August 27, 2007

Technical Specification Task Force (TSTF)
11921 Rockville Pike, Suite 100
Rockville, MD 20852

SUBJECT: DENIAL OF TSTF-495, REVISION 0, “BASES CHANGE TO ADDRESS GE PART 21 SC05-03.” DOCKET NO: PROJ0753 (TAC MD2672)

Dear Members of the TSTF:

By letter (ADAMS Accession No. ML061990227), dated July 18, 2006, you requested that the Nuclear Regulatory Commission (NRC) review and approve TSTF Traveler 495, Revision 0, “Bases Change to Address GE Part 21SC05-03.”

After careful review, the NRC staff has concluded that your request cannot be approved. The basis for this finding is documented in the enclosed Safety Evaluation. As a result of the completion of NRC review associated with TSTF-495, Revision 0, TAC# MD2672 has been formally closed out. Please contact me with any questions or concerns at 301-415-1932 or at TJK1@nrc.gov.

Sincerely,

Timothy J. Kobetz, Chief ***/RA/***
Technical Specifications Branch
Division of Inspections & Regional Support
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

cc: See next page

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Safety Evaluation

<u>DISTRIBUTION</u>: ITSB R/F RidsNrrDirsItsb

ADAMS ACCESSION NUMBER: ML072340113

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SAFETY EVALUATION
U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
TECHNICAL SPECIFICATION TASK FORCE CHANGE TSTF-495, REVISION 0,
"BASES CHANGE TO ADDRESS GE PART 21 SC05-03."
DOCKET NO: PROJ0753 (TAC MD2672)

1.0 INTRODUCTION

By application dated July 18, 2006 (Reference 1), the Technical Specifications Task Force (TSTF) submitted an Improved Standard Technical Specifications Change Traveler, TSTF-495, Reference 1, entitled "Bases Change to Address GE Part 21 SC05-03" (Agenywide Documents Access and Management System (ADAMS) Accession No. ML061990227). The proposed change would modify the Bases section for Reactor Core Safety Limit 2.1.1.1 in the Standard Technical Specifications (STS) for BWR/4, NUREG-1433, (Reference 2) and for BWR/6, NUREG-1434 (Reference 3). It is claimed that the proposed change would allow a brief violation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) for a depressurization transient caused by pressure regulator failure. The Part 21 notification, submitted by the General Electric Nuclear Energy (GENE) Company on March 29, 2005, pursuant to 10 CFR 21.21(d), described a previously-analyzed event that could result in the SLMCPR being violated for a short period of time.

In the STS, Safety Limit (SL) 2.1.1.1 requires that, with the reactor steam dome pressure below 785 psig or core flow below 10 percent of rated, the reactor thermal power (RTP) shall be less than or equal to 25 percent of rated. These requirements preclude the need for critical power ratio (CPR) calculations below 785 psig and provide fuel cladding protection during start-up conditions, since the GEXL Critical Power correlation, used in plant safety analyses, is not approved as a licensing model below 785 psig, and at less than 25 percent RTP, the CPR limits are not applicable. If the conditions are not met, they must be restored within 2 hours, and the reactor must be shut down.

In the Part 21 notification, GENE reported that earlier computer analytical models predicted that during a pressure regulator failure-open (PRFO) event, a reactor level swell would result in a turbine trip and subsequent reactor scram. However, newer computer models predict that level may not increase to the turbine trip setpoint and may be terminated by main steam isolation valve (MSIV) closure scram at the low pressure isolation setpoint (LPIS). Thus, steam dome pressure could decrease to below 785 psig briefly with thermal power still above 25 percent of rated, in violation of the safety limit.

In TSTF-495, justification is provided based on preliminary Owners Group and GE analyses which show that the Critical Power Ratio will actually increase, resulting in additional thermal margin. The analyses, which utilize the GEXL Critical Power Correlation, predict that the critical bundle power increases during depressurization events, while the actual bundle power decreases.

Enclosure

2.0 REGULATORY EVALUATION

Title 10 of the Code of Federal Regulations (CFR), Part 50, Section 50.36, *Technical Specifications*, subpart 50.36(c)(1)(i)(A) *Safety Limits* defines safety limits for nuclear reactors as limits on important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.

A summary statement of the bases or reasons for technical specifications, other than those addressing administrative controls, must be included in each facility license application, as required by 10CFR50.36(a). The technical specifications are considered part of the operating license. Bases to the technical specifications provide additional details and insights into the intent of the specifications, but are not actually part of the technical specifications.

TSTF-495 proposes to modify the applicable safety analysis portion of the Reactor Core Safety Limit Bases to clarify that the Safety Limit does not apply to momentary depressurization transients. The Applicable Safety Analysis of the Bases portion of the Primary Containment Isolation Instrumentation specification would be modified to delete a sentence that states that the Main Steam Line pressure - Low Function serves to ensure that Safety Limit 2.1.1.1 is not exceeded.

Standard Technical Specifications, Section 5.5.14(b)(1), "Technical Specifications (TS) Bases Control Program," states that licensees may make changes to Bases without prior NRC approval, provided the changes do not involve a change in the TS incorporated in the license. The proposed change to the TS Bases has the effect of relaxing, and hence, changing, the TS Safety Limit. An exception to a stated TS safety limit must be made in the TS and not in the TS Bases. In addition, a potential exists that the requested change in the TS Bases could have an adverse effect on maintaining the reactor core safety limits specified in the Technical Specifications, and thus, may result in violation of the stated requirements. Therefore, from a regulatory standpoint, the proposed change to the TS Bases is not acceptable.

