[7950-01-P]

NUCLEAR REGULATORY COMMISSION
Notice of Opportunity To Comment on Model Safety Evaluation on
Technical Specification Improvement To Revise Control Rod Notch Surveillance Frequency,
Clarify SRM Insert Control Rod Action, and
Clarify Frequency Example

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for comment.

SUMMARY: Notice is hereby given that the staff of the Nuclear Regulatory Commission (NRC) has prepared a model safety evaluation (SE) relating to the revision of Standard Technical Specifications (STS), NUREG-1433 (BWR/4) and NUREG-1434 (BWR/6). Specifically the SE addresses: (1) the revision of the TS surveillance requirement (SR) 3.1.3.2 frequency in STS 3.1.3, "Control Rod OPERABILITY", (2) a clarification to the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in STS 3.3.1.2, Required Action E.2, "Source Range Monitor Instrumentation" (NUREG-1434 only), and (3) the revision of Example 1.4-3 in STS Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension. The NRC staff has also prepared a model license amendment request and a model no significant hazards consideration (NSHC) determination relating to this matter. The purpose of these models are to permit the NRC to efficiently process amendments that propose to modify TS control rod SR testing frequency, clarify TS control insertion requirements, and clarify SR frequency discussions. Licensees of nuclear power reactors to which the models apply could then request amendments, confirming the applicability of the SE and NSHC determination to their plant licensing basis. The NRC staff is requesting comment on the model SE, model amendment request, and model NSHC

determination prior to announcing their availability for referencing in license amendment applications.

DATES: The comment period expires [insert date 30 days from date of publication in the *[Federal Register]*. Comments received after this date will be considered if it is practical to do so, but the Commission is able to ensure consideration only for comments received on or before this date.

ADDRESSES: Comments may be submitted either electronically or via U.S. mail. Submit written comments to Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, Mail Stop: T-6 D59, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Hand deliver comments to: 11545 Rockville Pike, Rockville, Maryland, between 7:45 a.m. and 4:15 p.m. on Federal workdays. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike (Room O-1F21), Rockville, Maryland. Comments may be submitted by electronic mail to CLIIP@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Timothy Kobetz, Mail Stop: O-12H2, Techncial Specifications Branch, Division of Inspection & Regional Support, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, telephone 301-415-1932.

SUPPLEMENTARY INFORMATION:

Background

Regulatory Issue Summary 2000-06, “Consolidated Line Item Improvement Process for Adopting Standard Technical Specification Changes for Power Reactors,” was issued on March 20, 2000. The consolidated line item improvement process (CLIIP) is intended to improve the efficiency of NRC licensing processes, by processing proposed changes to the

STS in a manner that supports subsequent license amendment applications. The CLIIP includes an opportunity for the public to comment on proposed changes to the STS after a preliminary assessment by the NRC staff and finding that the change will likely be offered for adoption by licensees. This notice solicits comment on a proposed change to the STS that modifies a TS control rod SR testing frequency, clarifies TS control rod insertion requirements, and clarifies SR frequency discussions. The CLIIP directs the NRC staff to evaluate any comments received for a proposed change to the STS and to either reconsider the change or announce the availability of the change for adoption by licensees. Licensees opting to apply for this TS change are responsible for reviewing the staff's evaluation, referencing the applicable technical justifications, and providing any necessary plant-specific information. Each amendment application made in response to the notice of availability will be processed and noticed in accordance with applicable rules and NRC procedures.

This notice involves the modification of TS control rod SR testing frequency, clarification of TS control insertion requirements, and clarification of SR frequency discussions. This change was proposed for incorporation into the standard technical specifications by the Owners Groups participants in the Technical Specification Task Force (TSTF) and is designated TSTF-475 Revision 1. TSTF-475 Revision 1 can be viewed on the NRC's web page at http://www.nrc.gov/reactors/operating/licensing/techspecs.html.

