50-416

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

June 12, 1996

Mr. C. Randy Hutchinson
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: ISSUANCE OF AMENDMENT NO.123 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M94857)

Dear Mr. Hutchinson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 123 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment revises the Technical Specifications (TSs) in response to your application dated February 22, 1996.

The amendment increases the safety function lift setpoint tolerances for the safety and relief valves (S/RVs) that are listed in Surveillance Requirement (SR) 3.4.4.1 of the TSs. The tolerances for these valves are increased from ± 1 percent of the safety function (i.e., safety valve actuation) lift setpoint to ± 3 percent of the 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 ± 3 percent.

The S/RV safety lift setpoints will still be set within a tolerance of ± 1 percent, but the setpoints will be tested to within the ± 3 percent. If a valve is tested and the lift setpoint is found outside the 3 percent tolerance, two additional valves are to be tested and the setpoint for the failed valve will be reset to within the ± 1 percent tolerance. Therefore, the 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 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 Grand Gulf are addressed in the enclosed SE. As described in the staff's SE for NEDC-31753P and this SE, these analyses would be part of future reload analyses which 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 future operating cycles.

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We recognize that your staff has not completed all of the reload analyses for the Operating Cycle 9 which will follow the upcoming refueling outage; however, these analyses will be performed for Operating Cycle 9 before operation in that operating cycle, as they would be performed for future operating cycles as part of the reload analyses for that operating cycle.

A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Jack N. Donohew

Jack N. Donohew, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: 1. Amendment No. 123 to NPF-29
2. Safety Evaluation

cc w/encls: See next page

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2. Safety Evaluation

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 123
License No. NPF-29

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated February 22, 1996, 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, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   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.

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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-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 123, are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Jack N. Donohew, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 12, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 123

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

REMOVE	INSERT
3.4-10	3.4-10

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of seven S/RVs shall be OPERABLE,

AND

The relief function of six 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.	12 hours
	AND	
	A.2 Be in MODE 4.	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: Number of S/RVs — Setpoint (psig) 8 — 1165 ± 34.9 6 — 1180 ± 35.4 6 — 1190 ± 35.7	In accordance with the Inservice Testing Program

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 123 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated February 22, 1996, the licensee (Entergy Operations, Inc.) submitted a request to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TSs). The proposed change would increase the safety function lift setpoint tolerances for the safety and relief valves (S/RVs) that are listed in Surveillance Requirement (SR) 3.4.4.1 of the TSs. The tolerances would be increased from the current ± 1 percent of the safety function lift setpoint to ± 3 percent (i.e., the 3 percent tolerance).

The safety lift setpoints will still be set within a tolerance of ± 1 percent, but the setpoints will be tested to within ± 3 percent to determine acceptance or failure of the as-found valve lift setpoint. Valves with setpoints found outside the 3 percent tolerance during testing would lead to additional valves being tested and the setpoints being reset to within the ± 1 percent tolerance. Therefore, the 3 percent tolerance would determine whether additional valves would be tested.

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 show the setpoint tolerances would be ± 3 percent. LCO 3.4.4 defines how many S/RVs are required to be operable for plant operation.

The licensee provided a plant-specific report, GGNS-96-0005, "Grand Gulf Nuclear Station, Engineering Report, Safety/Relief Valves Safety Function Lift Setpoint Tolerance Relaxation, Summary Report," dated February 1996, which was attached to its letter of February 22, 1996.

The proposed change does not alter the 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.

Phone conference calls were held with the licensee on April 10 and 11, 1996, and discussions were held at the site during the NRC project manager's visit to the site on April 18, 1996. These calls and discussions were held to

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clarify statements made in Attachment 2 to the February 22, 1996, submittal about the justification for the proposed change. The clarifications involved the following:

- motor-operated valve capability;
- which analysis was performed for the maximum extended operating domain with the 3 percent tolerance;
- the specific pump whose seizure was the most limiting for single loop operation;
- the effect of the relief function on the initial conditions of the loss of coolant accident; and
- the inservice testing requirements on testing S/RVs and the history of testing these valves at Grand Gulf.

The information provided by the licensee was within the scope of the notice of proposed license amendment.

