50-397

February 28, 1994

MEMORANDUM FOR: Theodore R. Quay, Project Director (13E-17)
Project Directorate V
Division of Reactor Projects III, IV, V

FROM: Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Safety & Analysis

SUBJECT: WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PLANT NO. 2 - POWER UPRATE REVIEW (TAC NO. M87076)

Enclosed is the Reactor Systems Branch input to the Safety Evaluation Report being prepared by your Project Directorate for the subject power uprate license amendment. It is my understanding that you will use this and other technical branch inputs for developing the overall staff safety evaluation for this license amendment. Enclosure 2 is our SALP evaluation for this review.

/s/

Robert C. Jones, Chief
Reactor Systems Branch
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Enclosures:
As stated

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ENCLOSURE

INTRODUCTION

Washington Public Power Supply System (WPPSS), the licensee for Nuclear Plant No. 2 (WNP-2), submitted a request by letter on July 9, 1993 to uprate the licensed power level from 3323 MWt to 3486 MWt (Reference 1). This represents approximately a 4.9% increase in thermal power with a 5% increase in rated steam flow. The planned approach to achieve the higher power level consists of (1) an increase in the core thermal power to create an increased steam flow, (2) a corresponding increase in feedwater flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along extension of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in General Electric report NEDC 31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," June 1991 (Reference 2). The operating pressure will be increased approximately 15 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow. The increased core power will be achieved by utilizing a flatter radial power distribution while still maintaining limiting fuel bundles within their constraints.

2.2 Thermal Limits Assessment

The operating limit MCPR is determined on a cycle specific basis from the results of reload analysis, as described in General Electric report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactors Power Uprate," July 1991; and Supplements 1 and 2 (Reference 3). The MAPLHGR and LHGR limits will also be maintained as described in this reference. The plant specific safety evaluation for WNP-2 is contained in References 4, 5, and 6.

2.3 Reactivity Characteristics

2.3.1 Power/Flow Operating Map

The uprated power/flow operating map includes the operating domain changes for uprated power. The map includes the increased core flow (ICF) range and an uprated Extended Load Line Limit Analysis (ELLLA). The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Power has been rescaled so that uprated power is equal to 100% rated power. The changes to the power/flow operating map are consistent with the previously NRC approved generic descriptions given in NEDO-31984.

2.4 Stability

Ongoing activities by the BWR Owners' Group and the NRC are addressing ways to minimize the occurrence and potential effects of power oscillations that have been observed for certain BWR operating conditions (as required by General Design Criteria 12 of 10 CFR 50 Appendix A). GE has documented information and cautions concerning this possibility in Service Information Letter (SIL) 380 and related communications. The NRC has documented its concerns in NRC Bulletin No. 88-07 and Supplement 1 to that bulletin. While a more permanent resolution is being developed, Technical Specifications and associated implementing procedures, as requested by the NRC Bulletin, have been incorporated by the licensee which restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Specific operator actions have been established to provide clear instructions for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions.

The restrictions recommended by NRC Bulletin 88-07 and Supplement 1 to that Bulletin will continue to be followed by the licensee for uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWR Owners' Group and the NRC. This is acceptable to the staff.

2.5 Reactivity Control

2.6.1 Control Rod Drives and CRD Hydraulic System

The control rod drive (CRD) system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated steam flow and dome pressure.

The increase in dome pressure due to power uprate produces a corresponding increase in the bottom head pressure. Initially, rod insertion will be slower due to the high pressure. As the scram continues, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Therefore, an increase in the reactor pressure has little effect on scram time. The licensee has indicated that CRD performance during power uprate will meet current Technical Specification requirements. The licensee will continue to monitor by various surveillance requirements the scram time performance as required in the plant Technical Specifications to ensure that the original licensing basis for the scram system is preserved. For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flow required for CRD cooling and driving are assured by the automatic opening of the system flow control valve, thus compensating for the small increase in reactor pressure. Prior to implementation of power uprate, the flow control valves and CRD pumps will be tested to ensure they are capable of operating within their acceptable range with power uprate. The CRD system should therefore continue to perform all its safety-related functions at uprated power with ICF, and should function adequately during insert and withdraw modes.

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) provide this protection. The setpoints for the relief function of the SRVs are increased 15 psi for power uprate.

The operating steam dome pressure is defined to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to uprated power. The uprate dome pressure increase will require a change in the SRV setpoints. The appropriate increase in the SRV setpoints also ensures that adequate differences between operating pressure and setpoints are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in the number of unnecessary SRV actuations.

