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July 30,1999

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Subject: River Bend Station
Docket No. 50-458
License No. NPF-47
License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station

File Nos.: G9.5, G9.42

RBEXEC-99-025
RBF1-99-0215
RBG-45077

Ladies and Gentlemen:

According to the provisions of Sections 50.90 and 50.4 of Title 10 of the Code of Federal Regulations (CFR), Entergy Operations, Inc. (EOI), hereby applies for amendment of Facility Operating License No. NPF-47 and Appendix A - Technical Specifications, for River Bend Station (RBS). The proposed change will extend operation of RBS from its current licensed power level of 2894 megawatts thermal (MWt) to an uprated power level 3039 MWt, an increase of five percent. The proposed changes have been developed using generic guidelines for boiling water reactors (BWR) power uprates described in General Electric (GE) reports NEDC-31897P-A, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," May 1992, and NEDC-31894P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," July 1991; and Supplements, hereinafter referred to as "Reference 1" and "Reference 2," respectively.

Enclosure 1 is an oath and affirmation executed in accordance with 10 CFR 50.30(b). Enclosure 2 is a detailed description of the specific proposed changes for implementing uprated power operation, with the technical bases for each of the changes. Enclosure 3 contains the affected page listing and a copy of the affected Operating License and TS pages marked to show the proposed changes. Enclosure 4 contains the affected pages

A001

change: NRC PDR

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listing and a copy of the appropriate Operating License and TS pages incorporating the proposed changes.

Enclosure 5 is the requisite Environmental Assessment (EA) and Finding of No Significant Impact (FONSI). The EA/FONSI was prepared in accordance with the National Environmental Policy Act (NEPA) and EOI's implementing procedures. It addresses specific issues and potential environmental impacts associated with power uprate at RBS.

An affidavit executed by GE supporting a request for proprietary treatment of portions of the submitted information in accordance with 10 CFR 2.790(b)(1) is provided in Enclosure 6. Enclosure 7 contains the detailed plant-specific submittal information required by the generic guidelines (References 1 and 2). The portions of the report that are proprietary should be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4). Section 11.4 of Enclosure 7 contains a non-proprietary, no significant hazards evaluation for the proposed changes. Enclosure 8 lists the commitments made in this submittal.

Two items are not included in Enclosure 7, but are discussed in detail in Enclosure 2. The items include a change in the safety relief valve (SRV) setpoint tolerance to $\pm$ 3% and the treatment of non-limiting transient events. We would also like to note that EOI and GE performed a review of approximately 400 NRC Requests for Additional Information on previously approved power uprate submittals. As appropriate, the results of this review were incorporated into the GE report (Enclosure 7).

Actual design modifications required by power uprate largely involve setpoint changes. Major physical modifications are limited. A new high-pressure turbine rotor was installed during Refueling Outage 8, and new springs will be installed in the main steam SRVs. These modifications may be implemented during a Fall 2000 outage or by startup from Refueling Outage 10 (scheduled to begin September 2001). We are currently assessing the possibility of implementing a portion of the uprate by flow-only increases, in the Fall 2000, after receipt of NRC's approval of the changes herein. If we elect to do a partial uprate prior to Refueling Outage 10, we will determine what portion of the TS changes must be implemented for the incremental increase and determine if further NRC review is necessary. Accordingly, we request that the NRC complete its review of the proposed changes by September 2000. We request that the changes be effective upon issuance of the license amendment, with full implementation by Cycle 11.

We would like to meet with the NRC reviewers by October 1999 to discuss the material enclosed. We will coordinate a meeting through the RBS NRC Project Manager, Robert Fretz. If you have any questions about this change, please contact Barry Burmeister at (225) 381-4148.

Sincerely,

Randall K. Edington

Enclosures
RKE/RJK/bmb

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References:

1. NEDC-31897P-A, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," May 1992.
2. NEDC-31894P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," July 1991; and Supplements.

ENCLOSURE 1

BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-47

DOCKET NO. 50-458

IN THE MATTER OF

ENTERGY GULF STATES, INC.

ENTERGY OPERATIONS, INC.

AFFIRMATION

I, Randall K. Edington, state that I am Vice President – River Bend Station, Entergy Operations, Inc. (EOI), that on behalf of EOI, I am authorized to sign and file with the U. S. Nuclear Regulatory Commission, this River Bend Station License Amendment Request (LAR) 1999-15, consisting of proposed changes to the River Bend Station Technical Specifications, that I signed this letter as Vice President - River Bend Station, for Entergy Operations, Inc.; and that the statements made and the matters set forth herein are true and correct to the best of my knowledge, information, and belief.