2.0 BACKGROUND

Although the setpoint tolerances would be increased in the TSs, the licensee stated that the S/RV setpoints would still be set to within the current 1 percent tolerance, and the valves tested and found to have setpoints outside the 1 percent tolerance would have their setpoints reset to within the 1 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 Grand Gulf TSs by which a valve which is tested is considered a failure. This change would 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 BWR (Boiling Water Reactor) Owners' Group (BWROG) on March 8, 1993, the staff stated that it had reviewed the General Electric Company (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 American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME 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), motor-operated valves, 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 and the hydrodynamic loads on the S/RV discharge lines and containment.

GGNS is a BWR and the topical report NEDC-31753P, and the staff's evaluation of that topical report, applies to GGNS.

These conditions must be reassessed in each reload analysis for a refueling outage and the operating cycle that follows the refueling outage. This reload analysis is for the fuel which will be added to the core during the refueling outage. For this proposed amendment to the Grand Gulf license, the 3 percent tolerance will be the failure criteria for the S/RVs tested during the upcoming Refueling Outage 8 and the reload analysis will apply to the Operating Cycle 9.

The S/RVs are the 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 Final Safety Analysis Report (UFSAR) for the Grand Gulf Nuclear Station, Unit 1. In the letter of February 22, 1996, the licensee explained that the S/RVs at Grand Gulf are Dikkers, 8 X 10, direct-acting spring-loaded safety valves with attached pneumatic cylinder for relief mode operation. The licensee stated that 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 UFSAR, credit is taken for the dual purpose S/RVs in the valves' 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 proposed change will alter the latter pressure range to 1200 to 1225 psig.

The plant has total of 20 S/RVs installed on the four main steam lines (MSLs). The inservice testing (IST) program for the S/RVs is in accordance with the IST requirements in Section XI, 1980 Edition, of the ASME Code, with Addenda through Winter 1980 (Subsections IWA and IWV). 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 refueling outage. 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.

3.0 EVALUATION

In an attachment to its letter, the licensee provided its justification for the proposed change to the TSs, and addressed the conditions listed by the staff in its evaluation of NEDC-31753P and identified above.

Perform A Transient Analysis Using Staff Approved Methodology

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

However, the licensee further explained that AOOs at GGNS that result in S/RV actuation take credit for operation of seven of these valves, the minimum number required by the GGNS TSs. Therefore, it was necessary for the licensee to determine if the proposed $\pm$ 3 percent tolerance limit would affect any of the previously analyzed AOOs.

The licensee stated that each of the previously analyzed operating occurrences were analyzed using NRC approved methodology, as described in the staff's SE for NEDC-31753P, and the $\pm$ 3 percent tolerance limit, and none of the analyses resulted in reactor steam dome pressures exceeding the ASME Code pressure vessel limit for GGNS. The maximum allowed reactor steam dome pressure in the Grand Gulf TSs is an initial condition for the design basis Loss of Coolant Accident (LOCA).

This 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.

Analysis of the Design Basis Overpressurization Event

The licensee stated in its application that the GGNS design basis (i.e., worst case) overpressurization event is a closure of all main steam isolation valves (MSIVs) when the reactor is operating at 100 percent of rated power and 105 percent of rated 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., Grand Gulf, 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 Grand Gulf analysis only credits six of the 20 S/RVs for actuation in the relief mode, as stated in LCO 3.4.4.

Although the design basis overpressurization analysis has not been performed for the upcoming Operating Cycle 9 (Refueling Outage 8 for this operating cycle should begin in October 1996), it will be performed before plant restart from the refueling outage for plant operation in the operating cycle. The licensee stated that this analysis is performed as part of the normal reload analysis process before each operating cycle and, for upcoming Operating Cycle 9, the analysis will be performed by GE using the NRC approved ODYN methodology described in GESTAR-II, NEDE-24011-P-A.

Further, the licensee stated that overpressurization analyses performed for previous operating cycles using S/RV opening pressures in excess of the proposed 3 percent tolerance showed considerable margin (i.e., approximately 100 psi) to the ASME Code pressure vessel limit of 1375 psig for Grand Gulf (i.e., 110 percent of the reactor vessel design pressure of 1250 psig). Based on this and what the licensee stated was the relative insensitivity of these results to the fuel design parameters, the licensee concluded that future analyses should yield peak pressures with similar margin to the ASME Code limit.