3.2 Code Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload amendment submittal. The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The ASME code allowable peak pressure for the reactor vessel is 1375 psig (110% of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is an MSIV closure with a failure of the valve position scram. This transient was analyzed by the license with the following assumptions: 1). core power is 3702 MWt (106.2% of the uprated power of 3486 MWt); 2). end-of-cycle nuclear parameters; 3). six SRVs out-of-service; 4). no credit for the relief mode of the SRVs; 5). technical specification scram speed; 6). three second MSIV closure time; and 7). initial reactor dome pressure of 1050 psia. The SRV opening pressures were +3% above the nominal setpoint for the available valves. The calculated peak pressure was 1335 psig which is below the ASME allowable of 1375 psig which is acceptable.

The number of SRVs which will be assumed to be out of service is based on the maximum allowed by Technical Specification. Uprated conditions will produce a higher peak RPV pressure, and with reduced valve grouping, the reload analysis must show that it remains below the 1375 psig ASME code limit. The licensee's analysis plan is acceptable.

3.4 Reactor Recirculation System

Power uprate will be accomplished by operating along extensions of rod lines on the power/flow map with allowance for increased core flow. The cycle-specific core reload analyses will consider the full core flow range, up to 115 million lbm/hr. The evaluation by the licensee of the reactor recirculation system performance at uprated power with ICF determined that the core flow can be maintained. The design pressures for the Reactor Recirculation Control (RRC) System components includes the suction, discharge and flow control valves, recirculation pumps, and piping. Raising the steam pressure by 15 psig as a result of power uprate will raise the pump suction pressure by 17 psig and the pump discharge pressure by 45 psig. The licensee states that these increases in normal operating pressures are bounded by the system design pressure. Operation at uprated conditions will increase the RRC pump suction temperature by approximately one degree Fahrenheit which is also bounded by the system design temperature.

The pump speed and flow control valve position runback functions affected by power uprate and ELLL will be changed. The new cavitation interlock set point is 11 degrees Fahrenheit. The new flow control valve run back set point for the uprated power condition corresponds to a core flow of 48% of rated. The licensee concluded that the changes due to power uprate and ELLL are small and are bounded by the RRC design basis.

The licensee should commit to performing power uprate start up testing on the RRC system to demonstrate flow control over the entire pump speed range to enable a complete calibration of the flow control instrumentation including signals to the Process Computer. As stated in NEDO-31897, these tests should also assure no undue vibration occurs at uprate or ELLL conditions. In a letter dated January 6, 1994, the licensee committed to do start up testing after making the power uprate modifications, and prior to operating in the ELLL region. This commitment is acceptable to the staff.

3.7 Main Steam Isolation Valves (MSIVs)

The main steam isolation valves (MSIVs) have been evaluated by the licensee, and are consistent with the bases and conclusions of the generic evaluation. Increased core flow alone does not change the conditions within the main steam lines, and thus cannot affect the MSIVs. Performance will be monitored by surveillance requirements in the Technical Specification to ensure original licensing basis for MSIV's are preserved. This is consistent with the generic evaluation in NEDO-31894, and is acceptable to the staff.

3.8 Reactor Core Isolation Cooling System (RCIC)

The reactor core isolation cooling system (RCIC) provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system. The RCIC system has been evaluated by the licensee, and is consistent with the bases and conclusions of the generic evaluation. The recommendations of GE SIL 377 have been implemented at WNP-2 and the licensee has committed to additional testing to address all aspects of GE SIL 377. These tests will be conducted during power ascension testing for power uprate. This is acceptable to the staff. The staff requires that the licensee provide assurance that the RCIC system will be capable of injecting its design flow rates at the conditions associated with power uprate. Additionally, the licensee must also provide assurance that the reliability of this system will not be decreased by the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads.

3.9 Residual Heat Removal System (RHR)

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR system is designed to operate in the low pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The effects of power uprate on these operating modes are discussed in the following paragraphs.

3.9.1 Shutdown Cooling Mode

The operational objective for normal shutdown is to reduce the bulk reactor temperature to 125°F in approximately 20 hours, using two RHR loops. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. This increased time is judged to be insignificant.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," states that cold shutdown capability (200°F reactor fluid temperature) should be accomplished within 36 hours. FSAR Section 15.2.9 indicates that cold shutdown can be reached in a much shorter time even considering the availability of only one RHR heat exchanger. For power uprate, licensee analysis of the alternate path for shutdown cooling based on the criteria of Regulatory Guide 1.139 shows that the reactor can be cooled to 200°F in less than the 36-hour criterion.