Applicability

This proposed TS change to modify TS control rod SR testing frequency, clarify TS control insertion requirements, and clarify SR frequency discussions is applicable to BWR NSSS plants. The CLIIP does not prevent licensees from requesting an alternative approach or proposing the changes without the attached model SE and the NSHC. Variations from the approach recommended in this notice may, however, require additional review by the NRC staff and may increase the time and resources needed for by the NRC staff and may increase the time and resources needed for the review.

Public Notices

This notice requests comments from interested members of the public within 30 days of the date of publication in the *Federal Register*. After evaluating the comments received as a result of this notice, the staff will either reconsider the proposed change or announce the availability of the change in a subsequent notice (perhaps with some changes to the safety evaluation, model application or the proposed no significant hazards consideration determination as a result of public comments). If the staff announces the availability of the change, licensees wishing to adopt the change must submit an application in accordance with applicable rules and other regulatory requirements. For each application the staff will publish a notice of consideration of issuance of amendment to facility operating licenses, a proposed no significant hazards consideration determination, and a notice of opportunity for a hearing. The staff will also publish a notice of issuance of an amendment to operating license to announce the modification of the TS control rod SR testing frequency, TS control rod insertion requirements, and SR frequency discussions for each plant that receives the requested change.

Proposed Safety Evaluation

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

Consolidated Line Item Improvement Program

Technical Specification Task Force (TSTF) Change TSTF-475, Revision 1

Control Rod Notch Testing Frequency and

Source Range Monitor Technical Specification Action to Insert Control Rods

1.0 INTRODUCTION

By letter dated August 30, 2004, BWR OWNERS Group (BWROG) submitted a request for changes to NUREG-1433, Standard Technical Specifications General Electric Plants, BWR/4 (Reference 1), and NUREG-1434, Standard Technical Specifications General Electric Plants, BWR/6 (Reference 2). The proposed changes would: (1) revise the TS control rod notch surveillance frequency in TS 3.1.3, “Control Rod OPERABILITY,” (2) clarify the TS requirement for inserting control rods for one or more inoperable SRMs in MODE 5, and (3) revise one Example in Section 1.4 “Frequency” to clarify the applicability of the 1.25 surveillance test interval extension.

These changes are based on Technical Specifications Task Force (TSTF) change traveler TSTF-475, Revision 1, that proposes revisions to the reference BWR standard technical specifications (STS) by: (1) revising the frequency of SR 3.1.3.2, notch testing of each fully withdrawn control rod, from “7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM” to “31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM”, (2) adding the word “fully” to LCO 3.3.1.2 Required Action E.2 (NUREG- 1434 only) to clarify the requirement to fully insert all insertable control rods in core cells containing one or more fuel assemblies when the associated SRM instrument is inoperable, and (3) revising Example 1.4-3 in Section 1.4 “Frequency” to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the “FREQUENCY” column.

The purpose of these surveillances is to confirm control rod insertion capability which is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. Control rods and control rod drive (CRD) Mechanism (CRDM), by which the control rods are moved, are components of the CRD System, which is the primary reactivity control system for the reactor. By design, the CRDM is highly reliable with a tapered design of the index tube which is conducive to control rod insertion.

A stuck control rod is an extremely rare event and industry review of plant operating experience did not identify any incidents of stuck control rods while performing a rod notch surveillance test.

The purpose of these revisions is to reduce the number of control rod manipulations and, thereby, reduce the opportunity for reactivity control events.

2.0 REGULATORY EVALUATION

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC) 29, Protection against anticipated occurrence, requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in an event of anticipated operational occurrences. The design relies on the CRDS to function in conjunction with the protection systems under anticipated operational occurrences, including loss of power to all recirculation pumps, tripping of the turbine generator, isolation of the main condenser, and loss of all offsite power. The CRDS provides an adequate means of inserting sufficient negative reactivity to shut down the reactor and prevent exceeding acceptable fuel design limits during anticipated operational occurrences. Meeting the requirements of GDC 29 for the CRDS prevents occurrence of mechanisms that could result in fuel cladding damage such as severe overheating, excessive cladding strain, or exceeding the thermal margin limits during anticipated operational occurrences. Preventing excessive cladding damage in the event of anticipated transients ensures maintenance of the integrity of the cladding as a fission product barrier.

