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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 9, 2000

Mr. Randall K. Edington
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: INCREASE IN SAFETY RELIEF VALVE SETPOINT TOLERANCE (TAC NO. MA6185)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 109 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) as an initial response to your application to increase current licensed power level from 2894 megawatts thermal (MW_{th}) to 3039 MW_{th} (power uprate), dated July 30, 1999, as supplemented by your letter dated December 2, 1999.

The amendment increases the safety function lift setpoint tolerances for the main steam safety and relief valves (S/RVs) listed in Surveillance Requirement (SR) 3.4.4.1 of the TSs. The tolerances for these valves are increased from +0/-2 percent of the safety function (i.e., safety valve actuation) lift setpoint to +0/-3 percent of that setpoint. The Bases page in the TSs for the Limiting Condition for Operation 3.4.4, associated with SR 3.4.4.1, was also changed to show the setpoint tolerances for testing the S/RVs are now +0/-3 percent.

Although the S/RV SR test criteria will reflect the new tolerance of +0/-3 percent, the S/RV "as-left" safety lift point will still be set within a tolerance of +0/-2 percent. If a valve is tested and the lift setpoint is found outside the +0/-3 percent tolerance, two additional valves will be tested for each S/RV that fails its SR, and the setpoint for all valves will be set/reset to within the +0/-2 percent tolerance. Therefore, the +0/-3 percent tolerance is only the criteria for testing the valves and deciding if a valve has failed the test.

Approval of these proposed changes are based on the staff's Safety Evaluation (SE) for the General Electric topical report, NEDC-31753P, "BWROG [Boiling Water Reactor Owners Group] In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990. The SE for NEDC-31753P required certain analyses to be performed and the results of these analyses for River Bend are addressed in the enclosed SE. These analyses would be part of future reload analyses which includes the tolerances for S/RV lift setpoints in SR 3.4.4.1.

DF01

A copy of our related SE is enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Robert J. Fretz, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 109 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY GULF STATES, INC.**

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 109
License No. NPF-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Gulf States, Inc.* (the licensee) dated July 30, 1999, as supplemented by letter dated December 2, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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* Entergy Operations, Inc., is authorized to act as agent for Entergy Gulf States, Inc, and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

**Entergy Gulf States, Inc., has merged with a wholly owned subsidiary of Entergy Corporation. Entergy Gulf States, Inc. was the surviving company in the merger.

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment; and Paragraph 2.C.(2) of Facility Operating License No. NPF-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 109 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 9, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 109

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by Amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3.4-10	3.4-10
B 3.4-19	B 3.4-19

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of five S/RVs shall be OPERABLE.

AND

The relief function of four additional S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required S/RVs are as follows:		In accordance with the Inservice Testing Program
	Number of S/RVs	Setpoint (psig)	
	7	$\geq$ 1130.1 and $\leq$ 1165	
	5	$\geq$ 1144.6 and $\leq$ 1180	
	4	$\geq$ 1154.3 and $\leq$ 1190	

LCO (continued)	-3% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism. Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.
APPLICABILITY	In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the heat. In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.
ACTIONS	A.1 and A.2 With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required S/RVs are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.4.4.1 This Surveillance demonstrates that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the S/RV safety function

(continued)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 109 TO FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated July 30, 1999, Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) (Appendix A to Facility Operating License No. NPF-47) for the River Bend Station, Unit 1 (RBS). The proposed change, more commonly referred to as "power uprate," would revise the TSs and the operating license to extend operation of the station from its licensed power of 2894 megawatts thermal (MW_{th}) to the uprated power level of 3039 MW_{th}, an increase of 5 percent. Included in the power uprate license amendment application was a request to increase the main steam safety and relief valves (S/RV) safety mode/function setpoint tolerance defined in Surveillance Requirement (SR) 3.4.4.1 from +0/-2 percent to ±3 percent.

The purpose of this license amendment is to independently approve, prior to the issuance of the power uprate license amendment, a portion of the S/RV setpoint tolerance change requested. The proposed change would increase the safety function lift setpoint tolerances for the S/RVs listed in SR 3.4.4.1 from the current +0/-2 percent of the safety function lift setpoint to +0/-3 percent (i.e., a partial 3 percent tolerance). The remaining ("+3 percent") portion of the proposed setpoint tolerance change will be reviewed in conjunction with approval for the power uprate. In a letter dated December 2, 1999, the licensee stated that implementation of the entire ±3 percent tolerance would require physical changes, which are currently scheduled for refueling outage (RF) 10, to specific motor operated valve (MOV) settings to account for the possible increase in main steam line system pressure. This schedule was based upon fully implementing the power uprate amendment at this time, whereby the ultimate nominal lift setpoint tolerance will be ± 3 percent.