Randall K. Edington

Randall K. Edington

STATE OF LOUISISANA
PARISH OF WEST FELICIANA

SUBSCRIBED AND SWORN TO before me, a Notary Public, commissioned in the Parish and State above named, this 29th day of July, 1999.

(SEAL)

Claudia J. Hurst

Claudia Hurst
Notary Public

ENCLOSURE 2

**ENTERGY OPERATIONS, INC.
RIVER BEND STATION
DOCKET 50-458/LICENSE NO. NPF-47
LICENSE AMENDMENT REQUEST 99-15**

Licensing Document Involved

The proposed changes affect the River Bend Station (RBS) Operating License and Technical Specifications and associated Bases sections identified in Enclosure 3.

Background

The proposed amendment consists of a number of changes that will permit uprated power operation for RBS. RBS is a General Electric (GE) boiling water reactor (BWR), model BWR-6, with a Mark III Containment. The implementation of a power uprate at RBS is similar to that implemented by other BWR plants of similar design to RBS, as approved by the NRC. The enclosed evaluation and proposed changes follow the guidelines established during the approval of such similar plants.

The RBS design and safety analyses were performed for a maximum power level of approximately 3015 megawatts-thermal (MWt). This power level corresponds to about 105 percent of the rated steam flow for RBS. It corresponds to approximately 104.2 percent of the current licensed rated power level (2894 MWt). EOI is proposing an amendment to the RBS Operating License and Technical Specifications to operate RBS at power levels up to 105 percent of the current rated power level (i.e., approximately 0.8 percent above the previously analyzed "stretch power" level). The licensed power level would increase from 2894 MWt to 3039 MWt by these proposed changes.

The analyses and evaluations supporting these changes were completed using the guidelines in Topical Report NEDC-31897P-A, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," (Reference 1), approved by NRC letter dated September 30, 1991. Resolution of generic issues associated with power uprate was addressed in Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," (Reference 2), approved by NRC letter dated July 31, 1992.

The increase in electrical output is accomplished primarily by generation and supply of higher steam flow to the turbine generator. Continuing improvements in the analytical techniques (e.g., computer codes and data) throughout several decades of BWR safety technology, plant performance feedback, and improved fuel and core design have resulted in a significant increase in the margin between calculated safety analyses results and plant licensing limits. This available safety analysis margin, combined with the excess capability of as-designed equipment, systems, and components, allow for an increase of 5 percent in the full power rating of RBS without the need to perform major Nuclear Steam Supply System or Balance-of-Plant hardware modifications. The full power level can be increased safely, with installed systems and equipment, which are capable of performing their required functions at the uprated conditions.

The method for achieving higher power is to extend the power/flow map by increasing the core flow along the pre-uprate limiting flow control line. However, maximum recirculation flow will not exceed the pre-uprate limit.

The plant-specific safety analyses to support this change are documented in GE's report for RBS (Enclosure 7). This report demonstrates that RBS can operate safely with a 5 percent increase in maximum reactor thermal power and an associated 30 pounds per square inch (psi) increase in the operating reactor vessel pressure. This includes a corresponding increase in main turbine inlet steam flow and the corresponding increases in flow, temperature, pressure, and capacity in supporting systems and components.

Proposed Changes

Enclosure 3 is a table that summarizes the TS and Bases changes needed to support the power uprate. These changes are also identified in Table 11-1 of Enclosure 7. Enclosure 3 contains the affected page listing and copies of the appropriate Operating License and TS pages for RBS appropriately marked-up to show the proposed changes. Each operating license and TS change is evaluated below:

- **Operating License**

License Condition 2.C.(1), "Maximum Power Level, will be changed from "2894 megawatts thermal" to "3039 megawatts thermal" as the new value for 100 percent rated power.

- **Technical Specifications (TS)**

1. Rated Thermal Power is increased from 2894 MWt to 3039 MWt in Section 1.1, "Definitions," of the RBS TS.

Evaluation

This increase and redefinition of rated thermal power for RBS follows the generic guidelines of NEDC-31897P-A (Reference 1) for GE BWR power uprates. NEDC-31897P-A provides generic licensing criteria, clarified methodology, and a defined scope of analytical evaluations and equipment review to be performed to demonstrate the ability to operate safely at the uprated power level. TS parameter values, which are expressed as a percentage of rated reactor thermal power or steam flow, were not changed because the uprated values were used in the bounding analyses and evaluations required by Reference 1, unless otherwise specified in this submittal.

