Entergy

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RBG-47350

May 28, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request
Changes to Technical Specification 2.1.1, "Reactor Core SLs"
River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

References

1. GE- Nuclear Energy, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit", MFN 05-021, March 29, 2005
2. Grand Gulf Nuclear Station Unit 1, Issuance of Amendment No. 191, RE.: Extended Power Uprate (pages 324-325), dated July 18, 2012 (TAC NO. ME 4679)

Dear Sir or Madam:

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), Entergy Operations, Inc. (Entergy) is submitting a request for an amendment to the River Bend Station (RBS) Technical Specifications (TS). The proposed amendment reduces the minimum reactor dome pressure associated with the critical power correlation from 785 psig to 685 psig in TS 2.1.1 Safety Limits, and associated Bases, and adds references to address the condition reported by GE Nuclear Energy in Reference 1.

The proposed changes have been evaluated in accordance 10 CFR 50.92(c). It has been determined that the changes involve no significant hazards considerations. Attachment 1 provides the No Significant Hazards Consideration for the change.

Attachment 1 provides a description of the proposed change. Attachment 2 provides the existing TS pages marked up to show the proposed changes. Attachment 3 provides the existing TS Bases pages marked up to show the proposed change (for information only). Attachment 4 contains the Regulatory Commitments.

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NRR

This change has been reviewed and approved by the Onsite Safety Review Committee (OSRC).

Although this request is neither exigent nor emergency, your prompt review is requested.

If you have any questions or require additional information, please contact Mr. Joseph A. Clark at (225) 381-4177.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 29, 2013.

Sincerely,

Eric W. Olson

EWO/JAC/bmb

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. Changes to Technical Specification Bases Pages – For Information Only
4. List of Regulatory Commitments

LAR 2013-07
RBF1-13-0042

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Attachment 1

RBG-47350

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

On March 29, 2005, GE Nuclear Energy (GE) issued a Safety Communication (SC 05-03) in accordance with 10 CFR 21.21(d) to Entergy, invoking a Reportable Condition for Potential Violation of Low Pressure Technical Specification (TS) Safety Limit (SL) (Reference 1). GE identified an unanalyzed condition where a Pressure Regulator Failure Open (PRFO) – Maximum Demand Anticipated Operational Occurrence (AOO) may cause a TS SL to be violated since reactor dome pressure could drop below the current TS SL 2.1 value of 785 psig while reactor power is above 25% of rated. GE identified that even plants with an MSIV low pressure isolation setpoint ≥ 785 psig may experience an AOO that potentially could violate the SL.

GE informed the affected licensees that their recent code calculations showed that during the PRFO transient, many of the plant's reactor pressure would fall below the TS current pressure safety limit. Depending upon the Low Pressure Isolation Setpoint (LPIS), the margin to the low pressure TS SL may not be adequate. GE recommended lowering the low pressure TS safety limit to 700 psia (685 psig), as supported by the expanded GEXL correlation applicability range for GE14 and GNF2 fuels that are currently co-resident in RBS Reactor Core.

GE advanced fuel designs have an NRC approved critical power correlation with a lower-bound pressure significantly below the 785 psig reactor steam dome pressure specified in TS Reactor Core Safety Limits 2.1.1.1 and 2.1.1.2. Entergy proposes to utilize this fact and reduce the reactor steam dome pressure consistent with the approved lower-bound pressure for the GE fuel comprising the River Bend Station (RBS) core. GE fuel utilizes the GEXL 14 and GEXL 17 critical power correlations, with an approved pressure range from 700 to 1400 psia (685 to 1385 psig). Entergy has determined that changing the pressure limits in TS Safety Limit 2.1 to 685 psig as permitted by References 2 and 3, provides greater margin for the PRFO transient, such that the dome pressure will remain above the revised TS 2.1 limit.

Accordingly, pursuant to 10 CFR 50.90, Entergy hereby requests an amendment to the RBS Operating License Technical Specifications (TS). The proposed amendment revises the reactor dome pressure from 785 psig to 685 psig in TS Safety Limits 2.1.1 and TS Bases 2.1.1 to resolve the pressure regulator failure open (PRFO) transient.

RBS proposed TS change follows the NRC approved TS change for Grand Gulf Nuclear Station by License Amendment No. 191, dated July 18, 2012 (TAC NO. ME 4679).

2.0 PROPOSED CHANGE

2.1 The proposed changes to the Technical Specifications are as follows:

The reactor dome pressure 785 psig is revised to 685 psig in TS Safety Limits 2.1.1.1 and 2.1.1.2.

