

Section 5.6.5 - TVA stated that "...T_{hot} is expected to increase by 0.4 degree F for the 1.4% uprate and is considered to be the most sensitive operating parameter with respect to corrosion..." TVA also stated that "...these changes are expected to have an insignificant effect on the tube corrosion mechanisms since they are relatively minor and are comparable to the range of uncertainties used in assessing corrosion..."(1) TVA should expand on why the increase in T_{hot} is the most sensitive operating parameter with respect to corrosion. (2) If the increase in T_{hot} is within the range of uncertainties used in assessing corrosion and is relatively minor, TVA needs to describe the uncertainties in terms of quantitative or qualitative analysis to support the above statement.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2:

Section 5.6.5 - TVA stated that "...With regard to pre-heater wear, the 1.4% uprate conditions result in a slight increase in flow through the main feedwater nozzle which can impact the rate of wear. This slight increase in flow is not expected to result in a significant increase in the wear rate, and the resultant flow is within the pre-heater design flow..." (1) What is the flow rate through the main feedwater nozzle after the uprate? (2) What is the design flow rate for the pre-heater? (3) Does increase in T_{hot} affect the pre-heater wear?

Response:

Refer to Table 3-1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. The OTSG does not have a single feedwater nozzle or preheater region as is associated with the question.

Question 3:

Section 5.6.5 - TVA stated that "...For anti-vibration bar (AVB) wear, the slightly increased steam flow and reduced steam pressure can impact the flow induced vibration and wear. The revised design conditions will have a negligible impact on the projected AVB wear rate..." These two statements seem to be incongruent. The first statement indicates that the increase in steam flow and pressure reduction will affect the AVB wear. The second statement indicates that these changes will have negligible impact on the AVB wear rate. TVA needs to clarify the ambiguity.

Response:

The DBNPS steam generators are OTSG designs and do not employ anti-vibration bars. Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for a discussion of tube-to-tube support plate wear.

Question 4:

Section 5.6.5 - TVA needs to address (1) whether the steam generator tubes would satisfy Regulatory Guide 1.121 under the power uprate condition. (2) the impact of the power uprate on the tube inspection during future outages.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 5:

Section 5.6.6 - TVA performed a preliminary assessment to confirm that the existing 40% through wall plugging criteria will remain adequate for the power uprate condition. Provide the final assessments for staff review.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 6:

Section 5.6.7 - Discuss whether the increase in T_{hot} would affect the proposed outside diameter stress corrosion cracking (ODSCC) voltage-based alternate repair criteria (ARC).

Response:

This question is not applicable to the DBNPS license amendment application since the DBNPS has not applied for a voltage-based ARC for the tubes.

Question 7:

Section 5.6.7 - TVA stated that "...The ODSCC ARC was developed to replace the application of the generic 40% depth plugging criterion for tube cracking at elevations corresponding to tube support plate intersections..." It should be noted that the ODSCC ARC are applicable only to predominate axial tube cracking at tube support plates. The ARC are not applicable to circumferential cracking. Clarify if that is the intent of the above statement.

Response:

This question is not applicable to the DBNPS license amendment application since the DBNPS has not applied for a voltage-based ARC for the tubes.

Question 8:

Section 5.6.7 - TVA stated that "...The loading conditions compared to applicable criteria are only operative during faulted conditions, since the tube degradation is confined to the tube/tube support plate intersection crevice during normal operation ..." (1) Clarify the above statement. Specifically, what is meant by "...the loading conditions compared to applicable criteria are only operative during faulted conditions...?" (2) Do the temperature and primary-to-secondary pressure differential change for the faulted condition under power uprate?

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 9:

Section 5.6.7 - TVA stated that "...the structural and leakage criteria do apply during the application of faulted loading conditions; however, these are unaffected by the 1.4% uprate ..."(1) Discuss how the conclusion was reached. (2) Was there any calculations or assessments performed?

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 10:

Section 5.6.7 - TVA needs to address (1) the impact of the power uprate on tube degradation itself, i.e., would the power uprate affect the ODSCC degradation mechanism? (2) The impact of power uprate on the methodology (the assumptions and parameters used) for condition monitoring and operational assessments.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 11:

TVA needs to make an overall conclusion as to the structural and leakage integrity of steam generator tubes under power uprate conditions.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question on Section 4.2.5 -Steam Generator Blowdown System:

In the submittal, you have indicated that the required flow rates in the steam generator blowdown system are not expected to be significantly affected by the 1.4% power uprate. The reason you gave was that the power uprate will not significantly impact addition of dissolved solids and particulates into the steam generators. Please, provide technical basis justifying that the power uprate will not significantly change dissolved solids and particulates introduced into the steam generators and there will be no need, therefore, for changing the flow rates in the blowdown system.

Response:

This question is not applicable to the DBNPS license amendment application. As noted in Section 3.7.5 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, the Steam Generator Blowdown System is only used during startup, shutdown, and at low power levels, and the proposed power uprate will have no effect on system operation.

REACTOR VESSEL FLUENCE

Question:

In section 5.1.2, TVA indicates that existing neutron fluence projections bound the corresponding projections for the 1.4% uprated conditions. What are the existing values and the uprated values?

Response:

The controlling beltline material for the DBNPS reactor vessel is the upper shell to lower shell circumferential weld, WF-182-1, with a current inside surface fluence of 1.07E+19 n/cm^2. This value is reported in Topical Report BAW-2108, Revision 1. With the power uprate, the reactor vessel fluence at the upper shell to lower shell circumferential weld, WF-182-1 is expected to increase to 1.1235E+19 n/cm^2. Refer to Section 3.6.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

APPLICABILITY OF BEAVER VALLEY RAI QUESTIONS TO PROPOSED DAVIS-BESSE POWER UPRATE

For the proposed power uprate, the Davis-Besse Nuclear Power Station (DBNPS) has taken into consideration the Requests for Additional Information (RAIs) made by the NRC staff in their review of a similar power uprate license amendment application for the Beaver Valley Power Station (BVPS) Units 1 and 2 (Amendment No. 243 to Facility Operating License No. DPR-66, and Amendment No. 122 to Facility Operating License No. NPF-73, dated September 24, 2001). The February 20, May 7, and May 18, 2001 BVPS RAI responses were considered. In addition, RAIs contained in e-mail records dated May 3, 2001, and May 21, 2001 were considered.

The following includes the questions addressed by the BVPS in the above-referenced correspondence, and a DBNPS-specific response for each question.

FEBRUARY 20, 2001 LETTER

Question 1:

In your submittal dated January 18, 2001, you enclosed the Caldon, Inc. Engineering Report, ER-157P, "Supplement to Topical report ER-80P: Basis for a Power Uprate With LEFM✓™ or CheckPlus™ System, Revision 2," dated December 2000. It is the NRC staff's understanding that Caldon has decided to revise this topical report. With respect to those units utilizing the LEFM✓™ system, it is recommended that amendment requests for a 1.4 percent power uprate should base their justification on Caldon Topical Report ER-160P, which the NRC staff approved by its January 19, 2001, Safety Evaluation (SE) for Watts Bar (ADAMS accession number ML010260074).

Response:

The DBNPS is utilizing the LEFM CheckPlus™ system described in ER-157P. ER-157P reconciles the bounding uncertainties of the LEFM CheckPlus™ system described in ER-80P and ER-160P using the same methodology. ER-157P further describes the impact of the 1.7% uprate based on the LEFM CheckPlus™ system as compared to the current nozzle-based calorimetric and 1.4% uprate with a LEFM✓™ system.