Enclosure 7 provides results of evaluations supporting the proposed uprated power operation consistent with the methodology presented in Reference 1. The GE report concludes that an uprated power rating of 3039 MWt can be achieved without significant impact on equipment or safety analyses.

2. The Thermal Power Safety Limit of TS 2.1.1.1will be lowered from 25 to 23.8 percent. This change is to maintain the same power value with respect to absolute thermal power, flow, and pressure.

Evaluation

As discussed in Section 9.1 of Enclosure 7, the Thermal Power Safety Limit is based upon generic analyses, with BWR-6 being the limiting design. To maintain the basis for thermal transient analyses in design, the Reactor Thermal Power (RTP) is reduced to 23.8 %. Decreasing this limit assures continued compliance with all safety limits at the uprated conditions. With this new value, the results of fast transients in the power range are not significantly affected by power uprate because of the protection provided by off-rated power dependent limits. This change therefore maintains the current threshold value at which thermal limits are to be monitored in terms of absolute power. Therefore, this change will also maintain consistency with the original conditions in the transient analyses contained in the RBS Updated Safety Analysis Report (USAR), Chapter 15.

Reference to thermal power appears in TS 1.4, Examples 1.4-2 and 1.4-3, and TS 3.2.1, 3.2.2, 3.2.3, 3.3.1.1, 3.4.3.1 and 3.7.5.

3. The Reactor Steam Dome Pressure used to determine Control Rod Scram times on Table 3.1.4-1 will increase from 1050 to 1059 psig.

Evaluation

As discussed in Section 2.5.1 of Enclosure 7, the SCRAM time requirements are unchanged for the uprated conditions. The current upper limit on Reactor Steam Dome Pressure in Table 3.1.4-1 will continue to support operation at the uprated conditions. This revision to the peak test pressure will also bound expected surveillance test pressure conditions.

4. The Control Rod Drive (CRD) charging water header minimum pressure value will be changed from 1520 to 1540 psig. This change will maintain scram time design.

Evaluation

As discussed in Section 2.5.1 of Enclosure 7, this pressure is increased to maintain pre-uprate margins to the TS Surveillance limits. The CRD system will maintain current design reactor reactivity insertion time design limits. Impact on the structural and functional integrity of the CRD system and reactor vessel were evaluated. The change will affect TS 3.1.5 (and Bases), TS 3.9.5 (and Bases), and TS 3.10.8 (and Bases).

5. The Standby Liquid Control (SLC) system Boron-10 enrichment and concentration criteria contained in TS 3.1.7 will be increased.

Evaluation

As discussed in Section 9.3.1 of Enclosure 7, the Anticipated Transients Without Scram (ATWS) mitigation requirements defined in 10 CFR 50.62 were analyzed for the uprated conditions to demonstrate compliance with the ATWS acceptance criteria. These analyses were performed in accordance with accepted methods and requirements. Results of the analyses show RBS continues to meet ATWS acceptance criteria by increasing SLC system Boron-10 enrichment (E) and concentration (C) criteria from 413 to 570. Note; the maximum concentration is limited by SR 3.1.7.5. This limit is included in TS 3.1.7, Action A, and SR 3.1.7.3.

6. The surveillance test discharge pressure for the standby liquid control pump is increased from 1220 psig to 1250 psig. This value appears in SR 3.1.7.7 and the corresponding Bases Section B 3.1.7.

Evaluation

As discussed in Section 6.5 of Enclosure 7, the surveillance test pressure is based on the maximum SLCS injection pressure, including allowances for system test inaccuracies. Therefore, the SLC pump discharge pressure is increased from the pre-power uprate value of 1220 psig to 1250 psig to account for the increase in system injection pressure at power uprate conditions. Increasing the test pressure by 30 psi assures the continued capability of these positive displacement pumps to deliver design rated flow at operating pressures expected at the uprated conditions. This change, therefore, maintains the original intent of SR 3.1.7.7.

7. The allowable value for the Reactor Vessel Steam Dome Pressure - High Scram setpoint is increased from 1079.7 psig to 1109.7 psig. The chosen allowable value is acceptable based on the analytical limit for the parameter and the reactor vessel design.