2.2 The TS BASES changes will be incorporated into TS upon receipt of the NRC approved License Amendment in accordance with TS 5.5.6, Bases Control Program. The BASES pages are provided for NRC information only.

The referenced GESTAR II compliant or NRC approved correlations, References 2 and 3, are contained in fuel design information reports in accordance with GESTAR-II Section 1.2.7. Therefore, these documents are included by reference.

3.0 BACKGROUND

GE References 2 and 3 documented the expanded pressure range for GEXL correlations for the current co-resident Fuels, GE14 and GNF2 in the RBS Reactor Core. Subsequently, GE Part 21 (Reference 1) identified an Anticipated Operational Occurrence (AOO) due to Pressure Regulator Maximum Demand Open (PRFO) transient that could potentially result in violations of the low pressure Safety Limits in TS 2.1.1 and 2.1.2 as it is currently set at 785 psig. .

Entergy reviewed the GEXL14 and GEXL17 correlations approved by the NRC in NEDC-32851P-A, Rev. 4 (Reference 2) and NEDC-33292P, Rev. 3 (References 3 and 5), and concluded that the GEXL14 and GEXL17 correlations apply for GE14 and GNF2 fuel respectively. Since RBS core has both GE14 and GNF2 fuel, RBS is proposing to reduce the current 785 psig reactor dome pressure limit in Safety Limits 2.1.1 and 2.1.2 and associated TS Bases to 685 psig, using GESTAR II or NRC approved documents (References 2, 3 and 5). The proposed reduction in dome pressure is consistent with that used in the GESTAR II or NRC approved critical power correlations for GE14 and GNF2 fuel designs.

This change offers a greater pressure margin such that the reactor pressure remains above the proposed low pressure safety limit of 685 psig in the event of a PRFO transient. Thus the proposed change, in addition to compliance with the updated GEXL pressure range documented by GEH for GE 14 and GNF2 fuel designs, resolves the concern reported by GEH in Safety Communication SC05-03.

The proposed TS change follows the TS changes previously approved for Grand Gulf by License Amendment 191, dated July 18, 2012 (TAC NO. ME 4679), Reference 4.

4.0 TECHNICAL ANALYSIS

The pressure regulator failure open event involves the failure of the pressure regulator in the open direction causing the turbine control valves to fully open, including the bypass valves. This causes the reactor to depressurize rapidly. When, the MSL low pressure setpoint is reached, the MSIVs start to close and a reactor scram occurs. The scram terminates the event and compliance with the SL is automatically restored as reactor power is quickly reduced to below 23.8%. The fuel cladding integrity is never challenged because in pressure decrease events like this in a BWR, fuel critical power is rising and therefore MCPR rises during the event.

TS Safety Limits are specified to ensure that acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). Safety Limits are set such that fuel cladding integrity is maintained and no significant fuel damage would occur if the Safety Limits are not exceeded. In accordance with the standard Improved Technical Specifications (ITS), RBS specifies Safety Limits in TS 2.1.1 to require that thermal power shall be ≤ 23.8% Rated Thermal Power (RTP) when reactor steam dome pressure is < 785 psig or core flow is < 10% of RTP. This Safety Limit was introduced to ensure the validity of MCPR calculations when power is > 23.8% and the reactor pressure is within the validity range of GEXL correlation. GEH has updated the validity range of the GEXL 14 and GEXL 17 Correlations via References 2, 3 and 5, which allows the pressure to be reduced to 685 psig instead of 785 psig for MCPR calculations to be valid. Therefore a greater pressure range is available for transients to demonstrate compliance with MCPR limits. Thus, the proposed change offers a greater pressure margin for PRFO than what is currently available.

The proposed changes to the TS Safety Limits are based upon GESTAR II compliant reports (Reference 2 and 3). The change was also approved for Grand Gulf Nuclear Station Unit 1 (Reference 4). This proposed TS change follows NRC approved methodology.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy requests a License Amendment to RBS Operating License Technical Specifications (TS) to make the following changes:

- Reduce the reactor dome pressure from 785 psig to 685 psig in TS 2.1.1.

Entergy has evaluated the proposed TS changes using the criteria in 10 CFR 50.92, and has determined that the proposed changes do not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration determination.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Decreasing the reactor dome pressure limit in Technical Specification Safety Limits 2.1.1 for reactor Rated Thermal Power range effectively expands the validity range for the GEXL 14 and GEXL 17 correlations and the calculation of Minimum Critical Power Ratio Safety Limit (MCPR). MCPR rises during the pressure reduction following the scram that terminates the PRFO transient. Since the change does not involve a modification of any plant hardware, the probability and consequence of the PRFO transient are essentially unchanged. The reduction in the reactor dome pressure safety limit from 785 psig to 685 psig provides greater margin to accommodate the pressure reduction during the transient within the revised TS limit.