In addition, the licensee committed in its July 30, 1999, application to incorporate the recommendations of General Electric Company (GE) proprietary topical report NEDC-31753P, "BWROG [Boiling Water Reactors Owners Group] In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990, and its associated Safety Evaluation Report (SER), to reset all tested valves to within a 2 percent tolerance band (typically ±1 percent). Accordingly, this commitment will be implemented by adjusting the "as-left" safety lift point for all tested S/RVs to within a tolerance of +0/-2 percent. The licensee also committed to another recommendation of NEDC-31753P to increase the test sample size by two valves for each valve found outside its set point tolerance. These recommendations will be incorporated into the RBS Inservice Testing (IST) program.

The Bases page in the TSs for the Limiting Condition for Operation (LCO) associated with SR 3.4.4.1 (i.e., LCO 3.4.4) would also be changed to reflect the +0/-3 percent setpoint tolerances. LCO 3.4.4 defines how many S/RVs are required to be operable for plant operation.

The proposed change does not alter the actual safety lift setpoints for the S/RVs, the frequency of verifying these setpoints, the number of these valves required to be operable in LCO 3.4.4, or other surveillance requirements in the TSs on these valves.

2.0 BACKGROUND

The S/RVs provide overpressure protection for the reactor coolant system (i.e., reactor vessel, main steam lines, and associated piping) and are discussed in Section 5.2.2 of the Updated Safety Analysis Report (USAR) for the RBS. In its letter of July 30, 1999, the licensee explained that the S/RVs at RBS are Crosby, direct-acting spring-loaded safety valves with attached pneumatic cylinder for relief mode operation. Each S/RV performs its intended function through two modes of operation:

- Safety mode by direct action of the steam pressure against a single spring-loaded disk that will open when the valve inlet pressure force exceeds the spring force. The safety function set pressure is determined by changing the value of the compressed spring force.
- Relief mode by using an auxiliary actuating device consisting of a pneumatic piston/cylinder and a mechanical linkage assembly which opens the valve by overcoming the spring force.

As explained in the USAR, credit is taken for the dual purpose S/RVs in the valves' American Society of Mechanical Engineers (ASME) Code-qualified modes of safety operation. This is to say, when system pressure increases to the relief pressure setpoint of a group of S/RVs having the same relief setpoint, half of these valves are assumed to operate in the relief mode. They are opened by pneumatic power actuation. If the system pressure increases to the spring setpoint of a group of valves, those valves not already open are assumed to begin opening and to reach full-open at 103 percent of the setpoint.

The range of the maximum pressure limit for the power-actuated relief mode is 1125 to 1155 psig and the current spring-action safety mode is 1175 to 1215 psig.

The plant has a total of 16 S/RVs installed on the 4 main steam lines (MSLs). The inservice testing (IST) for the S/RVs is in accordance with ASME/American National Standards Institute (ANSI) OM-1987, Part 1, or the ASME/ANSI OM as approved by the RBS IST program. The setpoints for the S/RVs are less than the reactor coolant design pressure of 1250 psig. The valves are removed from the MSLs for testing during a RF. They are tested on a test fixture for both the safety and relief modes of operation and seat leakage. The safety mode of operation is tested to determine the as-found lift setpoint of the valve and the tolerance with respect to the required setpoints stated in SR 3.4.4.1 of the TSs.

Although the setpoint tolerances would be increased in the TSs, S/RV setpoints would still be set to within the current +0/-2 percent tolerance, and the valves tested and found to have setpoints outside the +0/-2 percent tolerance would have their setpoints reset to within the +0/-2 percent tolerance or replaced by valves with setpoints set to that tolerance. The purpose of the proposed change is to change the criteria in the RBS TSs by which a valve which is tested is considered a failure. This change has the potential to significantly reduce the plant personnel radiation exposure resulting from unnecessary S/RV refurbishment and testing, and the number of valves being tested.