Evaluation

As discussed in Section 5.3.1 and consistent with the analytical limit in Table 5-1 of Enclosure 7, the reactor vessel steam dome high pressure scram limit is increased because the steam dome operating pressure is increased. Operating pressure for uprated power is increased to assure that satisfactory reactor pressure control is maintained. The operating pressure was chosen on the basis of steam line pressure drop characteristics and the steam flow capability of the turbine. Satisfactory reactor pressure control requires an adequate flow margin between the uprated operating conditions and the steam flow capability of the turbine control valves at their maximum stroke. An operating dome pressure of 1055 psig, which is 30 psi higher than the current operating dome pressure, is expected. Therefore, the high pressure scram is increased appropriately by 30 psi to preserve existing margins to reactor scram.

The high pressure scram terminates a pressurization transient not terminated by direct scram or high neutron flux scram. The setting is maintained above the nominal reactor vessel operating pressure and below the specified analytical scram limit used in the safety analyses. The revised high pressure scram setpoint will preserve the hierarchy of pressure setpoints. This means that the high pressure scram setpoint will remain below the opening setpoint of the Main Steam Relief Valves (MSRV). The MSRV nominal setpoints are also increased 30 psi, as discussed in proposed change "9" below. This hierarchy of setpoints provides assurance that there is a low probability of opening more than one MSRV without scram intervention.

The revised Reactor Vessel Steam Dome Pressure - High scram trip allowable value is consistent with the discussion in Section 5.3.1 and analytical limit in Table 5-1. The new calculated limit is determined by the use of the GE setpoint Methodology as described in Section 5.1 and Reference 3 of Section 5 in the GE report. The Allowable Value appears in Section 3.3.1.1, Table 3.3.1.1-1, Function 3, in the RBS TS.

8. The allowable value for the ATWS-RPT Reactor Steam Dome Pressure - High setpoint is increased from 1135 psig to 1165 psig, a 30 psi increase. The Allowable Value appears in TS SR 3.3.4.2.4.

Evaluation

As discussed in Section 5.3.2, and consistent with the analytical limit in Table 5-1 of Enclosure 7, the ATWS-RPT high pressure setpoint initiates a trip of the recirculation pumps, thereby adding negative reactivity following events in which a scram does not (but should) occur. As discussed in Section 5.3.2 of Enclosure 7, the analytical limit for the ATWS-RPT high pressure setpoint was increased 30 psi in the power uprate ATWS safety evaluations to account for the 30 psi increase in vessel operating pressure, MSRV setpoints, etc. The analyses demonstrate that the ATWS criteria are met with the higher analytical limits. Therefore, the allowable value is increased consistent with the analytical limit used in the safety analysis. Raising the ATWS-RPT high pressure setpoint to correlate with the increased operating pressure and analytical limit will tend to prevent unnecessary recirculation pump trips following pressurization transients with reactor scram (e.g., turbine trip or load rejection with bypass). Recirculation pump operation following a scram allows for better mixing of the reactor coolant and reduces thermal stratification in the vessel.

The new calculated limit is determined by the use of the GE setpoint Methodology as described in Section 5.1 and in Reference 3 of Section 5 in the GE report.

9. The MSRV lift setpoints will be increased. The safety function, relief function and the Lo-Lo Set function will need revision.

Evaluation

As discussed in Section 5.3.3 of Enclosure 7, consistent with the increase in nominal reactor dome pressure shown in Table 1-2, and the analytical limit in Table 5-1, the MSRVs are designed to prevent overpressurization of the reactor pressure vessel during abnormal operational transients. The MSRV lift setpoints are increased to accommodate the increase in operating pressure of 30 psi that accompanies power uprate. The increase in MSRV setpoints ensures that adequate margins are maintained so that the increase in dome pressure during normal operation does not result in an increase in the number of unnecessary MSRV actuation's. The setpoint increase also maintains the hierarchy of pressure setpoints described in these proposed changes. Transient evaluations include a positive 3 percent tolerance to the nominal setpoints. As described in Section 3.2 of Enclosure 7, transient peak vessel pressure increases at uprated conditions, but remains below the 1375 psig American Society of Mechanical Engineers (ASME) Code limit.

The adequacy of BWR MSRVs to operate at uprated temperatures and pressures has been evaluated generically in Section 4.6 of Reference 2. The reactor operating pressure and temperature increases of less than 40 psi and 5°F, respectively, used in that evaluation bound the uprated operating conditions.