3.0 TECHNICAL EVALUATION

In General Electric (GE) fuel cladding integrity evaluations, the GEXL critical power correlation is applicable for all critical power calculations at pressures greater than or equal to 785 psig and core flows greater than or equal to 10 percent of rated flow. This restriction is a result of limited test data at low pressure and flow conditions. For operation at low pressures or low flows, such as startup, an alternate basis is used, as documented in B 2.1.1 of the STS.

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9 percent of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The SLMCPR is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using approved General Electric Critical Power correlations (i.e, GEXL).

The GE Part 21 report SC05-03 describes a revised accident analysis for the Pressure-Regulator Failure in the Open (PRFO) direction transient which results in a temporary pressure drop below 785 psig while reactor power is greater than 25 percent of Rated Thermal Power, resulting in a violation of Reactor Core Safety Limit 2.1.1.1. Using previous GE analysis methods, the PRFO transient was terminated by direct turbine trip and subsequent reactor scram, which resulted from the reactor water level swell following the PRFO. Specifically, in the postulated event the Electro Hydraulic Control (EHC) system pressure regulator fails in such a manner that it results in a demand to open the turbine steam admission valves, i.e., stop valves, control valves, and bypass valves. As a result, the reactor depressurizes, which causes the formation of voids within the reactor core. The core voiding increases the reactor water level until it reaches the level of the main turbine trip setpoint. The turbine will trip, which in turn sends a direct signal (via the stop valve position switches) to the reactor protection system (RPS) and the reactor will automatically shut down, terminating the transient. New analysis methodologies predict a different series of events: the PRFO occurs as before and the reactor depressurizes; however, the reactor level does not swell to the setpoint to cause a main turbine trip. Therefore, the transient is not terminated as quickly as the earlier methods predicted and the reactor depressurization continues until pressure reaches the setpoint of the Main Steam Line Isolation Valve (MSIV) Closure containment isolation signal. The MSIV closure is a direct input, via their position switches, to the RPS. The reactor scrams and the transient is terminated. Under this series of events, the delay in the termination of the transient introduces the possibility for the reactor pressure to drop below 785 psig before the reactor has been shutdown. Therefore, the reactor power could still be above 25 percent when the reactor steam dome pressure is less than 785 psig, violating the Safety Limit. This condition would only be present for a short time before the RPS is initiated by the MSIV closure.

By letter dated April 7, 2007 (Reference 6), the applicant responded to an NRC staff request for additional information (RAI) regarding additional analyses which were performed for the Part 21 evaluation or subsequently using different plant parameters from those originally assumed. The staff was concerned that other plant transients could potentially result in a larger steam dome depressurization or for longer duration of SL violation than the PRFO transient discussed. In its response, the applicant stated that all Final Safety Analysis Report Anticipated Operational Occurrence (AOO) events that may result in a steam dome pressure decrease were reviewed to determine if they could result in the steam dome pressure dropping below the SL. The PRFO transient was the only transient identified that could cause a drop in pressure below 785 psig with reactor power above 25 percent RTP.

The staff agrees with the applicant’s position that the PRFO transient does not threaten fuel cladding integrity, since the margin to SLMCPR increases with decreasing reactor pressure.

However, the staff is concerned that in some depressurization events which occur at or near full power, there may be enough bundle stored energy to cause some fuel damage. If a reactor scram does not occur automatically, the operator may have insufficient time to recognize the condition and to take the appropriate actions to bring the reactor to a safe configuration.

4.0 CONCLUSION

The proposed change to the STS Bases modifies the corresponding Technical Specifications by providing an exception to the explicit safety limit. The staff therefore concludes that the proposed TSTF-495 is unacceptable.

5.0 REFERENCES

1. Technical Specifications Task Force, Standard Technical Specification Change Traveler- 495, "Bases Change to Address GE Part 21 SC05-03," Revision 0 (ADAMS Accession No. ML061990227).

2. NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4," Revision 3.1, dated June 2004 (ADAMS Accession No. ML062510089).

3. NUREG-1434, "Standard Technical Specifications, General Electric Plants, BWR/6," Revision 3.1, dated June 2004 (ADAMS Accession No. ML062510089).

4. GE Energy - Nuclear, 10 CFR Part 21 Communication, SC05-03, "Potential to Exceed Low Pressure Technical Specifications Safety Limits," March 29, 2005.

5. Letter from T. J. Kobetz (NRC) to the Technical Specifications Task Force, "Request for Additional Information (RAI) Regarding TSTF-495, Rev. 0, 'Bases Change to Address GE Part 21 SC05-03'," dated January 11, 2007 (ADAMS Accession No. ML070100146).

6. Letter from Technical Specifications Task Force, “Response to NRC "Request for Additional Information (RAI) Regarding TSTF-495, Revision 0, 'Bases Change to Address GE Part 21 SC05-03'," dated January 11, 2007, April 7, 2007 (ADAMS Accession No. ML070960090).

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Date: August 10, 2007