3.0 TECHNICAL EVALUATION

In order to perform this SE, the NRC staff reviewed the following information provided by the BWROG to justify the submitted license amendment request for STS NUREG-1433 and NUREG-1434 to revise the weekly control rod notch frequency to monthly, clarify the SRM TS action for inserting control rods, and the applicability of the 25% allowance in Example 1.4-3. Specifically, the following documents were reviewed during the NRC staff's evaluation:

- TSTF letter TSTF-04-07 - Provided a description of the proposed NUREG-1433 and NUREG-1434 changes. TSTF-475 would change the weekly rod notch frequency to monthly, clarify the SRM TS actions for inserting control rods, and clarify the applicability of the 25% allowance in Example 1.4-3 (Reference 3).
- TSTF letter TSTF-06-13 - Provided responses to NRC staff request for additional information (RAI) on (1) industry experience with identifying stuck rods, (2) tests that would identify stuck rods, (3) continue compliance with SIL 139, (4) industry experience on collet failures, and (4) applying the 25% grace period to the 31 day control rod notch SR test frequency (Reference 4).
- BWROG letter BWROG-06036 – Provided the GE Nuclear Energy Report, "CRD Notching Surveillance Testing for Limerick Generating Station," in which CRD notching frequency and CRD performance were evaluated (Reference 5).

- TSTF letter TSTF-07-19 - Provided response to NRC staff RAI on CRD performance in Control Cell Core (CCC) designed plants, including TSTF-475, Revision 1 (Reference 6).

The CRD System is the primary reactivity control system for the reactor. The CRD System, in conjunction with the Reactor Protection System, provides the means for the reliable control of reactivity changes to ensure under all conditions of normal operation, including anticipated operational occurrences that specified acceptable fuel design limits are not exceeded. Control rods are components of the CRD System that have the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System.

The CRD System consists of a CRDM, by which the control rods are moved, and a hydraulic control unit (HCU) for each control rod. The CRDM is a mechanical hydraulic latching cylinder that positions the control blades. The CRDM is a highly reliable mechanism for inserting a control rod to the full-in position. The collet piston mechanism design feature ensures that the control rod will not be inadvertently withdrawn. This is accomplished by engaging the collet fingers, mounted on the collet piston, in notches located on the index tube. Due to the tapered design of the index tube notches, the collet piston mechanism will not impede rod insertion under normal insertion or scram conditions.

The collet retainer tube (CRT) is a short tube welded to the upper end of the CRD which houses the collet mechanism which consist of the locking collet, collet piston, collet return spring and an unlocking cam. The collet mechanism provides the locking/unlocking mechanism that allows the insert/withdraw movement of the control rod. The CRT has three primary functions: a) to carry the hydraulic unlocking pressure to the collet piston, b) to provide an outer cylinder, with a suitable wear surface for the metal collet piston rings, and c) to provide mechanical support for the guide cap, a component which incorporates the cam surface for holding the collet fingers open and also provides the upper rod guide or bushing.

According to the BWROG, at the time of the first CRT crack discovery in 1975 each partially or fully withdrawn operable control rod was required to be exercised one notch at least once each week. It was recognized that notch testing provided a method to demonstrate the integrity of the CRT. Control rod insertion capability was demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal.

It was determined that during scrams, the CRT temperature distribution changes substantially at reactor operating conditions. Relatively cold water moves upward through the inside of the CRT and exits via the flow holes into the annulus on the outside. At the same time hot water from the reactor vessel flows downward on the outside surface of the CRT. There is very little mixing of the cold water flowing from the three flow holes into the annulus and the hot water flowing downward. Thus, there are substantial through wall and circumferential temperature gradients during scrams which contribute to the observed CRT cracking.