3.9.2 Suppression Pool Cooling Mode

The functional design basis for suppression pool cooling mode (SPCM) stated in the FSAR is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate, since the peak suppression pool temperature analysis by the licensee confirms that the pool temperature will stay below its design limit at uprated conditions.

3.9.3 Containment Spray Cooling Mode

The containment spray cooling mode provides water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during post-accident conditions. Power uprate increases the containment spray temperature by only a few degrees. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure since these parameters reach peak values prior to actuation of the containment spray.

3.10 REACTOR WATER CLEANUP (RWCU) SYSTEM

The RWCU system pressure and temperature will increase slightly as a result of power uprate. The licensee has evaluated the impact of these increases and has concluded that uprate will not adversely affect RWCU system integrity. The cleanup effectiveness of the RWCU system may be slightly diminished as a result of increased feedwater flow to the reactor; however, the current limits for reactor water chemistry will remain unchanged with power uprate.

4.0 ENGINEERED SAFETY FEATURES

4.2 Emergency Core Cooling Systems (ECCS)

The effect of power uprate and the increase in RPV dome pressure on each ECCS system is addressed below. Also as discussed in the FSAR, compliance to the NPSH requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig and the maximum expected temperature of pumped fluids. The pumps are assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit (212 degrees Fahrenheit). Assuming a LOCA occurs during operation at the uprated power, the suppression pool temperature will remain below its NPSH limit. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

4.2.1 High Pressure Core Spray (HPCS)

The HPCS system was evaluated by the licensee and is consistent with the bases and conclusions contained in the generic evaluation for power uprate. This is acceptable to the staff.

4.2.2 Low Pressure Core Injection System ((LPCI) mode of RHR)

The hardware for the low pressure portions of the RHR are not affected by power uprate. The upper limit of the low pressure ECCS injection setpoints will not be changed for power uprate, therefore the low pressure portions of these systems will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. In addition, the RHR system shutdown cooling mode flow rates and operating pressures will not be increased. Therefore, since the system do not experience different operating conditions due to power uprate, there is no impact due to power uprate. This is consistent with the bases and conclusions of the generic power uprate evaluation.

4.2.3 Low Pressure Core Spray (LPCS) System

The hardware for the low pressure core spray are not affected by power uprate. The upper limit of the low pressure core spray injection setpoints will not be changed for power uprate, therefore the low pressure portions of this system will not experience any higher pressures. The licensing and design flow rates of the low pressure ECCS will not be increased. Therefore, since this system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. Also, the impact of power uprate on the long term response to a LOCA will continue to be bounded by the short term response. The LPCS is bounded by the generic evaluation.

4.2.4 Automatic Depressurization Systems (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA with HPCS failure. This function allows low pressure coolant injection (LPCI) and core spray (CS) to flow to the vessel. The ADS initiation logic and ADS valve control are adequate for uprate. Plant design requires a minimum flow capacity equivalent to 5 SRVs/ADS valves as shown in the licensee analysis for SRV setpoint tolerance and out-of-service analysis to be discussed later in this evaluation. ADS initiates on Low Water Level 1 and a signal that at least one LPCI or LPCS pump is running with permissive from Low Water Level 3. ADS is activated following a maximum time delay of 120 seconds, after the initiating signals if these conditions are met. The ability to perform these functions is not affected by power uprate.

4.3 ECCS Performance Evaluation

The emergency core cooling systems (ECCS) are designed to provide protection against hypothetical loss-of-coolant accidents (LOCAs) caused by ruptures in the primary systems piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Appendix K. The Siemens Nuclear Power (SNP) 8x8 and 9x9-9x fuel, used in WNP-2 was analyzed by the licensee (Reference 5) with the NRC-approved methods. The results of the ECCS-LOCA analysis using NRC-approved methods are discussed in the following paragraphs.

The licensee used the staff approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The S/G-LOCA analysis for WNP-2 was performed by the licensee with SNP 8x8 and 9x9-9x fuel in accordance with NRC requirements in NEDC-32115P and demonstrates conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K PCT as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases. The WNP-2 specific analysis was performed at uprated power and the bounding ELLL region using a conservatively high Peak Linear Heat Generation Rate (PLHGR) and a conservatively low Minimum Critical Power Ratio (MCPR). In addition, some of the ECCS parameters were conservatively established relative to actual measured ECCS performance. The nominal (expected) PCT is 992°F. The statistical Upper Bound PCT is below 1440°F. The Licensing Basis PCT for WNP-2 is 1440°F which is well below the acceptance criteria of 10 CFR 50.46 PCT limit of 2200°F. The analysis also meets the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in Table 4-2 of the WPPSS Licensing Topical Report. Therefore, WNP-2 meets the NRC S/G-LOCA licensing analysis requirements.