ER-157P is a required reference in the DBNPS license amendment application because the DBNPS is requesting a 1.63% uprate.

Question 2:

The NRC staff has not approved a topical report for the use of the CheckPlus™ system. In light of the pending revisions to ER-157P please provide justification for the use of the CheckPlus™ system in support of the 1.4 percent power uprate request (i.e., please provide justification that the CheckPlus™ system is at least as good as the LEFM✓™ system).

Response:

The DBNPS is being equipped with the LEFM CheckPlus™ system. As described above, justification for the LEFM CheckPlus™ system is described in Caldon, Inc. Engineering Report ER-157P. The staff is requested to review ER-157P for its applicability to the DBNPS 1.63% power uprate request. ER-157P characterizes the performance of both the LEFM✓™ and LEFM CheckPlus™ systems using measured data for systems in service. Therefore, the performances of LEFM systems are slightly better than that reported in ER-80P. DBNPS personnel have reviewed ER-157P and found it acceptable and applicable to the proposed power uprate.

Question 3:

The staff SE on Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System," dated March 8, 1999 (accession number 9903190053), included 4 additional criteria to be addressed by a licensee requesting power uprate (see page 5 of March 8, 1999, SE). Your submittal did not address all of these criteria. Please address each of the four criteria.

Response:

The four criteria contained in ER-80P are listed below, followed by the DBNPS-specific response.

<u>Criterion 1</u>

The licensee should discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

Response to Criterion 1

Refer to Section 1.0 of Enclosure 1 Attachment 3 and the Enclosure 1, Attachment 9 DBNPS-specific Response to Question 2 (TXX-99203), of the DBNPS license amendment application.

Criterion 2

For plants that currently have LEFM installed, the licensee should provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in topical report ER-80P.

Response to Criterion 2

This criterion is not applicable to the DBNPS since the DBNPS currently uses venturis to obtain the daily calorimetric heat balance measurements. The DBNPS is installing a new LEFM CheckPlus™ System as the basis for the requested uprate. It is planned for installation in the upcoming Thirteenth Refueling Outage.

Criterion 3

The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feed water instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

Response to Criterion 3

Enclosure 1 Attachment 8 of the DBNPS license amendment application describes the proposed methodology for determining the uncertainties in calorimetric thermal power measurements and reactor coolant system flow measurements. The total power calorimetric measurement error, both with and without the LEFM, is calculated. The evaluation utilizes Statistical Core Design (SCD) methodology, which is an alternative approach to the currently accepted plant methodology regarding the development and treatment of instrument uncertainties. This methodology complies with the recommendations of ANSI/ISA-67.04 Part I - 1994, and NRC Regulatory Guide 1.105, Revision 3. The SCD methodology is described in Topical

Report BAW-10187P-A, "Statistical Core Design for B&W-Designed 177-FA Plants," B&W Fuel Company, Lynchburg, Virginia, March 1994.

Criterion 4

Licensees for plant installations where the ultrasonic meter (including the LEFM) was not installed and flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. The justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

Criterion 4 does not apply to DBNPS. The calibration factor for the DBNPS spool pieces will be established by tests of these spools at Alden Research Laboratories, prior to installation in the plant. These will include tests of a full-scale model of the DBNPS hydraulic geometry and tests in a straight pipe. An Alden data report for these tests and a Caldon engineering report evaluating the test data will be on file. The calibration factor used for the LEFM CheckPlus™ at DBNPS will be based on these reports. The uncertainty in the calibration factor for the spools will be based on the Caldon engineering report. The site-specific uncertainty analysis will document these analyses. This document will be maintained on file, as part of the technical basis for the DBNPS uprate.

Final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process verifies bounding calibration test data (See Appendix F of ER-80P). This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Enclosure 1 Attachment 8 of the DBNPS license amendment application. Final commissioning is expected to be completed in April 2002.

May 7, 2001 LETTER

Question 1:

Describe how the proposed power uprate will change the plant emergency and abnormal procedures.

Response:

Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2:

Describe any new risk-important operator actions required as a result of the proposed power uprate. Describe changes to any current risk-important operator actions that will occur as a result of the power uprate. Explain any changes in plant risk that result from changes in risk-important operator actions.

(e.g., Identify operator actions that will require additional response time or will have reduced time available. Identify any operator actions that are being automated as a result of the power uprate. Provide justification for the acceptability of these changes).

Response:

Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. Additionally, there are no changes anticipated to current risk-important operator actions as a result of the power uprate. The power uprate is not expected to have any significant effect on the manner in which the operators control the plant, including operator response times described in the USAR and modeled in the individual plant examination(IPE), either during normal operations or transient conditions. There are no operator actions that are being automated as a result of the uprate.

Question 3:

Describe any changes the proposed power uprate 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 power uprate and how operators were tested to determine they could use the instruments reliably.

Response:

A control room audible and visual annunciator will be provided to alarm LEFM trouble or failure. The LEFM also provides local visual indication designed to indicate when LEFM maintenance is required. This indication will also be logged on the plant computer terminal in the control room.

The Integrated Control System will be tuned for the power uprate conditions as identified in Section 3.7.6 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

No other changes to controls, displays or setpoints are required as a direct result of the proposed power uprate. No existing controls, displays, or alarms are being upgraded from analog to digital. The proposed power uprate does not result in changes to plant operating conditions that would require control system setpoint modifications.

The below Response to Question 5 addresses operator training.

Question 4:

Describe any changes the proposed power uprate will have on the Safety Parameter Display System. How will the operators know of the changes?

Response:

Refer to Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

The below Response to Question 5 addresses operator training.

Question 5:

Describe any changes the proposed power uprate will have on the operator training program and the plant reference control room simulator, and provide the implementation schedule for making the changes.

Response:

Refer to Section 4.1.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for a discussion of the impact of the proposed power uprate on the plant simulator. As described in Section 4.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, changes associated with the power uprate will be treated in the same manner consistent with any other plant modification, and will be included in Operator Training accordingly.

May 18, 2001 LETTER

Question:

To complete its review of the proposed license changes, the staff requests a description of the programs and procedures that will control calibration of the Caldon Leading Edge Flow Meter (LEFM) and the pressure and temperature instrumentation whose measurement uncertainties affect the power calorimetric uncertainties listed in table 12 of WCAP-15264. In this description, please include the procedures for:

1. Maintaining calibration,
2. Controlling software and hardware configuration,
3. Performing corrective actions,
4. Reporting deficiencies to the manufacturer, and
5. Receiving and addressing manufacturer deficiency reports.

Response to Part 1:

The plant process computer is utilized to perform the power calorimetric computation, based on input from temperature, pressure, and flow rate instrumentation in the feedwater, main steam, RCS makeup, and RCS letdown systems.

The LEFM CheckPlus™ system will provide feedwater flow and temperature input to the plant process computer. As stated in Section 1.0 of Enclosure 1 Attachment 3 of the DBNPS license amendment application, procedures for maintenance and calibration of the LEFM CheckPlus™ system will be developed for the DBNPS based on the vendor's recommendations.

The instrumentation utilized in the power calorimetric computation will continue to be maintained under the existing DBNPS instrumentation calibration procedures. The DBNPS calibration procedures, including calibration frequencies, were reviewed and found to be consistent with the assumptions and methodologies utilized in the heat balance uncertainty analysis provided in Enclosure 1 Attachment 8 of the DBNPS license amendment application.