In its letter to the BWROG dated March 8, 1993, the staff stated that it had reviewed GE proprietary topical report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Topical Report," dated February 1990. In the Safety Evaluation (SE) to that letter, the staff concluded that the topical report was acceptable as the basis for (1) increasing the S/RV setpoint tolerance to ±3 percent, and (2) the frequency of testing the valves is one-half the number of valves at least once per 18 months and all within 40 months, with two additional valves tested for each valve found outside the acceptable tolerance, subject to the following conditions:

- Perform a transient analysis of all abnormal operational occurrences (AOOs) as described in NEDC-31753P using the ±3 percent setpoint tolerance for the safety mode of S/RVs. The standard reload methodology (or other method approved by the staff) should be used for this analysis.
- Perform an analysis of the design basis overpressurization event using the ±3 percent tolerance limit to confirm that the vessel pressure does not exceed the ASME Boiler Pressure Vessel Code upset limit.
- The plant specific analyses described above should assure that the number of S/RVs included in the analyses correspond to the number required to be operable in the TSs.
- Evaluate the performance of high pressure systems (e.g., pump capacity, discharge pressure), MOVs, and vessel instrumentation and associated piping using the ±3 percent tolerance limit.
- Evaluate the ±3 percent tolerance limit on any plant-specific alternative operating modes (e.g., increased core flow, extended operating domain).
- Evaluate the effect on the ±3 percent tolerance limit on the containment response during loss-of-coolant accidents (LOCA) and the hydrodynamic loads on the S/RV discharge lines and containment.

Therefore, since RBS is a BWR/6, topical report NEDC-31753P, and the staff's evaluation of that topical report, applies to RBS.

These conditions must be reassessed in each reload analysis for a refueling outage and the operating cycle that follows the refueling outage. For this proposed amendment to the RBS license, the partial 3 percent tolerance will be the failure criteria for the S/RVs tested during the upcoming RF 9 and the reload analysis will apply to the Operating Cycle 10.

3.0 EVALUATION

In an attachment to its letter, the licensee provided its justification for a proposed setpoint tolerance change of ±3 percent with respect to the power uprate license amendment request. In the attachment, the licensee addressed the conditions listed by the staff in its evaluation of NEDC-31753P identified above. In addition, the licensee stated in its December 2, 1999, letter that it has evaluated the ±3 percent S/RV setpoint tolerance at current operating levels, and the study results show that plant and system performance "remains within the RBS design and licensing basis."

3.1 Perform a Transient Analysis Using Staff Approved Methodology

The licensee explained that Section 4.3 of NEDC-31753P stated that group 3 plants (BWR/5 and BWR/6 design) do not require an evaluation of AOOs because only changes are proposed to the safety mode of actuation of the S/RVs, and RBS is a BWR/6 design with S/RVs that have safety and relief modes of actuation.

The licensee further explained that AOOs at RBS which result in S/RV actuation take credit for 5 of these valves in the safety mode of operation, the minimum number required by the RBS TSs. However, since this amendment applies to a more conservative testing tolerance of +0/-3 percent with no change to the upper surveillance test limit and the tolerance used to set the S/RV's lift point following testing, there would be no change to analysis showing that reactor steam dome pressure does not exceed the ASME Code pressure vessel limit for RBS. The maximum reactor steam dome pressure allowed by RBS TSs is an initial condition for the design basis LOCA, and does not change as a result of this license amendment.

Analysis would be conducted by the licensee for the future operating cycles as part of each reload analysis and be completed before the operating cycle following the reload in the refueling outage. Therefore, since the current transient analysis is unaffected by this change and future cycles will require a new analysis associated each reload, the staff finds this acceptable.

3.2 Analysis of the Design Basis Overpressurization Event

The licensee stated in its application that the RBS design basis (i.e., worst case) overpressurization event is a closure of all main steam isolation valves (MSIVs) when the reactor is operating at 102 percent of rated power and 107 percent of rated core flow (Increased Core Flow). It is assumed that a reactor scram on MSIV position fails and, therefore, the scram occurs on high neutron flux. The BWR/6 design (i.e., RBS, Unit 1) meets the allowance in ASME Code, Section III, Article NB 7542, that up to half of the installed S/RVs may take credit for the auxiliary actuating device (i.e., the relief mode of actuation); however, the RBS analysis only credits 4 of the 16 S/RVs for actuation in the relief mode, as stated in LCO 3.4.4.

Since this amendment applies to a more conservative testing tolerance of +0/-3 percent, with no change to the upper surveillance test limit and tolerance used to set the S/RV's lift point following testing, there would be no change to design basis overpressurization analysis. Moreover, the licensee is required to perform this analysis using NRC approved methodology described in General Electric Standard Application for Reactor Fuel (GESTAR-II), NEDE-24011-P-A as a part of the normal reload analysis process before each operating cycle, including the upcoming Operating Cycle 10. This analysis will be bound by the tolerance for S/RV safety lift setpoints in SR 3.4.4.1. Therefore, the staff finds this acceptable.