Subsequently, many BWRs have reduced the frequency of notch testing for partially withdrawn control rods from weekly to monthly. The notch test frequency for fully withdrawn control rods are still performed weekly. The change, for partially withdrawn control rods, was made because of the potential power reduction required to allow control rod movement for partially withdrawn control rods, the desire to coordinate scheduling with other plant activities, and the fact that a large sample of control rods are still notch tested on the weekly basis. The operating experience related to the changes in CRD performance also provided additional justification to reduce the notch test frequency for the partially withdrawn control rods.

In response to the NRC staff RAIs and to support their position to reduce the CRD notch testing frequency, the BWROG provided plant data and GE Nuclear Energy report, CRD Notching Surveillance Testing for Limerick Generating Station (CRDNST). The GE report provided a description of the cracks noted on the original design CRT surfaces. These cracks, which were later determined to be intergranular, were generally circumferential, and appeared with greatest frequency below and between the cooling water ports, in the area of the change in wall thickness. Subsequently, cracks associated with residual stresses were also observed in the vicinity of the attachment weld. Continued circumferential cracking could lead to 360 degree severance of the CRT that would render the CRD inoperable which would prevent insertion, withdrawal or scram. Such failure would be detectable in any fully or partially withdrawn control rod during the surveillance notch testing required by the Technical Specifications. To a lesser degree, cracks have also been noted at the welded joint of the interim design CRT but no cracks haven been observed in the final improved CRT design. In a request for additional information, BWROG response of being unable to find a collet housing failure since 1975 supported the NRC staff review of not finding a collet housing failure. To date, operating experience data shows no reports of a severed CRT at any BWR. No collet housing failures have been noted since 1975. On a numerical basis for instance, based on BWROG assumption that there are 137 control rods for a typical BWR/4 and 193 control rods for a typical BWR/6, the yearly performance would be 6590 rod notch tests for a BWR/4 plant and 9284 for a BWR/6 plant. For example, if all BWRs operating in the US are taken into consideration, the yearly performances of rod notch data would translate into approximately 240,000 rod notch tests without detecting a failure.

In addition, the IGSCC crack growth rates were evaluated, at Limerick Generating Station, using GE's PLEDGE model with the assumption that the water chemistry condition is based on GE recommendations. The model is based on fundamental principles of stress corrosion cracking which can evaluate crack growth rates as a function of water oxygen level, conductivity, material sensitization and applied loads. It was determined that the additional time of 24 days represented an additional 10 mils of growth in total crack length. The small difference in growth rate would have little effect on the behavior between one notch test and the next subsequent test. Therefore, from the materials perspective based on low crack growth rates, a decrease in the notch test frequency would not affect the reliability of detecting a CRDM failure due to crack growth.

Also, the BWR scram system has extremely high reliability. In addition to notch testing, scram time testing can identify failure of individual CRD operation resulting from IGSCC-initiated cracks and mechanical binding. Unlike the CRD notch tests, these single rod scram tests cover the other mechanical components such as scram pilot solenoid operated valves, the scram inlet and outlet air operated valves, and the scram accumulator, as well as operation of the control rods. Thus, the primary assurance of scram system reliability is provided by the scram time testing since it monitors the system scram operation and the complete travel of the control rod.

Also, the HCUs, CRD drives, and control rods are also tested during refueling outages, approximately every 18-24 months. Based on the data collected during the preceding cycle of operation, selected control rod drives, are inspected and, as required, their internal components are replaced. Therefore, increasing the CRD notch testing frequency to monthly would have very minimal impact on the reliability of the scram system.