Response to Part 2:

The methods for controlling software and hardware configuration for the LEFM CheckPlus™ system are addressed, in part, in Section 3.2 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. The LEFM CheckPlus™ system is designed as a non-safety related system for the DBNPS. Configuration management of the LEFM CheckPlus™ system will be in accordance with DBNPS procedure NG-EN-00307 "Configuration Management". Software control for the LEFM CheckPlus™ system will be in accordance with DBNPS procedure NG-EN-00332 "Computer Software/Hardware Administrative Control". The Caldon software and firmware Verification and Validation program is described in Topical Report ER-80P, Section 6.4, "Quality Measures in Design, Fabrication and Factory Acceptance Testing of the LEFM√."

Plant process computer software changes required to adapt the power calorimetric computation to the new LEFM CheckPlus™ system inputs will be performed in accordance with the DBNPS software control procedure. This procedure ensures that proper documentation, testing and reviews are conducted.

Hardware and setpoint changes are made in accordance within the plant design change process, with the same process being applied to both safety and non-safety systems and components.

Response to Part 3:

Corrective actions are required whenever conditions are identified outside of the design or operability requirements. Conditions adverse to quality are identified using the existing plant corrective action process.

The corrective action process provides for identification of corrective actions, evaluation of steps to prevent reoccurrence, and, if appropriate, root cause analysis. The corrective action process applies to both safety related and non-safety-related systems and components.

Response to Part 4:

Conditions identified with vendor equipment are reported to the vendor and processed in accordance with the existing plant corrective action process.

Response to Part 5:

In accordance with the existing plant corrective action process, conditions identified by vendors, or identified in industry events, are collected and evaluated as to the applicability to the DBNPS.

May 3, 2001 E-MAIL

Question on Section 3.6.7.3 U-Bend Fatigue Evaluation:

In this section, the licensee stated that "...a preliminary assessment indicates that the existing 40-percent through wall plugging criterion for steam generator tubes will remain adequate. FENOC will perform a calculation to substantiate the adequacy of the plugging criterion..." The licensee needs to discuss its preliminary assessment and pending calculation regarding the adequacy of the 40-percent through-wall plugging criterion under the power uprate conditions.

Response:

Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question on Section 3.6.7.5 Inspection Program and Tube Repair Criteria:

The licensee discussed the impact of the power uprate on steam generator tube degradation mechanisms such as anti-vibration bar wear and degradation at the tube support plate intersections. As discussed in the licensee's inspection reports and phone calls, the following degradation was identified in Beaver Valley Unit 1 during the tube inspection performed in the Spring of 2000: primary water stress corrosion cracking (PWSCC) in row 1 U-bend, PWSCC at the top of the tubesheet, outside diameter stress corrosion cracking (ODSCC) in the sludge pile region, ODSCC at the tube support plate intersections, and cold leg thinning. The following degradation was identified in Beaver Valley Unit 2 during the inspection performed in the Fall of 2000: anti-vibration bar wear, ODSCC at tube support plate intersections, outside diameter degradation at the top of the tubesheet. The licensee needs to discuss the impact of the power uprate on those degradation mechanisms that were not discussed in the January 18, 2001, submittal.

Response:

The tube degradation mechanisms identified at Beaver Valley Units 1 and 2 are not ongoing forms of degradation at the DBNPS. Refer to Section 3.6.7.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application for ongoing forms of degradation applicable to the DBNPS Steam Generators.

May 21, 2001 E-MAIL

Question 1:

The proposed technical specification (TS) bases B 3/4.4.7.1.1 indicates that the total relieving capacity for all main steam safety valves (MSSVs) is 108% of the total steam flow at rated thermal power. This capacity has been reduced from the current value of 110%. Provided justification of this proposed change in light of ASME code requirements for safety valves.

Response:

This question does not apply to DBNPS. The capacity of the MSSVs remains acceptable. Refer to section 3.7.1.1 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 2:

It is indicated in your submittal that the design bases transients and accidents have been evaluated at the uprated power level and the results of the analyses demonstrated that all the applicable acceptance criteria for each event continued to be met at the 1.4% power uprate conditions (considering the updated primary and secondary system temperatures, pressures, flows, etc). Please provide detailed results of the re-analyses in the following areas:

a) Major assumptions used in the re-analyses. Provide justification for any assumptions which are deviate from that used in the existing analyses.

b) Describe methods and computer codes used for the re-analyses and confirm that they are previously approved by the staff. Provide justification for any changes in methodology from the existing analyses.

c) Provide the results of the re-analyses including primary and secondary system peak pressure, minimum DNBR, and/or amount of failed fuel.

Response to Part 2a):

The evaluation of the DBNPS USAR Chapter 15 accidents concluded that the existing analyses that have been performed are valid for the power uprate. No new USAR Chapter 15 analyses were performed. New analyses were performed for the mass and energy release rates included in USAR Chapter 6 for the LOCA and MSLB events, however, no new assumptions were imposed for these calculations.

Response to Part 2b):

The mass and energy release rates for the Loss of Coolant Accident (LOCA) and the Main Steam Line Break (MSLB) analyses were recalculated using RELAP5/MOD2-B&W. The system responses, including the mass and energy release rates, for the LOCA and MSLB accidents have historically been generated using CRAFT2 and TRAP2, respectively. FRA-ANP has performed numerous benchmarks of RELAP5/MOD2-B&W to plant transient data and scaled and integral systems test data, and has compared the results to existing USAR analyses. The comparisons demonstrate that RELAP5/MOD2-B&W predicts the key phenomena of interest and, given the same conservative initial and boundary conditions, will calculate results similar to those produced by the previous analysis codes.

The NRC has reviewed and approved RELAP5/MOD2-B&W for both LOCA (BAW-10192P-A) and MSLB applications (BAW-10193P-A) relative to the overall system response. Since the analysis of record for the DBNPS is based on RELAP5/MOD2-B&W, the mass and energy release rates for the power uprate were also generated with RELAP5/MOD2-B&W.

Response to Part 2c):

The mass and energy release re-analyses results demonstrate that peak system pressure is not challenged. The calculation methods ensure that the mass and energy release is maximized. A different set of calculations is performed to challenge minimum DNBR and the amount of failed fuel. No re-analyses were required for these parameters.

Question 3:

In Section 3.7.4 of Enclosure 1 of your submittal, discuss the affect from higher decay heat to the adequacy of the safety related condensate storage tank volume in light of: a) To support AFW for achieving plant cooldown to RHR initiation, and b) To assure SBO coping analysis remain valid.

Response to Part 3a):

Section 3.7.4 of Enclosure 1 Attachment 3 of the DBNPS license amendment application identifies that the effect from higher decay heat on the adequacy of the condensate storage tank volume to support AFW for achieving plant cooldown to hot shutdown is minimal.

Response to Part 3b):

Refer to Sections 3.10.3.7 and 4.3 of Enclosure 1 Attachment 3 of the DBNPS license amendment application.

Question 4:

Please submit information that discusses effect of power uprate on ATWS analyses, including any changes in important core or energy release assumptions.

Response:

Refer to Section 3.10.3.18 of Enclosure 1 Attachment 3 of the DBNPS license amendment application. The effect of power uprate on the ATWS analyses is bounded by the existing design.

Docket Number 50-346
License Number NPF-3
Serial Number 2692
Enclosure 2

PROPOSED TECHNICAL SPECIFICATIONS AND BASES CHANGES
REVISION BAR FORMAT

(16 pages follow)

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2817 MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s).