The impact of power uprate on the containment dynamic loads due to MSRV discharge has also been evaluated. As discussed in Section 4.1.2 of Enclosure 7, the vent thrust loads with power uprate were calculated to be less than the loads used in the containment analysis. The effect of power uprate on MSRV air-clearing, the discharge line, the pool boundary pressure, and submerged structure drag loads is also discussed in Section 4.1.2 of Enclosure 7. That discussion concludes that the small increase in the setpoint pressure is within the conservatism in the SRV loads for RBS.

The change in the Relief and Lo-Lo Set (LLS) function setpoints are:

SRVs **Current Values** **Proposed Values**

a. Relief function

Low	1103	+/- 15 psig	1133	+/- 15 psig
Medium	1113	+/- 15 psig	1143	+/- 15 psig
High	1123	+/- 15 psig	1153	+/- 15 psig

b. LLS function

Low - Open	1033	+/- 15 psig	1063	+/- 15 psig
Close	926	+/- 15 psig	956	+/- 15 psig
Medium - Open	1073	+/- 15 psig	1103	+/- 15 psig
Close	936	+/- 15 psig	966	+/- 15 psig
High - Open	1113	+/- 15 psig	1143	+/- 15 psig
Close	946	+/- 15 psig	976	+/- 15 psig

These values appear in SR 3.3.6.4.3 in the TS and in the Bases of TS 3.6.1.6. The new calculated limit is determined by the use of the GE Setpoint Methodology as described in Section 5.1 and in Reference 3 of Section 5 in the GE report. The change in the Safety setpoints are:

Number of SRVs	Current Values		Proposed Values	
7	$\geq$ 1141.7	$\leq$ 1165	1195	+/- 3 %
5	$\geq$ 1156.4	$\leq$ 1180	1205	+/- 3 %
4	$\geq$ 1166.2	$\leq$ 1190	1210	+/- 3 %

These values appear in SR 3.4.4.1 in the TS. The change to the tolerance range of the Safety valves is discussed below.

b. In addition to the changes in the Unit TSs to account for the increased reactor pressure RBS proposes changing the present the -2 / +0 % tolerance on the safety function lift setpoint for the SRVs to +/-3%. This change is consistent with the assumptions of the safety analysis for power uprate and the recommendations of the Licensing Topical Report (LTR), NEDC-31753P.

Evaluation

This change would affect TS Surveillance Requirement 3.4.4.1 and Bases. The RBS Inservice Testing Program (IST) controls the frequency of SRV testing as required by RBS Technical Specifications; therefore, this proposal will also incorporate changes to applicable IST procedures. RBS will incorporate the recommendations of the LTR, NEDC-31753P and the associated SER, by resetting the safety function lift setpoints for all tested valves to within ±1% of the design lift setpoint and increasing the test sample size by two valves for each valve found outside of the ± 3% safety function lift setpoint. RBS will test the SRV in accordance with ASME/ANSI OM-1987, Part 1, or the ASME/ANSI OM as approved by the RBS IST program. The SER on this LTR, accepted that a generic change of SRV setpoint tolerance to ±3% is acceptable, provided certain plant specific analyses are performed.

Each RBS SRV is a Crosby, direct-acting, spring loaded, safety valve with attached pneumatic cylinder for relief mode operation. RBS has a total of sixteen SRVs installed on the four main steam lines. All valves are of the same design, and each valve can be operated in either the safety or the relief mode.

The RBS IST and associated plant procedures ensure testing on each SRV in accordance with ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition, (no Addenda) (Subsections IWA and IWV). Currently, each SRV that is removed during a refueling outage is tested on a test fixture, and is certified for relief mode operation and seat leakage. Also, the SRV is tested for safety mode set pressure, to ensure that the safety function lift setpoint is within -2%, +0% of the set pressure, as required by RBS Technical Specification 3.4.4.1.

NEDC-31753P and the NRC SER on this LTR indicate that a change of SRV setpoint tolerance to ±3% is acceptable, provided certain plant specific analyses are performed. Each of the requested analyses, along with a summary of the RBS specific evaluation, follows.