The NRC staff has reviewed the BWROG TSTF's proposal to amend the TS SR 3.1.3.2, "Control Rod OPERABILTY" from seven days to monthly. Based on the following evaluation condition: (1) slow crack growth rate of the CRT; (2) the improved CRT design; (3) a higher reliable method (scram time testing) to monitor CRD scram system functionality; (4) GE chemistry recommendations; and (5) no known CRD failures have been detected during the notch testing exercise, the NRC staff concluded that the changes would reduce the number of control rod manipulations thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRD system. Therefore, the NRC staff finds the change acceptable with the commitment to implement GE water quality for the CRD system recommendations. Furthermore, the utilities should consider the replacement of the CRT, when possible, with the GE CRT improved design.

The NRC staff has reviewed the BWROG TSTF-475 proposal to amend the NUREG-1434, Specification 3.3.1.2, Required Action E.2 from "Initiate action to insert all insertable control rods in core cells containing one or more fuel assemblies" to "Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies." The NRC staff finds the revision acceptable because the requirement to insert control rods is meant to require control rods to be fully inserted and adding “fully” does not change but clarifies the intent of the action.

The NRC staff has reviewed the BWROG TSTF-475 proposal to amend Example 1.4-3 in Section 1.4 “Frequency,” to make the 1.25 provision in SR 3.0.2 to be equally applicable to time periods specified in the “FREQUENCY” column and in the NOTE in the “SURVEILLANCE” column. The NRC staff finds this change acceptable since the revision would make it consistent with the definition of specified “Frequency” provided in the second paragraph of Section 1.4 which states that the specified “Frequency” is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The specified “Frequency” consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.”

3.1 Conclusion

The NRC staff has reviewed the licensee’s proposal to amend existing TS sections SR 3.1.3.2, “Control Rod OPERABILTY,” NUREG- 1434, LCO 3.3.1.2 Required Action E.2, “Source Range Monitor (SRM) Instrumentation,” and Example 1.4-3, “Frequency” applicable to SR 3.0.2. The NRC staff has concluded that the TS revisions will have a minimal affect on the high reliability of the CRD system while reducing the opportunity for potential reactivity events;

thus, meeting the requirement of CFR, Part 50, Appendix A, GDC 29. Therefore, the staff concludes that the amendment request is acceptable.

Based on the considerations discussed above, the Commission has concluded 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the [ ] State official was notified of the proposed issuance of the amendment. The State official had [(1) no comments or (2) the following comments - with subsequent disposition by the staff].

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards considerations, and there has been no public comment on the finding [FR]. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) [and (c)(10)]. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, on the basis of 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. NUREG-1433, “Standard Technical Specifications General Electric Plants, BWR/4, Revision 3,” August 31, 2003
2. NUREG-1434, “Standard Technical Specifications General Electric Plants, BWR/6, Revision 3,” August 31, 2003
3. Letter TSTF-04-07 from the Technical Specifications Task Force to the NRC, TSTF-475 Revision 0, “Control Rod Notch Testing Frequency and SRM Insert Control Rod Action,” May 5, 2005, ADAMS accession number ML042520035
4. Letter TSTF-06-13 from the Technical Specifications Task Force to the NRC, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0,” dated July 3, 2006, ADAMS accession number ML0618403421
5. Letter BWROG-06036 from the BWR Owners Group to the NRC, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0,” dated November 16, 2006, Enclosure of the GE Nuclear Energy Report, “CRD Notching Surveillance Testing for Limerick Generating Station,” dated November 2006, ADAMS accession number ML0632502580
6. Letter TSTF-07-19 from the Technical Specifications Task Force to the NRC, “Response to NRC Request for Additional Information Regarding TSTF-475, Revision 0,” dated May 22, 2007, ADAMS accession number ML0714204280

THE FOLLOWING EXAMPLE OF AN APPLICATION WAS PREPARED BY THE NRC STAFF TO FACILITATE USE OF THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS (CLIIP). THE MODEL PROVIDES THE EXPECTED LEVEL OF DETAIL AND CONTENT FOR AN APPLICATION TO REVISE TECHNICAL SPECIFICATIONS REGARDING REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY, CLARIFICATION OF SRM INSERT CONTROL ROD ACTION, AND A CLARIFICATION OF A FREQUENCY EXAMPLE. LICENSEES REMAIN RESPONSIBLE FOR ENSURING THAT THEIR ACTUAL APPLICATION FULFILLS THEIR ADMINISTRATIVE REQUIREMENTS AS WELL AS NUCLEAR REGULATORY COMMISSION REGULATIONS.