Figure 2.1-1 Reactor Core Safety Limit

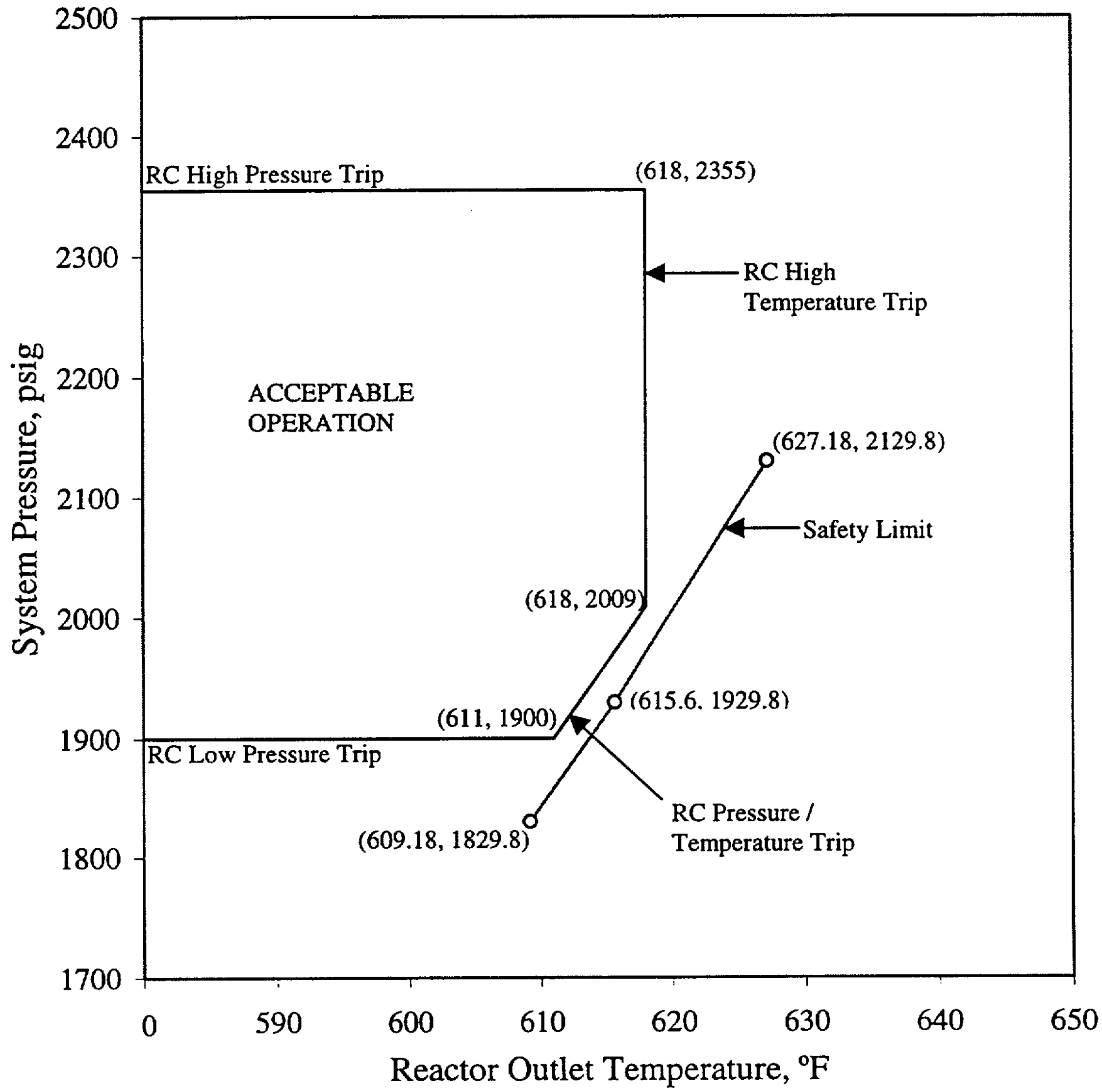


Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Allowable values
1. Manual reactor trip	Not applicable.
2. High flux	≤104.9% of RATED THERMAL POWER with four pumps operating* ≤80.6% of RATED THERMAL POWER with three pumps operating*
3. RC high temperature	≤618°F*
4. Flux -- Δflux/flow(1)	Pump allowable values not to exceed the limit lines shown in in the CORE OPERATING LIMITS REPORT for four and three pump operation.*
5. RC low pressure(1)	≥1900.0 psig*
6. RC high pressure	≤2355.0 psig*
7. RC pressure-temperature(1)	≥(16.25T_{out}°F - 8034) psig*
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop* ≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop* ≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating*
9. Containment pressure high	≤4 psig*

2.1.1 AND 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB using critical heat flux (CHF) correlations. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The B&W-2 and BWC CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to all B&W fuel with zircaloy or M5 spacer grids. The minimum value of the DNBR, accounting only for DNBR correlation uncertainty, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC). The minimum value of DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.313 (BWC) and accounts for all uncertainty values considered with the statistical core design methodology. The value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR equal to or greater than the correlation limit is predicted for the maximum possible thermal power of 110.2% of 2817 MWt when the reactor coolant flow is 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required measured flow is 389,500 GPM). This curve is based on the design hot channel factors with potential fuel densification and fuel rod bowing effects.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than the Statistical Design Limit (SDL) of 1.313 (BWC) and a local quality at the point of minimum DNBR less than +22% (B&W-2) or +26% (BWC) for that particular reactor coolant pump situation. The DNBR curve for three pump operation is less restrictive than the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation Allowable Values specified in Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shut-down bypass activated. This high pressure setpoint is lower than the normal low pressure setpoint so that the reactor must be tripped before the bypass is initiated. The high flux setpoint of ≤5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches the Allowable Value ≤ 104.9% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be at a thermal power of 110.2% of 2817 MWt, which was used in the safety analysis.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of the figure in the CORE OPERATING LIMITS REPORT are produced.