Although the licensee has not completed its analysis of the design basis overpressurization event for the next operating cycle using the 3 percent tolerance to confirm that the vessel pressure does not exceed the ASME upset limit, this will be performed and verified before operation in the next operating cycle. As described in the staff's SE for NEDC-31753P and this SE, future reload analyses including this analysis will be bound by the tolerance for S/RV safety lift setpoints in SR 3.4.4.1.

Number of Valves Taken Credit For Are Required To Be Operable

The licensee stated that 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 seven safety mode S/RVs and six

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.

Evaluate Performance of High Pressure Systems, Motor Operated Valves (MOVs), and Vessel Instrumentation and Piping

The licensee evaluated the effect of the proposed 3 percent tolerances on the plant high pressure systems, the MOVs, and the vessel instrumentation and piping:

High Pressure Systems:

Grand Gulf plant 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.

For HPCS, the design discharge pressure of the pumps and the discharge pressure to the reactor vessel is 1575 psig. This design pressure is well above the pressures that could result from the S/RV pressure setpoints and the proposed 3 percent tolerance (i.e., 1226 psig). It was designed to deliver a minimum flow of 550 gpm to the reactor vessel at a vessel pressure of 1177 psi above pump suction pressure. The licensee stated that a review of HPCS pre-operational test data indicated that the system would deliver at least 550 gpm at the higher vessel pressure allowed by the proposed 3 percent tolerance. The licensee concluded that adequate margin would be maintained for the HPCS with the new 3 percent tolerance.

For RCIC, although the system is not taken credit for in the Grand Gulf vessel pressurization safety analyses and the system operating parameters would not affect the analyses, the licensee reviewed the effects of the proposed change on the system. The pump discharge pressure is 1525 psig, the discharge piping has a design pressure of 1500 psig, and the inlet piping from the reactor vessel to RCIC is rated for the same pressure as the vessel. These pressures are above the pressures that could result from the S/RV setpoints and the proposed 3 percent tolerance. For RCIC injection at the increased vessel pressure, turbine steam flow is expected to increase 2.7 percent in order to maintain the 800 gpm design flow rate. Based on vendor turbine performance curves, the licensee states that RCIC can accomodate the increased steam flow and function at higher pressures. Therefore, the licensee concluded that adequate margin is available for RCIC.

For SLC, the licensee stated that 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.

Motor-Operated Valves (MOVs):

MOV dynamic testing is done with the valve in place and is performed at the highest differential pressure achievable under normal operations and, 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. The adequacy of MOV settings was assessed by the licensee by evaluating the adequacy of the MEDP calculation assumptions and the resulting MEDP values established for those MOVs potentially affected by the 3 percent tolerance. The licensee stated that the previous MEDP calculations using reactor pressure based solely on the S/RV safety lift setpoint (i.e., neglecting the relief mode) and the current 1 percent tolerance bounds the calculations using the S/RV relief mode of operation and the 3 percent tolerance. Therefore, the current MEDP calculations and MOV settings are sufficient for the proposed change.

Vessel Instrumentation:

Instruments may be affected by the increased vessel pressure resulting from the proposed higher setpoint tolerance. The instruments in high pressure systems such as the control rod drive and SLC systems are designed to operate at higher pressures than could result from the proposed tolerance change. The licensee stated in the engineering report attached to its letter of February 22, 1996, that a review of vendor information for each instrumentation indicated the increased vessel pressure was within the design pressure of the instrumentation. The calibration information and calibration range was also reviewed and found adequate for the increased vessel pressure. The licensee concluded that the proposed change had no impact on vessel instrumentation.

Vessel Instrumentation Piping:

In Section 2.6.2 of the plant-specific report attached to its letter, the licensee explained that the piping connected to the reactor vessel is within the reactor coolant pressure boundary (RCPB). The piping is designed to at least the rated vessel design pressure of 1250 psig for the vessel. This piping is also protected by the S/RVs which satisfy the ASME Code requirements for overpressure protection for the vessel.

The licensee stated that the pressure transients associated with upset and faulted conditions are analyzed in the UFSAR and are bounded by core reload analyses which use a 6 percent tolerance for S/RV safety mode operation in evaluating maximum overpressure events. This tolerance is greater than the proposed 3 percent tolerance and the licensee concluded that the instrument piping has adequate margin for overpressure protection.