3.3 Number of Valves Taken Credit for are Required to be Operable

The number of S/RVs assumed in the analyses described above is consistent with LCO 3.4.4, which is associated with SR 3.4.4.1 in the TSs. The above analyses take credit for 5 safety mode S/RVs and 4 relief mode S/RVs, and both of these numbers are required to be operable in LCO 3.4.4. The numbers in LCO 3.4.4.1 are not being changed by this proposed change on the setpoint tolerances.

3.4 Evaluate Performance of High Pressure Systems, MOVs, and Vessel Instrumentation and Piping

The licensee evaluated the effect of the proposed ±3 percent tolerances on the plant high pressure systems, MOVs, and the vessel instrumentation and piping in conjunction with its analysis for power uprated conditions. Since there is no change in upper tolerance limit (currently +0 percent) or change in the TS for maximum allowable steam dome pressure being considered at this time, the performance of high pressure systems will not be affected. Therefore, the staff concludes that the +0/-3 percent tolerance change is acceptable for the following plant systems:

High Pressure Systems

RBS has three high pressure, reactor vessel injection systems: high pressure core spray (HPCS), reactor core isolation cooling (RCIC), and standby liquid control (SLC). The licensee addressed each of these systems.

The HPCS system is designed to pump water into the reactor vessel over a wide variety of pressures in the event of a small break LOCA which does not immediately depressurize the reactor vessel. The HPCS surveillance test requirement is based on the HPCS flow versus reactor pressure under rated injection conditions (i.e., 200 psid reactor pressure), which is not affected by changes to S/RV setpoints or setpoint tolerances.

The RCIC system provides core cooling in the event of a transient where the reactor vessel is isolated from the main condenser, concurrent with the loss of all feedwater flow, and where reactor pressure is too high to allow initiation of low pressure systems. Since there is no change in upper tolerance limit or change in the TS for maximum allowable steam dome pressure, there will be no change in RCIC system performance.

The SLC system operation is not affected by the increase in S/RV safety setpoint tolerance. The pressure used for SLC performance is based on the S/RV relief settings and the safety settings are the only settings (i.e., the tolerance of the safety lift setpoint) being changed by this proposed action. Therefore, the licensee concluded that this proposed action does not affect the SLC.

MOVs

MOV dynamic testing is performed with the valve in place and at the highest differential pressure achievable under normal operations. Therefore, the test parameters are not affected by S/RV safety setpoint tolerances which are for accident conditions. However, MOV operator settings are based on the calculated maximum expected differential pressure (MEDP) for the valve which would include accidents. Since there is no change in the TS for maximum allowable steam dome pressure, the current MEDP calculations and MOV settings are sufficient for the proposed change.

Vessel Instrumentation and Piping

The reactor vessel instrument piping is designed to at least the rated vessel design pressure of 1250 psig. This piping is also protected by the S/RVs which satisfy the ASME Code requirements for overpressure protection for the vessel. However, since there is no change to the upper S/RV setpoint tolerance limit or the TS for maximum allowable steam dome pressure, reactor vessel instrument and its associated piping is not impacted.

3.5 Plant-Specific Alternate Operating Modes

Plant-specific transient and accident analysis performed in support of the current Cycle 9 operation incorporates the following alternate operating modes: (1) Maximum Extended Load Line Limit Analysis, (2) Single Loop Operation, (3) Increased Core Flow, and (4) Feedwater Temperature Reduction. The licensee stated that analysis for the ±3 percent tolerance maintained adequate margins for these operating modes. Since (1) the proposed surveillance test tolerance change evaluated under this license amendment is within this tolerance, (2) there is no change in the allowed reactor steam dome pressure, and (3) actual S/RV safety mode as-left lift setpoints do not change, the analysis incorporating the above-listed operating modes is bounded and acceptable. Plant-specific transient and accident analysis will be required to be performed by the licensee for the future operating cycle as part of each reload analysis and be completed before the operating cycle following the reload in the refueling outage.