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

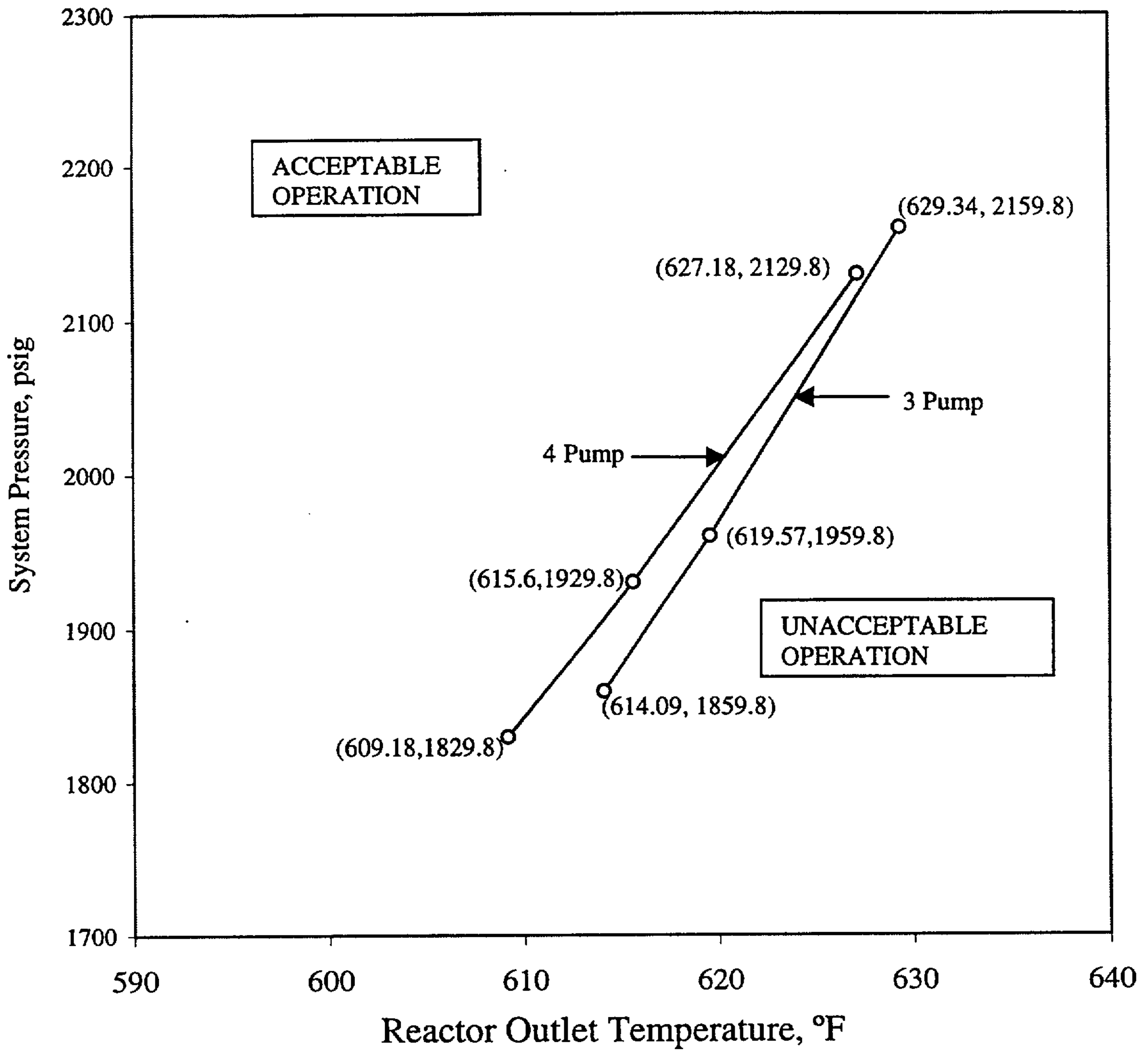
During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux setpoint. The Allowable Value for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection. The RC high pressure trip is, therefore, set lower than the set pressure for these valves, 2500 psig (nominal), even when accounting for the RPS RC pressure instrument string uncertainty. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature ($16.25T_{out}$-8034) psig, Allowable Values have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below the minimum allowable DNB ratio by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

Bases Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR



Pumps	Flow, gpm	Power	Required Measured Flow to ensure Compliance, gpm
4	380,000	110.2 %	389,500
3	283,860	89 %	290,957

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.9×10^{-4} Δk/k/°F whenever THERMAL POWER is < 80% of RATED THERMAL POWER,

b. Less positive than 0.0×10^{-4} Δk/k/°F whenever THERMAL POWER is ≥ 80% of RATED THERMAL POWER, and

c. Equal to or less negative than the limit provided in the CORE OPERATING LIMITS REPORT at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*# .

ACTION:

With the moderator temperature coefficient outside any of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 days after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

---

*With $k_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

TABLE 3.2-2

DNB MARGIN

Parameter	Required Measured Parameters with Four Reactor Coolant Pumps Operating	Required Measured Parameters with Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature T_H°F	≤ 610	≤ 610[1]
Reactor Coolant Pressure, psig.[2]	≥ 2064.8	≥ 2060.8 [1]
Reactor Coolant Flow Rate, gpm[3]	$\geq 389{,}500$	$\geq 290{,}957$

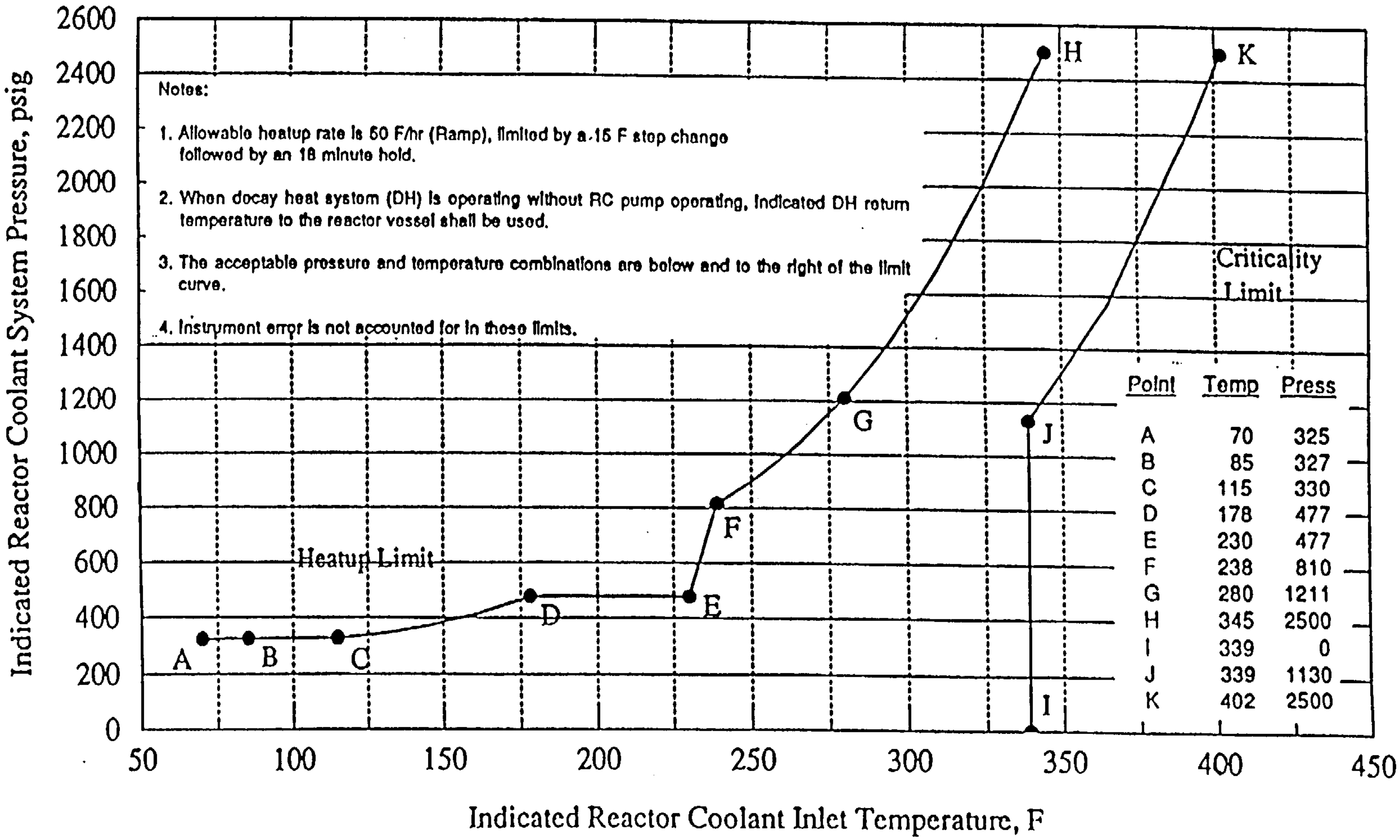
(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These minimum required measured flows include a flow rate uncertainty of 2.5%.