Evaluate Plant-Specific Alternative Operating Modes

The licensee evaluated the effect of the proposed 3 percent tolerances on the following plant-specific alternative operating modes: maximum extended operating domain, single loop operation, and feedwater heaters out of service.

Maximum Extended Operating Domain (MEOD):

The transient and accident analyses performed in support of the current Cycle 8 reload (i.e., the upcoming Refueling Outage 8) incorporated the extended power and increased core flow available under the MEOD for Grand Gulf. These analyses have been performed at rated conditions of 104.2 percent of rated power and 108 percent of rated flow, which bound the 100 percent power and 105 percent flow combination for the MEOD. As stated in UFSAR Section 5.2.2.2.2.1, these rated conditions are the most severe for Grand Gulf because the maximum stored energy exists at these conditions. These analyses were performed using safety function setpoint tolerances that exceeded the proposed 3 percent.

These analyses apply to Operating Cycle 9 because they have been done for the Cycle 8 refueling outage. They would 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.

Single Loop Operation (SLO):

The licensee stated that recirculation pump seizure is the limiting event at Grand Gulf for SLO and, based on the analysis of this transient, no plant parameters will be exceeded during SLO as a result of the proposed 3 percent tolerance.

Feedwater Heaters Out of Service (FWHOS):

The licensee stated that, in the fuel reload safety analyses for Refueling Outage 8, the feedwater controller failure (FWCF) with FWHOS is bounded by the FWCF event without bypass. This statement is consistent with statements in UFSAR Section 15.1.2.3.3 that the cases of FWCF with bypass and with feedwater heaters out of service were analyzed in previous cycles and shown to be bounded by the FWCF event without bypass case. Therefore, the FWHOS operational mode is bounded by the transient analysis using staff approved methodology discussed above.

The Cycle 8 reload safety analyses will be performed before the upcoming Operating Cycle 9. These analyses would be conducted 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.

Evaluate Containment Response During LOCA and Hydrodynamic Loads

The licensee evaluated the effect of the proposed 3 percent tolerances on (1) the containment response during LOCAs and (2) the hydrodynamic loads on the S/RV discharge lines and containment.

Containment Response During LOCAs:

The licensee stated that the most limiting event in terms of peak containment pressure and temperature, and peak suppression pool temperature for Grand Gulf is the design basis LOCA, a double ended guillotine rupture of the main steam line. The licensee stated that increasing the safety lift setpoint tolerances for the S/RVs has no effect on this event because the vessel depressurizes without any S/RV actuation. 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 3 percent tolerances 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.

Consistent with the above, the licensee has also stated that none of analyses of AOO transients, discussed at the beginning of this evaluation and which were analyzed using NRC approved methodology and the $\pm$ 3 percent tolerance limit, resulted in reactor steam dome pressures exceeding the ASME Code pressure vessel limit for GGNS. The reactor steam dome pressure is an initial condition for the design basis LOCA.

Hydrodynamic Loads on E/RV Discharge Lines:

The proposed increase in safety setpoint tolerances will increase the S/RV opening discharge pressure and flow into the discharge line to the suppression pool. The licensee stated that an evaluation of discharge line loads showed that the increase the tolerances will not result in any allowable stresses being exceeded in the discharge line piping and supports between the S/RVs and the first anchor point. The review of the analysis of the discharge line downstream of the first anchor points showed that the proposed change does not affect the existing analysis of the discharge loads for eight of the 20 S/RV lines. The remaining 12 discharge lines will have an increase in the thrust loads of no more than 0.7 percent. This increase in load is combined with the deadweight, thermal expansion, and other LOCA loads in the structural analysis and the thrust loads are a fraction of the total loads for the structural analysis on the discharge lines. The 0.7 percent increase is very small and is considered negligible by the licensee.

For the quencher where the S/RV discharge flow enters the suppression pool, the loads defined for the quencher during discharge were significantly greater than the Grand Gulf-specific loads because of the conservatism in the GE quencher design. Therefore, the licensee stated that the quencher loads are not affected by the proposed change.