1. Transient analysis of all abnormal operational occurrences, as described in NEDC-31753P, should be performed utilizing a ±3% setpoint tolerance for the safety mode of spring safety valves (SSVs) and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

   NEDC-31753P, Section 4.3, states that group 3 plants (BWR 5 and 6 design) are not affected by the evaluation of abnormal operational occurrences since this program only proposes changes to the safety mode of actuation. However, RBS abnormal operating occurrences that result in SRV actuation credit operation of five safety mode SRVs, the minimum number required by the RBS Operating License. Therefore, it was necessary for RBS to determine if the proposed setpoint tolerance request would affect any of the previously analyzed abnormal operating occurrences (Reference GESTAR II). Each of these abnormal operating occurrences were analyzed using safety function lift setpoints at the proposed ±3%. The release to the environment remains below 10 CFR 20 limits, there are no cladding failures (MCPR > SLMCPR), NSSS stresses remain below the Code Allowables, and containment stresses remain below the Code Allowables. These analyses were conducted at the uprated conditions.

2. Analysis of the design basis overpressurization event using the ±3% tolerance limit for the SRV setpoint is required to confirm that the vessel pressure does not exceed the ASME Code Pressure Vessel upset limit.

   The RBS design basis (worst case) overpressurization event is a closing of all main steam isolation valves while the reactor is operating at 102% rated power and 107% rated core flow (Increased Core Flow). Reactor scram on MSIV position is assumed to fail, so the scram is assumed to occur on high neutron flux. The BWR 6 design meets the ASME Section III Article NB 7542 allowance that up to half of the installed SRVs may actuate on the auxiliary actuating device (relief mode); however, it should be noted that the RBS analysis only credits four of the installed SRVs for actuation in the relief mode. The overpressurization analysis credits five SRVs in safety mode and four SRVs in relief mode.

Overpressurization analyses performed for the power uprate project compared to fuel Cycle 8 inputs using SRV opening pressures of ±3% showed margin to the ASME Code limit of 1375 psig as discussed in Section 3.2 of Enclosure 7. Based on these results and the relative insensitivity of the results to the fuel design parameters, future analyses are expected to yield peak pressures with similar margin to ASME Code limit. As discussed in the NEDC 31753-P SER, future reload safety analyses will bound the proposed ±3% tolerance. These analyses were conducted at the uprated conditions for the evaluation contained in Enclosure 7.

3. The plant specific analysis described in Items 1 and 2 should assure that the number of SSVs, SRVs, and relief valves (RVs) included in the analysis correspond to the number of valves required to be operable in the technical specification. This is discussed in Section 3.2 of Enclosure 7.

   The number of SRVs assumed in the analyses required in items 1 and 2 above is consistent with the Technical Specifications, Limiting Condition of Operation 3.4.4, by crediting operation of only five safety mode SRVs and four relief mode SRVs.

4. Re-evaluation of the performance of high pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves, and vessel instrumentation and associated piping have been completed, considering the ±3% tolerance limit (see Enclosure 7, NEDE-32778P).

   High Pressure Systems

   RBS has three high pressure vessel injection/spray systems: 1) high pressure core spray (HPCS), 2) reactor core isolation cooling (RCIC), and 3) standby liquid control (SLC).

   a) HPCS

      HPCS is an emergency core cooling system designed to deliver sufficient coolant to the reactor core in conjunction with other ECCS systems The HPCS system analysis for the uprate project discussed in section 4.2.1 of Enclosure 7 included the increased SRV setpoint tolerance. The results of this analysis were acceptable, and, adequate margin is maintained.

   b) RCIC

      The evaluation reviewed effects of the proposed change on the system flow, overspeed trip setpoint, and initiation time to rated flow. The RCIC system analysis for the uprate project discussed in section 3.8 of Enclosure 7 included the increased SRV setpoint tolerance. The results of this analysis were acceptable, and adequate margin is maintained.

c) SLC

The SLC system operation is not affected by the SRV safety setpoint tolerance increase as discussed in section 6.5 of Enclosure 7. The pressure used for system performance is based on the SRV relief settings of the system, forwhich the tolerance remains unchanged not the SRV safety settings.

Motor-Operated Valves (MOVs)

As described in the NEDC-31753P SER and the Technical Evaluation Report prepared as part of the NRC's evaluation, consideration should be given to testing MOVs exposed to reactor pressure at higher differential pressures. The MOV analysis for the uprate project discussed in section 4.1.4 of Enclosure 7 included the increased SRV setpoint tolerance. The results of this analysis indicate that a number of valves require calculation revisions, actuator adjustments and/or physical changes to ensure satisfactory performance as discussed in Section 4.1.4 of Enclosure 7.