The licensee also reevaluated the ECCS performance for single loop operation (SLO) using the S/G - LOCA methodology. The DBA size break is also limiting for SLO. Using the same assumptions in the S/G - LOCA calculation with no

MAPLHGR reduction, yields a calculated nominal and Appendix K PCT of 1184°F and 1504°F, respectively. Since the PCT was below the 10 CFR 50.46 limit of 2200°F, the licensee claimed that no MAPLHGR reduction is required for SLO. The staff asked the licensee to reconcile the fact that the S/G - LOCA analysis PCT results for SLO were higher than those presented for two loop operation, and no statistical analysis of the Upper Bound PCT had been provided for this case. The licensee reviewed this staff question, and has stated in a letter dated January 6, 1994 that the Upper Bound PCT for WNP-2 is 1450°F which is below the 1600°F limit. The current WNP-2 T/S applies a 0.01 adder to the SLMCPR when in SLO due to increased uncertainties. This is acceptable to the staff.

The licensee also evaluated the applicability of the S/G-LOCA methodology to WNP-2 which operates with Siemens Nuclear Power (SNP) 8x8 and 9x9-9x fuel. The dimensions and characteristics of the SNP fuel are similar to GE fuels. The MAPLHGR and PLHGR values used in the WNP-2 analysis are based on inputs for the SNP 8x8 and 9x9-9x fuel shown on Figures 5-2 through 5-5 in NEDC-32115P, respectively. The S/G - LOCA analysis is valid for fuel designs with comparable geometry and for MAPLHGR and PLHGR values less than or equal to those shown in the previously mentioned figures in NEDC-32115P. Since the geometry and characteristics of the SNP fuel used in WNP-2 are similar to those of for a typical GE BWR plant, the S/G-LOCA methodology is applicable to WNP-2 with SNP fuel.

The impact of Increased Core Flow (ICF), up to 115 Mlb/hr, on LOCA results was evaluated at the 3629 MWt power level using S/G-LOCA methodology for WNP-2. For a DBA recirculation line break with the same single failure (HPCS diesel) and using the same Appendix K and nominal assumptions the results show a decrease in the nominal PCT when compared to the base case. This decrease in PCT for the nominal ICF case is due to (1) the better heat transfer during flow coast-down from the higher initial flow, and (2) less subcooling in the downcomer which results in reduced break flow and later core uncovery.

9.0 REACTOR SAFETY PERFORMANCE FEATURES

9.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant cause by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. The licensee will use its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients. The limiting events for WNP-2 were identified. These are the same as those in the generic report on power uprate. The generic guidelines also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. Representative changes in core CPR's for analyzed were provided; however, specific core operating limits will be supplied for each specific fuel cycle. The power uprate with ELLL operation were presented for a representative core using the GEMINI transient analysis methods listed in the generic report.

The Safety Limit Minimum Critical Power Ratio (SLMCPR) will be confirmed for each operating fuel cycle, at the time of the reload analysis, using the NRC approved SNP methodology.

The limiting transients for each category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The results from these analyses developed the licensing basis for transient analyses at uprated power with ELLL operation. The limiting transient results were presented in the licensee submittal in Table 9-2. These were the applicable transients as specified in the generic power uprate guidelines report (NEDC-31897). Cycle specific analyses will be done at each reload and will be a part of the Core Operating Limits Report (COLA) developed by the licensee. This is acceptable to the staff and will be reviewed as part of the licensee's reload submittal.

9.3 SPECIAL EVENTS

9.3.1 Anticipated Transients Without Scram (ATWS)

A generic evaluation for the ATWS events is presented in Section 3.7 of Supplement 2 of the generic report (NEDC-31984) for BWR/5 reactors. This evaluation concludes that the results of an ATWS event are acceptable for the fuel, reactor pressure vessel (RPV), and that the containment response is also acceptable for a power uprate of 4.3%. The WNP-2 power increase is 4.9%, which is 0.6% above the generic evaluation. To provide assurance that the generic results will be met for WNP-2, the licensee performed a WNP-2 specific ATWS analysis for a 10% power uprate for the limiting transients. The results of this analysis for the ATWS event are acceptable for the fuel and RPV, and the containment response is also acceptable for a 4.9% power uprate. The licensee had GE perform a cycle specific analysis with the major parameters and characteristics of the SNP fuel; the results were bounded by the generic ATWS analysis, therefore, the plant's response to an ATWS event is acceptable.