U. S. Nuclear Regular Commission
Document Control Desk
Washington, D.C. 20555

SUBJECT: PLANT NAME
DOCKET NO. 50-
APPLICATION FOR TECHNICAL SPECIFICATION CHANGE REGARDING REVISION OF CONTROL ROD NOTCH SURVEILLANCE TEST FREQUENCY, CLARIFICATION OF SRM INSERT CONTROL ROD ACTION, AND A CLARIFICATION OF A FREQUENCY EXAMPLE USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

Gentleman:

In accordance with th provisions of 10 CFR 50.90 [LICENSEE] is submitting a request for an amendment to the technical specifications (TS) for [PLANT NAME, UNIT NOS.].

The proposed amendment would: (1) revise the TS surveillance requirement (SR) frequency in TS 3.1.3, “Control Rod OPERABILITY”, (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, required Action E.2, “Source Range Monitoring Instrumentation,” and (3) revise Example 1.4-3 in Section 1.4 “Frequency” to clarify the applicability of the 1.25 surveillance test interval extension.

Attachment 1 provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications. Attachment 2 provides the existing TS pages marked up to show the proposed change. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides a summary of the regulatory commitments made in this submittal.

[LICENSEE] requests approval of the proposed License Amendment by [DATE], with the amendment being implemented [BY DATE OR WITHIN X DAYS].

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated [STATE] Official.

I declare under penalty of perjury under the laws of the United Stats of America that I am authorized by [LICENSEE] to make this request and that the foregoing s true and correct. (Note that request may be notarized in lieu of using this oath or affirmation statement).

If you should have any questions regarding this submittal, please contact [NAME, TELEPHONE NUMBER]

Sincerly,

[Name, Title]

Attachments:
1. Description and Assessment
2. Proposed Technical Specification Changes
3. Revised Technical Specification Pages
4. Regulatory Commitments
5. Proposed Technical Specification Bases Changes

cc: NRC Project Manager
NRC Regional Office
NRC Resident Inspector
State Contact

ATTACHMENT 1
Description and Assessment

1.0 DESCRIPTION

The proposed amendment would: (1) revise the TS surveillance requirement (SR 3.1.3.2) frequency in TS 3.1.3, "Control Rod OPERABILITY", (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation", and (3) revise Example 1.4-3 in Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension.

The changes are consistent with Nuclear Regulatory Commission (NRC) approved Industry/Technical Specification Task Force (TSTF) STS change TSTF-475, Revision 1. The *Federal Register* notice published on [DATE] announced the availability of this TS improvement through the consolidated line item improvement process (CLIIP).

2.0 ASSESSMENT

2.1 Applicability of Published Safety Evaluation

[LICENSEE] has reviewed the safety evaluation dated [DATE] as part of the CLIIP. This review included a review of the NRC staff's evaluation, as well as the supporting information provided to support TSTF-475, Revision 1. [LICENSEE] has concluded that the justifications presented in the TSTF proposal and the safety evaluation prepared by the NRC staff are applicable to [PLANT, UNIT NOS.] and justify this amendment for the incorporation of the changes to the [PLANT] TS.

2.2 Optional Changes and Variations

[LICENSEE] is not proposing any variations or deviations from the TS changes described in the modified TSTF-475, Revision 1 and the NRC staff's model safety evaluation dated [DATE].