Vessel Instrumentation

The design pressure of process piping is adequate to provide margin above the pressure resulting from an increase in SRV setpoint tolerance. The analysis for the uprate project discussed in section 5 of Enclosure 7 included the increased SRV setpoint tolerance. The evaluation determined that there is an impact on vessel instrumentation piping as a result of the proposed safety setpoint tolerance change. These changes have been evaluated by power uprate.

5. Evaluation of the ±3% tolerance on any plant specific alternate operating modes (e.g., increased core flow, extended operating domain) should be completed. The analysis for the uprate project discussed in Table 1-2 of Enclosure 7 included the increased SRV setpoint tolerance. The results of this analysis were acceptable, and adequate margin is maintained for the following issues: Maximum Extended Load Line Limit Analysis (MELLLA), Single Loop Operation (SLO), Increased Core Flow (ICF) and Feedwater Temperature Reduction (FWTR).

6. Evaluation of the effect of the 3% tolerance limit on the containment response during loss of coolant accidents and the hydrodynamic loads on the SRV discharge lines and containment was completed. The analysis for the uprate project discussed in section 4.1.2 of Enclosure 7 included the increased SRV setpoint tolerance on the following: Containment Response During Loss of Coolant Accidents (LOCA), SRV Discharge Line Loads including 1) SRV to the first anchor point, 2) discharge line downstream of the SRV anchors, and 3) quenchers and Containment Hydrodynamic Loads. The results of this analysis were acceptable, and adequate margin is maintained.

The SER for NEDC-31753P provided four conclusions and limitations. The RBS proposal agrees with each conclusion and limitation with the exception that the valve test frequency will be in accordance with the River Bend Inservice Testing Program. In summary, the design effects of this change to the SRV setpoint tolerance have been included in the evaluations of the plant as part of Enclosure 7. The cycle specific changes are included in each cycle design as part of the current process.

Based on the above evaluation, EOI has concluded that this change is in accordance with approved methods and the results are within acceptable bounds. Therefore, this change is within the significant hazards considerations included in Enclosure 7.

10. The upper and lower bounds on reactor pressure, for purposes of performing reactor core isolation cooling (RCIC) pump flow rate surveillance tests at high pressure, are increased by 30 psi.

Evaluation

As discussed in Section 3.8, and consistent with the analytical limit in Table 5-1 of Enclosure 7, the reactor operating pressure range for RCIC surveillance tests at high pressure is increased to correspond with the increase in normal reactor operating pressure that accompanies power uprate. The change is needed to provide a more appropriate test range for the higher uprate reactor operating pressure. The requested changes will allow the quarterly demonstration of RCIC capability to be performed at normal reactor operating pressures, which meets the original intent of the TSs. As discussed in Enclosure 7, Sections 3.8, the pre-uprate flow rates remain valid for uprated power conditions.

These values appear in SR 3.5.3.3 and the associated Bases for RCIC in the TS. The change in pressure is from the pre-uprate values of 1045 psig to 1075 psig.

11. A change to the Main Steam Line Flow - High reactor isolation trip in the RBS TSs.

Evaluation

The main steam line flow rate is discussed in Section 5.3.4, the analyses is consistent with the uprated reactor conditions shown in Table 1-2, based on the analytical limits shown in Table 5-1 of Enclosure 7. The results of the main steam line flow analysis show the revised setpoint maintains the 140 % total flow limitation. They also ensure sufficient difference between steam line flow with one MSIV closed for testing and the high flow isolation setpoint to avoid spurious trips.

The results of the RCIC evaluation discussed in Section 5.3.15 continue to support the current limitations on maximum steam flow and will maintain operation consistent with the assumptions of the current safety analysis for power uprate. The RCIC Steam Line Flow High Allowable Value will be conservatively maintained at 135.5 inches of water although there will be a small increase in the RCIC steam flow rate at the uprated conditions.

These calculated limits are determined by the use of the GE setpoint Methodology as described in Section 5.1 and in Reference 3 of Section 5 of the GE report. These changes will modify Table 3.3.6.1-1, Item 1c, "Main Steam Line Flow Allowable Values," of the TS as follows:

Steam Line	Current Value	Proposed Value
A	151.0 psid	190.0 psid
B	161.0 psid	194.0 psid
C	158.0 psid	194.0 psid
D	169.0 psid	194.0 psid

12. A change to the Thermal Power limits of Specification 3.4.1, "Recirculation Loops Operating," during Single Loop Operation from 83% to 79% to maintain analysis assumptions.