Figure 3.4-2

Reactor Coolant System Pressure-Temperature Limits
For Heatup and Core Criticality for the First 20 EFPY



Point	Temp	Press
A	70	325
B	85	327
C	115	330
D	178	477
E	230	477
F	238	810
G	280	1211
H	345	2500
I	339	0
J	339	1130
K	402	2500

Figure 3.4-3

Reactor Coolant System Pressure-Temperature Limits
For Cooldown for the First 20 EFPY

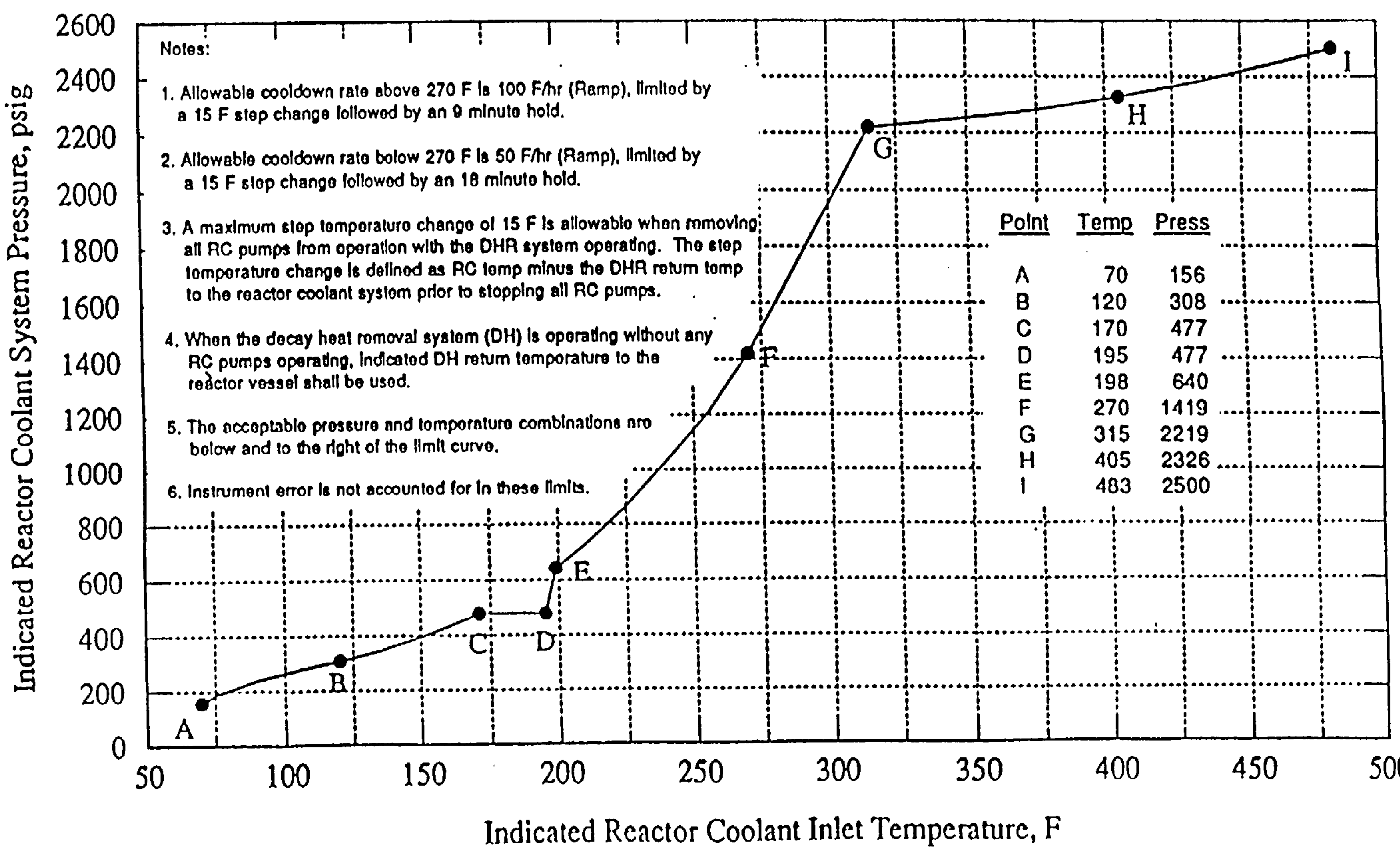
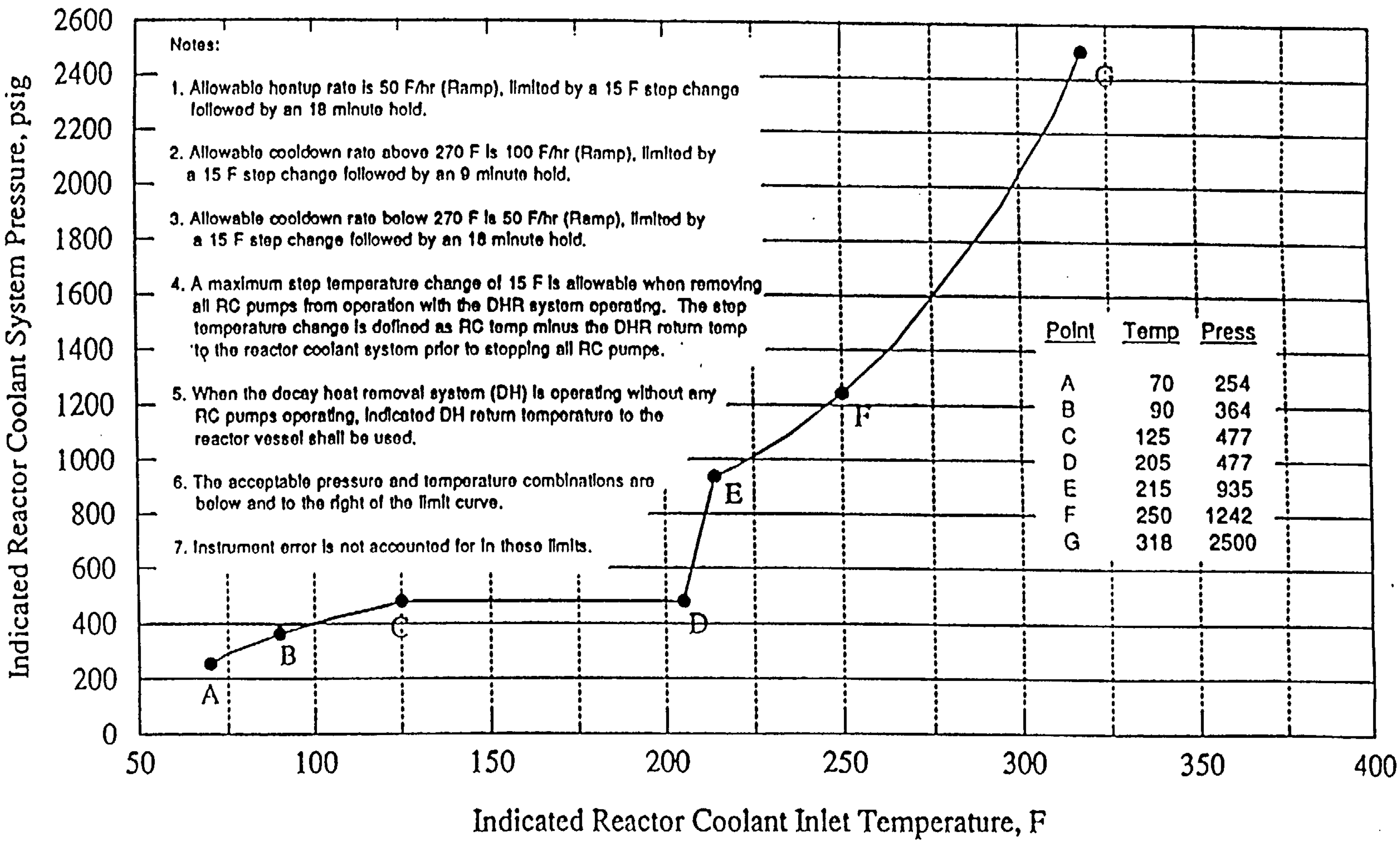