9.3.2 Station Blackout (SBO)

The WNP-2 SBO plant responses were evaluated at a steam flow increase of a 110% for power uprate. This corresponds to an increase of reactor thermal power of 3629 Mwt from 3323 Mwt. The WNP-2 response to a postulated SBO uses the RCIC and HPCS for core cooling. A coping evaluation was performed to demonstrate performance for four hours based on HPCS with backup provided by the RCIC system. However, the RCIC system is the preferred source for initial operation. No changes to the systems or equipment used to respond to a SBO are necessary due to power uprate. The analysis was done at uprate and ELLL operating conditions. The suppression pool temperature remained within design conditions, therefore all equipment that takes suction from the suppression pool will continue to operate when power is restored.

The evaluation assumes a reactor power of 3629 Mwt at an operating pressure of 1035 psia. The individual considerations evaluated for power uprate included the following: the regulatory basis; the event scenario; condensate inventory and reactor coolant inventory; station battery load; compressed air supply; and loss of ventilation to the control room, reactor protection system rooms and switchgear rooms, HPCS pump and auxiliary rooms, RCIC room, containment, suppression pool and spent fuel pool. The SBO analysis is acceptable to the SRXB staff. However, there are other technical issues that must be evaluated by other branches of the technical staff.

SAFETY/RELIEF VALVE (SRV) SETPOINT TOLERANCE AND OUT-OF-SERVICE ANALYSIS

The licensee also submitted an analysis (Reference 6) to support an increase in SRV setpoint tolerance and number of SRV's allowed out-of-service (OOS) for WNP-2. The report was in support of modifying the current in-service opening pressure setpoint tolerance from +1%/-3% to ± 3%, and allowance for up to two SRV's OOS. The analysis was performed assuming a thermal power level of 3629 Mwt, corresponding to 104.1% of the uprated power level of 3486 Mwt. The analysis addresses a core flow operating range from 108.5 Mlbm/hr to 115 Mlbm/hr at the thermal power of 3486 Mwt and an operating pressure of 1035 psia. This is applicable for the WNP-2 power uprate with ELLL operation. A setpoint tolerance of +3% above the nominal SRV setting was assumed in the analysis, where appropriate.

The SRV setpoint tolerance and out-of-service analysis integrates the analysis supporting over pressure protection, containment response, SRV Load Definition, Emergency Core Cooling System (ECCS) Loss-of-Coolant Accident (LOCA) analysis events, Abnormal Operational Occurrences (AOO), Anticipated Transients Without Scram (ATWS), High Pressure Core Spray/Reactor Core Isolation Cooling (HPCS/RCIC) performance, Standby Liquid Control System (SLCS) performance, common mode failure concerns, and Appendix R events.

The SRVs provide three main protection functions : 1). Overpressure relief operation (power relief mode); the valves open automatically to limit a vessel pressure rise. 2). Overpressure safety operation (spring safety mode) to prevent reactor vessel overpressurization. 3). Depressurization operation; the Automatic Depressurization System (ADS) valves open automatically as part of the ECCS, for events involving small breaks in the reactor pressure boundary.

General Electric submitted a generic topical report (Reference 7) for NRC review that supports changes to the current requirements for SRV setpoint tolerance changes. The staff prepared a Safety Evaluation Report (SER) that approved relaxing the setpoint tolerance limit from ±1% to ±3%, when plant specific analyses are submitted to support these changes (Reference 8). Each licensee implementing these changes must provide certain plant specific analyses to include the following:

1. Transient analysis of all AOOs listed in Reference 7 utilizing the ± 3% setpoint tolerance for the SRVs and utilizing staff approved methodology.

2. Analysis of the design overpressure event using the 3% tolerance limit to confirm that the vessel peak pressure does not exceed the ASME pressure vessel code upset limit.

3. The plant specific analysis in Items 1 and 2 above should assure that the number of SRV's included in the analysis corresponds to the number required to be operable in the Technical Specifications (T/S).

4. Reevaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.) motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the 3% limit.

5. Evaluation of the ± 3% tolerance on any plant operating mode such as increased core flow, extended operating domain (ELLL), and power uprate must be completed.