3.6 Evaluate Containment Response During LOCA and Hydrodynamic Loads

The licensee evaluated the effect of the entire ±3 percent tolerance on (1) the containment response during LOCAs, and (2) the hydrodynamic loads on the S/RV discharge lines and containment under power uprate conditions. The results of this analysis showed that adequate design margin was maintained under these conditions. Because the relief mode of actuation of the S/RVs is taken credit for during the LOCA and the relief settings are not being changed by this proposed change, the partial (and more conservative) 3 percent tolerance will not have an effect on the conditions of the blowdown from the primary coolant system into containment and the containment response for the LOCA. In addition, since there will be no change to the upper S/RV setpoint tolerance limit or the TS for maximum allowable steam dome pressure associated with this license amendment, there will also be no change to discharge line loads. Consequently, the partial tolerance change will not result in any changes to allowable stress calculations associated with the discharge line piping and supports between the S/RVs and the first anchor point. Therefore, the staff finds this acceptable.

3.7 Frequency of Testing the Valves

The staff stated in its SE for NEDC-31753P that the frequency of testing the valves should be one-half the number of valves at least once per 18 months (a refueling outage) and all within 40 months, with two additional valves tested for each valve found outside the acceptable tolerance. The licensee proposed that the current IST program continue to control the frequency of testing of valves including the S/RVs. IST for the S/RVs is presently in accordance with ASME/ANSI OM-1987, Part 1 (OM-1), or as stated by the licensee, in future years, the ASME/ANSI OM as approved by the RBS IST program. Using OM-1 to determine the test sample number is consistent with Section 4.4 of the Technical Evaluation Report attached to the staff's SE for NEDC-31753P in that it is stated in Section 4.4 that "the adoption of the industry practices proffered by ANSI/ASME-OM-1981 ["Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Valves"] would provide assurances that an adequate number of operable [S/RVs] exists to prevent the reactor pressure from exceeding design pressure." ASME/ANSI OM-1987, Part 1, Paragraph 1.3.3(b), requires that all valves of each type and manufacture shall be tested within each subsequent 5-year period, with a minimum of 20 percent of the valves tested within any 24 months. The 20 percent sample population shall be previously untested valves, if possible. OM-1 would, therefore, require testing of a minimum of 4 S/RVs.

As stated in USAR Section 5.2.2.10, the licensee had committed to inspecting and testing all S/RVs a maximum of every 3 years to ensure internal component service life is not exceeded. This section also states that all SRVs are refurbished at a maximum of 5 years. However, because of the potential for a large number of failures at the +0/-2 percent tolerance, the licensee has historically removed and tested all of the valves during each refueling outage to prevent removal and testing of additional valves from affecting the outage schedule. Test failures using the current +0/-2 percent tolerance for the past 3 refueling outages was approximately 52 percent, or an average of 8 failures per outage.

If the proposed partial 3 percent tolerance had been applied using the same data reported from the previous 3 refueling outages, the number of valve failures would have been 12 out of the 48 valves tested, or a probability of failure of 0.25 per valve tested. This smaller failure rate should lead to fewer valves tested in the future and potentially less occupational exposure associated with removing and testing the valves. Nevertheless, the staff concludes that the licensee's adoption of the ASME/ANSI OM-1987, Part 1 test frequency requirements is acceptable.

Evaluation Conclusions

The staff has reviewed the 6 conditions listed in the staff's SE on NEDC-31753P and provided in Section 2.0 above with respect to changing the setpoint tolerance from +0/-2 percent to +0/-3 percent. The staff concludes that the partial implementation of the proposed ±3 percent tolerance change adequately addresses the 6 conditions.

Therefore, the staff concludes that the proposed amendment to increase the safety lift setpoint tolerances to +0/-3 percent is acceptable. Changes to the Bases page in the TSs for the LCO associated with SR 3.4.4.1 (i.e., LCO 3.4.4) to show the safety lift setpoint tolerances of +0/-3 percent is also acceptable.

Approval of these proposed changes is based on the staff's SE for the GE proprietary topical report, NEDC-31753P. The SE for NEDC-31753P required certain analyses to be performed and the results of these analyses for RBS are addressed above. As described in the staff's SE for NEDC-31753P and this SE, future reload analyses must include the tolerances for S/RV lift setpoints in SR 3.4.4.1 and demonstrate that operation with these tolerances is acceptable, prior to operation in that operating cycle.

The format for the TS pages issued is different from the proposed format submitted by the licensee. The licensee proposed using the format of "nominal setpoint ± 36 (psig);" however, since the nominal setpoints are not increasing and the tolerance will only be partially changed, the format for the TS pages will be similar to the existing format which uses a range of values to specify the allowed SR values.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Louisiana State Official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 62712 dated November 17, 1999). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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