Hydrodynamic Loads on Containment:

The licensee stated that an increase in safety setpoint tolerances will not affect the loads on the containment during S/RV blowdown into the suppression pool. The S/RV blowdown loads on the submerged pressure boundary and submerged structures of containment are based on the peak bubble pressure which is determined with the generic GE methodology for the quencher. Based on a review of the GE methodology and the conservatisms in the methodology, the licensee stated that there is no effect for Grand Gulf on the S/RV pool boundary load definition and the loads currently defined for the submerged structures at Grand Gulf are not affected by the proposed change.

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 stated that two additional valves would be tested for each valve found outside the 3 percent tolerance; however, the number of valves to be tested each refueling outage (18 months currently for GGNS) will be determined using Section XI (Subsections IWA and IWV) of the currently licensed ASME Code, 1980 Edition, with Addenda through Winter 1980 instead of the one-half of all installed valves. Subsection IWV is on inservice testing of nuclear power plants and Table IWV-3510-1 is the testing schedule of safety and relief valves.

The licensee stated that the IST program controls the frequency of testing of valves including the S/RVs and this program is governed by Section XI of the ASME Code. Using this section 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." ANSI/ASME-OM-1981 would require testing five S/RVs and the currently licensed ASME Code would require testing six S/RVs.

As stated in UFSAR Section 5.2.2.10, the licensee had committed to removing and testing at least one-half of the installed S/RVs each refueling outage, and performing testing of additional S/RVs if any valve tested above the current criteria of 1 percent tolerance. However, because of the large number of failures at the 1 percent tolerance, the licensee removed and tested all of the valves during each refueling outage to prevent removal and testing of additional valves from affecting the outage schedule. The failures for the current 1 percent tolerance for the past seven refueling outages was 66, or an average of 9 failures an outage.

The valve failures reported for the previous 7 refueling outages for the safety lift setpoint found outside the proposed 3 percent tolerance was 10 failed valves out of 140 valves tested, or a probability of failure of 0.071 per valve tested. For five refuelings (including the last three refuelings), only one or no valve failed the test and this resulted in 3 failed valves out of 100 valves tested which would be a probability of failure of only 0.03 per valve tested. This smaller failure rate should lead to fewer valves tested in the future and less occupational exposure associated with removing and testing the valves.

4.0 EVALUATION CONCLUSIONS

The staff has reviewed the licensee's responses to the 6 conditions listed in the staff's SE on NEDC-31753P and listed in Section 2.0 above. The licensee's responses are discussed in Section 3.0 above. The staff concludes that the licensee has acceptably addressed the 6 conditions. The staff agrees with the licensee that the 0.7 percent increase in thrust loads for 12 S/RV discharge lines for the proposed 3 percent tolerance is negligible compared to the total loads on the lines.

For the frequency of testing the S/RVs, the licensee has proposed an alternative to the frequency of half the valves each refueling and all the valves in 40 months specified in the staff's SE for NEDC-31753P. The staff had based its judgement of an acceptable frequency on that frequency proposed by the BWROG in NEDC-31753 and the discussion of safety lift setpoint drift for Target Rock S/RVs in the TER attached to the SE. The licensee does not have Target Rock S/RVs at Grand Gulf and has proposed to use the table for testing S/RVs in Section XI, inservice testing of valves, of the current ASME code of record for the plant. The plant has Dikkers S/RVs which have a significantly smaller probability of a tested valve being outside the 3 percent tolerance than the Target Rock S/RVs. Also, at Grand Gulf, by LCO 3.4.4 in the TSs, only 13 of the 20 S/RVs are required to be operable and testing 6 valves by the current Grand Gulf ASME Code of record would have almost half of the required operable valves tested per refueling outage.

Based on this, the staff concludes that proposed alternative testing frequency for Grand Gulf is acceptable.

Therefore, the staff concludes that the proposed amendment to increase the safety lift setpoint tolerances to $\pm$ 3 percent is acceptable. The licensee also proposed to change 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 would be $\pm$ 3 percent. This change is also acceptable.

Approval of these proposed changes are based on the staff's SE for the General Electric topical report, NEDC-31753P. The SE for NEDC-31753P required certain analyses to be performed and the results of these analyses for Grand Gulf 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.

5.0 STATE CONSULTATION

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

6.0 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 and changes surveillance requirements. 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 (61 FR 13524). 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.

7.0 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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