Figure 3.4-4

Reactor Coolant System Pressure-Temperature Heatup and
Cooldown Limits for Inservice Leak and Hydrostatic Tests
for the First 20 EFPY



The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves, are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of twenty effective full power years.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The procedures described in Regulatory Guide 1.99, Rev. 2, were used for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and nickel content and neutron fluence.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure-temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

All pressure-temperature limit curve are applicable up to twenty effective full power years. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, 3.4-3 and 3.4-4.

CORE OPERATING LIMITS REPORT (Continued)

As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.37% of rated thermal power may be used when input for reactor thermal power measurement of feedwater mass flow is by the Leading Edge Flow Meter (LEFM) CheckPlus™ System.

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revision or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
1. The DBNPS plans to install the LEFM CheckPlus™ System in both feedwater trains in the upcoming Thirteenth Refueling Outage (13RFO). [Enclosure 1 Attachment 1 Page 1 and Enclosure 1 Attachment 2 Page 1]	1. By the end of 13RFO.
2. The installation and post-maintenance testing of the LEFM system will be completed prior to increasing power above the current limit of 2772 MWt. [Enclosure 1 Attachment 1 Page 5]	2. Prior to increasing power above the current limit of 2772 MWt.
3. New procedures for maintenance and calibration of the LEFM system will be developed based on vendor recommendations. [Enclosure 1 Attachment 1 Page 5, Enclosure 1 Attachment 3 Section 1.0, and Enclosure 1 Attachment 11 Page 7]	3. Prior to increasing power above the current limit of 2772 MWt.
4. The revision of each of the analyses used to determine core operating limits, specifically to accommodate the proposed power uprate, would be a substantial administrative burden. In lieu of this administrative burden, it is proposed to allow the present versions of the reports to apply to the proposed power uprate, conditioned upon the LEFM being used to measure feedwater mass flow as the input to the reactor thermal power measurement. Consistent with the approach taken by the Tennessee Valley Authority (Reference 6), a requirement will be placed in the DBNPS	4. Upon implementation of the license amendment.

COMMITMENTS	DUE DATE
USAR requiring that future, plant-specific revisions of these reports, incorporate consideration of the 1.63% power uprate. [Enclosure 1 Attachment 1 Page 9]	
5. The DBNPS LEFM CheckPlus™ systems to be installed at the DBNPS will be extensively tested and calibrated at Alden Research Laboratories, in site-specific piping configurations, prior to their installation. [Enclosure 1 Attachment 3 Section 3.2] The LEFM CheckPlus™ system will be calibrated in hydraulically similar piping at Alden Research Laboratories prior to installation. The results from the calibration laboratory report will be directly applicable to the plant-specific installation and will be incorporated in the site-specific uncertainty analysis for the LEFM CheckPlus™ system. [Enclosure 1 Attachment 9 Page 20] The Verification Test of the LEFM CheckPlus™ spool pieces is contracted by Caldon and will be performed at Alden Research Laboratories before the installation of the spool pieces into the main feedwater headers at the DBNPS. [Enclosure 1 Attachment 9 Page 24] The calibration factor for the DBNPS spool pieces will be established by tests of these spools at Alden Research Laboratories, prior to installation in the plant. These will include tests of a full-scale model of the DBNPS hydraulic geometry and tests in a straight pipe. An Alden data report for these tests and a Caldon engineering report evaluating the test data will be on file. The calibration factor used for the LEFM CheckPlus™ at DBNPS will be based on these reports. The uncertainty in the calibration factor for the spools will be based on the Caldon engineering report. The site-specific uncertainty analysis will document these analyses. This document will be maintained on file, as part of the technical basis for the DBNPS uprate. [Enclosure 1 Attachment 11 Page 4]	5. Prior to installation of the LEFM CheckPlus™ systems.

COMMITMENTS	DUE DATE
6. A requirement will be placed in the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM) to address LEFM unavailability. Should the LEFM system be unavailable, the current feedwater flow instrumentation will be used as input to the core power calorimetric, and the core power will be limited to the original licensed power level of 2772 MWt. [Enclosure 1 Attachment 1 Page 5 and Enclosure 1 Attachment 3 Section 1.0 (similar wording)] The reactor operators will be provided with procedural guidance for those occasions when the LEFM CheckPlus™ is not available. As summarized below, for those instances a new section of the DBNPS Technical Requirements Manual (TRM) will specify the appropriate actions to be taken when the LEFM CheckPlus™ system is unavailable. The DBNPS TRM and other appropriate plant procedures will specify that if the LEFM CheckPlus™ becomes unavailable during the interval between daily performances of the heat balance comparison with the neutron detector (Technical Specification Table 4.3-1), plant operations may remain at a thermal power of 2817 MWt while continuing to use the power indications from the neutron detector power range channels. However, in order to remain in compliance with the bases for operation at a Rated Thermal Power of 2817 MWt, the LEFM CheckPlus™ system must be returned to service prior to the next performance of the heat balance comparison required by Technical Specification Table 4.3-1. If the LEFM CheckPlus™ system has not been returned to service prior to the next performance of the heat balance comparison, the procedural guidance/TRM would require that the reactor power be reduced to, or maintained at, a power level of no greater than 2772 MWt. [Enclosure 1 Attachment 9 Page 23]	6. Upon implementation of the license amendment.

COMMITMENTS	DUE DATE
7. The DBNPS' current Steam Generator program follows the inspection guidelines contained in the latest revision of the EPRI PWR Steam Generator Examination Guidelines. The modest power uprate will not require a change to the program. The DBNPS currently inspects for all active and potential degradation. The pre-outage degradation assessment includes DBNPS-specific degradation as well as industry degradation...Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs are made. Potential degradation growth rate changes will be incorporated into the operational assessment associated with potential effects of the uprate. [Enclosure 1 Attachment 3 Section 3.6.7.3]	7. Following implementation of the license amendment.
8. The existing design basis is exceeded for the condensate flow rate in the Steam Jet Air Ejector (SJAE) tubes at current power levels. Tube velocities will be slightly increased at uprate conditions. Periodic preventive maintenance inspections will be conducted to monitor wear in the SJAE. [Enclosure 1 Attachment 3 Section 3.8.2]	8. Following implementation of the license amendment.
9. The increase in condensate flow due to power uprate is evaluated in the low pressure feedwater heaters. The low pressure feedwater heaters are designed to operate at 20% above the original design flow. This bounds the normal operating flow after the proposed power uprate except for the Feedwater Heater #1 shell side flow, which increased to approximately 23.5% above the design flow. Periodic preventive maintenance inspections will be conducted to monitor feedwater heater #1 shell side wear. [Enclosure 1 Attachment 3 Section 3.8.2]	9. Following implementation of the license amendment.