   The proposed change will continue to support the validity range for the GEXL correlations applied at RBS and the calculation of MCPR as approved. The proposed TS revision involves no significant changes to the operation of any systems or components in normal, accident or transient operating conditions. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed reduction in the reactor dome pressure safety limit from 785 psig to 685 psig is a change based upon previously approved documents and does not involve changes to the plant hardware or its operating characteristics. As a result, no new failure modes are being introduced. Therefore, the change does not introduce a new or different kind of accident from those previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The margin of safety is established through the design of the plant structures, systems, and components, and through the parameters for safe operation and setpoints for the actuation of equipment relied upon to respond to transients and design basis accidents. The proposed change in reactor dome pressure enhances the safety margin, which protects the fuel cladding integrity during a depressurization transient, but does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change does not alter the behavior of plant equipment, which remains unchanged. The available pressure range is expanded by the change, thus offering greater margin for pressure reduction during the transient

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based upon the above, RBS concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements and Criteria

10 CFR 50, Appendix A provides criteria for Emergency Core Cooling System (ECCS) performance and 10 CFR 50.36, Technical Specifications, requires safety system settings to ensure the integrity of the reactor pressure boundary during normal and abnormal operations and to mitigate transient and accident conditions. The proposed decrease in the reactor dome pressure limit in TS 2.1.1 follows the requirements cited above and ensures the fuel cladding integrity is maintained.

In conclusion, based on the considerations discussed above, (1) there is a 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 NRC'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.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22 (9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 Precedence

As previously noted the proposed TS change follows the TS changes previously approved for Grand Gulf Nuclear Station by License Amendment 191, dated July 18, 2012 (TAC NO. ME 4679) (Reference 4).

7.0 COORDINATION WITH PENDING TS CHANGES

There are no pending proposed TS changes that are being filed for license amendment that would impact the proposed TS changes.

8.0 REFERENCES

1. GE- Nuclear Energy, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit", MFN 05-021, March 29, 2005

2. NEDC-32851P-A, Rev. 4, "GEXL14 Correlation for GE14 Fuel", dated September 2007

3. NEDC-33292P, Rev 3, "GEXL17 Correlation for GNF2 Fuel", dated June 2009

4. Grand Gulf Nuclear Station Unit 1, Issuance of Amendment No. 191, RE.: Extended Power Uprate (pages 324-325), dated July 18, 2012 (TAC NO. ME 4679)

5. NEDC-33270P, Rev. 4, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)", dated October 2011

Attachment 2

RBG-47350

Technical Specification Markup

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < ~~785~~685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager and the corporate executive responsible for overall plant nuclear safety.

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Attachment 3

RBG-47350

Technical Specification BASES Mark-up

(For Information Only)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR SL is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR SL.

2.1.1.1 Fuel Cladding Integrity

The use of the fuel vendor's critical power correlations are valid for critical power calculations at pressures ≥ ~~785~~ 685 psig and core flows ≥ 10% of rated flow (Ref. 2.7 and 8). For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to ~~800~~ 700 psia indicate that the fuel

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% of original RTP. Thus, a THERMAL POWER limit of 23.8% RTP for reactor pressure < ~~785~~ 685 psig is conservative. Because of the design thermal hydraulic compatibility of the reload fuel designs with the cycle 1 fuel, this justification and the associated low pressure and low flow limits remain applicable for future cycles of cores containing these fuel designs.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating limit MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. Reference 6 describes the methodology used in determining the MCPR SL.

The calculated MCPR safety limit is reported to the customary three significant digits (i.e., X.XX); the MCPR operating limit is developed based on the calculated MCPR safety limit to ensure that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The fuel vendor's critical power correlations are based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the fuel vendor's correlation provide a reasonable degree of assurance that 99.9% of the rods in the core would not be susceptible to transition boiling during

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure-Low (continued)

is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below ~~785~~ 685 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 23.8% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure-Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure-Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 6 valves.

1.c. Main Steam Line Flow-High

Main Steam Line Flow-High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow-High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 1). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from

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Attachment 4

RBG-47350

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
TS BASES changes will be incorporated into TS upon receipt of the NRC approved License Amendment in accordance with TS 5.5.6, Bases Control Program.	X		Implementation