6. Evaluation of the effect of the 3% tolerance limit on the containment response during LOCA's and the hydrodynamic loads on the SRV discharge lines must be completed.

The licensee provided the plant specific analyses needed to support the changes in Reference 6. The new limits, as well as the present limits, are presented in Table 1-1 of the reference.

The results of the overpressure analysis; MSIV closure with indirect flux scram, with six SRVs OOS, and an initial operating pressure of 1050 psia, resulted in a peak reactor pressure of 1335 psia. This meets the ASME code allowable of 1375 psia, and is acceptable.

The most limiting thermal transient is the Load Rejection with Bypass Failure (LRNBP) event coincident with End-of-Cycle (EOC) Recirculation Pump Trip (RPT) out-of-service. This event is analyzed to determine the operating limit MCPR to assure safe plant operation. Since the peak MCPR occurs prior to the SRVs opening the setpoint tolerance has no impact on thermal limits.

The ECCS-LOCA analysis assumed the SRVs opening at the +3% above the nominal setpoints with four SRVs OOS. The break spectrum results show that for the large breaks the reactor vessel depressurizes through the break, while smaller breaks require the ADS for depressurization. The small break analysis assumed two ADS valves OOS. The results shown in Table 4-2 (REF.6) show that the PCT for small breaks meets the 10CFR 50.46 limit with various single failures assumed.

The HPCS and RCIC performance was evaluated for Loss-of-Feedwater (LOFW) events. The HPCS was evaluated assuming no RCIC and low pressure ECCS pumps are available, six SRVs OOS with relief mode 30 psi above the new nominal setpoints, SRVs close at 50 psi below the opening pressure, and the reactor initially at Level 3. The results show that the water level remains above the active fuel, which is acceptable. The RCIC was evaluated with the same assumptions as the HPCS analysis, except the HPCS and low pressure ECCS pumps were failed. These results also show no core uncovery, which is acceptable. Even though six SRVs were assumed OOS, there are other limiting concerns that will allow less OOS SRVs. These limiting concerns will govern the number of SRV's that will be allowed OOS.

The containment response to these assumptions are evaluated in another section. [Plant Systems Branch should supply this evaluation]

Other regulatory concerns such as alternate shutdown system performance-Appendix R (fire protection event) was also evaluated by the licensee. [Plant Systems Branch should supply this evaluation]

The SLCS performance was also evaluated by the licensee. The operating pressure range for SLCS is based on the lowest setpoint SRV available in the relief mode. The ability of the SLCS pumps to inject its design flow at higher pressures is not affected because these pumps are positive displacement type pumps and are designed to provide constant flow with increasing pressure. The licensee stated that the electric motors to drive these pumps should have sufficient horsepower margin to meet the pump power requirements.

The effect of SRVs OOS was also evaluated for the limiting ATWS event. The reactor is eventually shutdown by the SLCS. Initially the power is reduced by the recirculation pump trip (RPT) signal. The maximum vessel pressure occurs during the initial portion of the event. After the RPT, and subsequent actuation of the SRVs the event is terminated. With four SRVs OOS the peak pressure was calculated to be 1467 psig which is below the ASME code service level C value of 1500 psig. Thus it is concluded that four valves OOS does not violate the overpressurization criterion for an ATWS event.

The common mode failure aspects of allowed SRVs to be OOS was evaluated by the licensee. The condition with a large number of SRVs inoperable may be indicative of a common mode failure mechanism. This is the limiting concern that determines how many SRVs will be allowed OOS. It is recommended that as soon as two SRVs are found to be inoperable, the root cause of the failure should be determined immediately and verified that a common mode failure possibility does not exist. The analyses performed by the licensee for the SRV OOS evaluation assumed that common mode failures were not the cause of the SRV inoperability.

The staff concludes that a setpoint tolerance increase from +1%/-3% to ±3%, and allowing up to two SRVs to be out-of-service was properly analyzed by the licensee. These changes should not have the potential to significantly increase the probability or consequences of an accident previously analyzed, or create a new or different kind of accident then those previously analyzed, and should not exceed any regulatory requirements that may cause a significant reduction in the margin of safety for the WNP-2 plant.

REFERENCES

1. Letter to the NRC from J.V.Parrish, WPPSS, requesting an amendment to the WNP-2 operating license and Technical Specifications to increase licensed power level with Extended Load Line Limit, and a change in safety relief valve setpoint tolerance, dated July 9, 1993.

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