COMMITMENTS	**DUE DATE**
10. The deaerators heat and scrub incoming feedwater, heater drains, and water from miscellaneous sources to remove air and other non-condensable gases. At the increased feedwater flows after power uprate, the ability of the deaerators to remove non-condensables from the feedwater system will be verified by on-line chemistry testing. [Enclosure 1 Attachment 3 Section 3.8.3]	10. Following implementation of the license amendment.
11. Although its use for calorimetric input is not nuclear safety related, the CheckPlus™ system's software has been developed and will be maintained under a verification and validation (V&V) program. [Enclosure 1 Attachment 3 Section 3.2]	11. Following implementation of the license amendment.
12. Extraction steam flow increases approximately 3-5% from the current steam flows...The flows from the LP Turbine to Feedwater Heaters #1, #2, the Deaerator, and Heater #4 exceed the original design. Periodic preventive maintenance inspections will be conducted to monitor wear due to the increased flows from the LP Turbine to Heaters #1, #2, the Deaerator, and Heater #4. [Enclosure 1 Attachment 3 Section 3.8.4]	12. Following implementation of the license amendment.

COMMITMENTS	**DUE DATE**
13. However, shell side flow through both the first stage and second stage reheaters exceed design. Periodic preventive maintenance inspections of the MSR will be conducted to monitor wear. [Enclosure 1 Attachment 3 Section 3.8.4]	13. Following implementation of the license amendment.
14. The Heater Drain valve between Heater #4 and the Deaerator is currently in the wide open position for full power operation and is planned to be replaced in 13RFO with a higher capacity valve. Additionally, it is planned to replace the low pressure feedwater heater drain tank level control valves at the discharge of the heater drain tank pumps with higher capacity control valves in 13RFO. [Enclosure 1 Attachment 3 Section 3.8.5]	14. By the end of 13RFO.
15. Due to limited capacity of the heater drain pumps, the levels of the heater drain tanks will be monitored when extraction steam flow across the LP feedwater heaters is greatest (cold weather conditions. [Enclosure 1 Attachment 3 Section 3.8.5]	15. Following implementation of the license amendment.
16. All turbine generator components were determined to have sufficient margin to enable operation at the uprated power conditions without requiring equipment modifications, except for the sequencing of control valve operation, which will be modified in the next refueling outage (13RFO). This modification is necessary to ensure high pressure turbine first stage bucket design limits are not exceeded. The Control Valve Diode Function Generator (DFG) cards will be	16. By the end of 13RFO.

COMMITMENTS	DUE DATE
recalibrated to accommodate the change of the control valve sequencing. [Enclosure 1 Attachment 3 Section 3.8.12]	
17. Simulator changes resulting from the power uprate will mimic the control room changes by adding an annunciator window and LEFM panel. Simulator changes will be implemented as part of the plant modification. [Enclosure 1 Attachment 3 Section 4.1.1]	17. Upon implementation of the plant modification.
18. A review of the training simulator fidelity with the new power rating will be included at the next regularly scheduled review following the uprate. [Enclosure 1 Attachment 3 Section 4.1.1]	18. At the next regularly scheduled review following the uprate.
19. Flow-Accelerated Corrosion (FAC), in the piping systems at the DBNPS, is modeled using the CHECWORKS computer program. CHECWORKS models will be revised, as appropriate, to incorporate flow and thermodynamic states that are projected for uprated conditions. The results of these models will be factored into future inspection/pipe replacement plans consistent with the current Corrosion/Erosion Monitoring and Analysis Program (CEMAP). [Enclosure 1 Attachment 3 Section 4.1.3]	19. Following implementation of the license amendment.
20. The power uprate is not expected to have any significant effect on the manner in which the operators control the plant, including operator response times...The power uprate will lead to minor changes in several plant parameters. These parameters include, but are not limited to, the 100% value for Rated Thermal Power, Reactor Coolant System Delta Temperature, Steam Generator Pressure and Main Feedwater and Steam flows. Changes associated with the power uprate will be treated in the same manner consistent with any other plant modification, and	20. Upon implementation of the license amendment.

COMMITMENTS	DUE DATE
will be included in Operator Training accordingly. [Enclosure 1 Attachment 3 Section 4.2]	
21. The only change in the alarms for the Safety Parameter Display System (SPDS) is the high reactor power level. The current alarm is set at 107% of 2772 MWt (2966 MWt), which is above the current RPS trip setpoint of 105.1%. The new alarm will be set at 106% of 2817 MWt (2986 MWt), which remains above the proposed new RPS trip setpoint of 104.9%. [Enclosure 1 Attachment 3 Section 4.2]	21. Prior to increasing power above the current limit of 2772 MWt.
22. The DBNPS will include the LEFM CheckPlus™ system in the calibration and maintenance program, including the preventive maintenance program. The system will be monitored by the System Engineer for reliability. [Enclosure 1 Attachment 9 Page 25]	22. Upon implementation of the license amendment.
23. Final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process verifies bounding calibration test data (See Appendix F of ER-80P). This step provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Enclosure 1 Attachment 8 of the DBNPS license amendment application. Final commissioning is expected to be completed in April 2002. [Enclosure 1 Attachment 11 Page 4]	23. Upon implementation of the license amendment.
24. A new annunciator will be installed to indicate Caldon flowmeter system trouble. [Enclosure 1 Attachment 3 Section 4.2] A control room audible and visual annunciator will be provided to alarm LEFM trouble or failure. The LEFM also provides local visual indication designed to indicate when LEFM maintenance is required.	24. Prior to increasing power above the current limit of 2772 MWt.

COMMITMENTS	DUE DATE
This indication will also be logged on the plant computer terminal in the control room. [Enclosure 1 Attachment 11 Page 6]	
25. The instrumentation utilized in the power calorimetric computation will continue to be maintained under the existing DBNPS instrumentation calibration procedures. [Enclosure 1 Attachment 11 Page 7]	25. Following implementation of the license amendment.
26. Plant process computer software changes required to adapt the power calorimetric computation to the new LEFM CheckPlus™ system inputs will be performed in accordance with the DBNPS software control procedure. This procedure ensures that proper documentation, testing and reviews are conducted. [Enclosure 1 Attachment 11 Page 8]	26. Prior to increasing power above the current limit of 2772 MWt.
27. A review of FIV analysis for plugs and stabilizers supplied by ABB/CE is ongoing and will be completed prior to implementation of the proposed power uprate. [Enclosure 1 Attachment 3 Section 3.6.7.2]	27. Prior to increasing power above the current limit of 2772 MWt.
28. The above discussion is specific to hardware supplied by B&W/FTI. A review of qualification reports and design calculations for repair hardware supplied by ABB/CE is ongoing and will be completed prior to implementation of the proposed power uprate. [Enclosure 1 Attachment 3 Section 3.6.7.3]	28. Prior to increasing power above the current limit of 2772 MWt.
29. The increase in generator output will require minor adjustments to several ICS modules that use MCR to determine their settings. No additional ICS tuning is expected. [Enclosure 1 Attachment 3 Section 3.7.6] The Integrated Control System will be tuned for the power uprate conditions… [Enclosure 1 Attachment 11 Page 6]	29. Following implementation of the license amendment.