Evaluation

As discussed in Section 3.4 of Enclosure 7, the current limitations on maximum core power during Single Loop Operations (SLO) will be maintained. Consequently, the maximum power for SLO will be limited to the previous level. This will result in the uprated limit being reduced by the ratio of the current rated percentage to uprated percentage power values (i.e., 100/105), and decrease the thermal power value by 100/105 for single loop operation (SLO), to maintain the same SLO absolute thermal power range (83% to 79%). This change will maintain operation consistent with the assumptions of the safety analysis for power uprate.

This change will also require associated changes in the Bases of this specification.

13. An increase of 30 psi in the Pressure Isolation Valve (PIV) surveillance test SR 3.4.6.1 of the RBS TSs.

Evaluation

As identified in Section 1.3 of Enclosure 7 the reactor pressure is increased by 30 psi. The increase the reactor pressure will require an increase in the PIV test pressure by the same amount as the nominal reactor dome pressure increase shown in Table 1-2 (from 1010 psig to 1040 psig, and from 1040 psig to 1070 psig). Therefore, the increase for the PIV surveillance test corresponds with the increase in normal reactor operating pressure that accompanies power uprate. The change is needed to provide a more appropriate test range for the higher uprate reactor operating pressure.

14. The Reactor Coolant System pressure and temperature limits are changed to account for the increased neutron flux resulting from power uprate conditions, as required, in Figure 3.4.11-1 of the RBS TS.

Evaluation

As discussed in Section 3.3.1 of Enclosure 7, the reactor pressure vessel (RPV) embrittlement caused by neutron exposure of the vessel wall is predicted to increase the integrated fluence over the period of plant life. Operation with power uprate results in a higher neutron flux, which would affect vessel toughness.

The maximum operating dome pressure for power uprate is changed from that for original power operation. Therefore, a change in the RPV hydrostatic and leakage test pressures is required as mentioned above. Since the vessel remains in compliance with the regulatory requirements, operation with power uprate will not have an adverse effect on the reactor vessel fracture toughness. This change would replace the "Minimum Reactor Pressure Vessel Metal Temperature vs. Reactor Vessel Pressure" curves, as discussed in Section 3.3.1.1, with those shown in Figures 3-2a and 3-2b. These changes will maintain operation consistent with the assumptions of the safety analysis for power uprate.

15. A change is proposed to Specification 3.4.12 and Bases of the RBS TS to increase the reactor steam dome operating pressure from 1045 psig to 1075 psig.

Evaluation

As discussed in Section 3.2 of Enclosure 7 the design of the reactor coolant pressure boundary remains at 1250 psig and the ASME Code allowable peak remains at 1375 psig. The power uprate analysis remains within the ASME limits with an initial pressure of 1078 psig and maintains operation consistent with the assumptions of the safety analysis for power uprate. The increase in the reactor steam dome operating pressure Limiting Condition for Operation (LCO) by the same amount as the nominal operating dome pressure increase is shown in Table 1-2 (from 1045 psig to 1075 psig). This value is the basis for the initial value used in the reactor overpressure protection analysis described in Section 3.2.

Bases Changes

Change to the TS Bases are proposed for consistency with the power uprate safety analyses concerning turbine bypass capacity. This change is in addition to the Bases changes corresponding to proposed changes above. The increase in reactor power will not result in a reduction in the available turbine bypass capacity. Rather, the change will reduce the capacity when expressed as a percentage of total reactor power. This change is discussed in Section 7.3 of Enclosure 7. As a result, the bypass capacity described in Bases Section 3.7.5 will be revised to identify the capacity of the system as from 10% to 9.5 % of the Nuclear Steam Supply System rated flow. This change is included with Enclosure 2

Other Issues

In addition to the RBS amendment request, the development of this request identified the need to address an issue not included above or in Enclosure 7. The issue is the treatment of non-limiting transients that are not required for the design and justification of the uprated power level. The NRC SER on the generic GE guidelines, Reference 1, Section 5.3.2, states that only the limiting transients in Appendix E of the GE guidance need to be revised for power uprates. The list of transients evaluated does not include all transients currently in Chapter 15 of the RBS USAR. The transients not analyzed as part of the uprate evaluations are not limiting transients.

EOI proposes that this USAR information be relocated to a historical appendix of the USAR. Such a relocation will be consistent with guidance contained in NEI 98-03, Revision 1, "Guidelines for Updating Final Safety Analysis Reports."

References

1. NEDC-31897P-A, "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," May 1992.
2. NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," July 1991; and Supplements.