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

[LICENSEE] has reviewed the proposed no significant hazards consideration determination (NSHCD) published in the *Federal Register* as part of the CLIIP. [LICENSEE] has concluded that the proposed NSHCD presented in the Federal Register notice is applicable to [PLANT] and is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

3.2 Verification and Commitments

As discussed in the notice of availability published in the *Federal Register* on [DATE] for this TS improvement, the [LICENSEE] verifies the applicability of TSTF-475 to [PLANT], and commits to establishing Technical Specification Bases for TS as proposed in TSTF-475, Revision 1.

These changes are based on TSTF change traveler TSTF-475 (Revision 1) that proposes revisions to the BWR STS by: (1) revising the frequency of SR 3.1.3.2, notch testing of fully withdrawn control rod, from “7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM” to “31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM”, (2) adding the word “fully” to LCO 3.3.1.2 Required Action E.2 (NUREG- 1434 only) to clarify the requirement to fully insert all insertable control rods in core cells containing one or more fuel assemblies when the associated SRM instrument is inoperable, and (3) revising Example 1.4-3 in Section 1.4 “Frequency” to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the “SURVEILLANCE” column in addition to the time periods in the “FREQUENCY” column.

4.0 ENVIRONMENTAL EVALUATION

[LICENSEE] has reviewed the environmental evaluation included in the model safety evaluation dated [DATE] as part of the CLIIP. [LICENSEE] has concluded that the staff’s findings presented in that evaluation are applicable to [PLANT] and the evaluation is hereby incorporated by reference for this application.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION PAGES

ATTACHMENT 4

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by [LICENSEE] in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to [CONTACT NAME].

REGULATORY COMMITMENTS	DUE DATE/EVENT
[LICENSEE] will establish the Technical Specification Bases for [TS B 3.1.3, TS B 3.1.4, and TS B 3.3.1.2] as adopted with the applicable license amendment.	[Complete, implemented with amendment OR within X days of implementation of amendment]
[LICENSEE] will establish the water quality controls as recommended by SIL No. 148, Water Quality Control for the Control Rod System," September 15, 1975	[Complete, implemented with amendment OR within X days of implementation of amendment]

ATTACHMENT 5

PROPOSED CHANGES TO TECHNICAL SPECIFICATION BASES PAGES

Proposed No Significant Hazards Consideration Determination

Description of Amendment Request: [Plant Name] requests adoption of an approved change to the Standard Technical Specifications (STS) for General Electric (GE) Plants (NUREG-1433, BWR/4 and NUREG-1434, BWR/6) and plant specific technical specifications (TS), that allows: (1) revising the frequency of SR 3.1.3.2, notch testing of fully withdrawn control rod, from "7 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of RWM" to "31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM", (2) adding the word "fully" to LCO 3.3.1.2 Required Action E.2 (NUREG- 1434 only) to clarify the requirement to fully insert all insertable control rods in core cells containing one or more fuel assemblies when the associated SRM instrument is inoperable, and (3) revising Example 1.4-3 in Section 1.4 "Frequency" to clarify that the 1.25 surveillance test interval extension in SR 3.0.2 is applicable to time periods discussed in NOTES in the "SURVEILLANCE" column in addition to the time periods in the "FREQUENCY" column. The staff finds that the proposed STS changes are acceptable because the number of control rod manipulations is reduced thereby reducing the opportunity for potential reactivity events while having a very minimal impact on the extremely high reliability of the CRD system as discussed in the technical evaluation section of this safety evaluation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

Criterion 1—The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated

The proposed change generically implements TSTF-475, Revision 1, "Control Rod Notch Testing Frequency and SRM Insert Control Rod Action." TSTF-475, Revision 1 modifies

NUREG-1433 (BWR/4) and NUREG-1434 (BWR/6) STS. The changes: (1) revise TS testing frequency for surveillance requirement (SR) 3.1.3.2 in TS 3.1.3, "Control Rod OPERABILITY", (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, Required Action E.2, "Source Range Monitoring Instrumentation" (NUREG-1434 only), and (3) revise Example 1.4-3 in Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension. The consequences of an accident after adopting TSTF-475, Revision 1 are no different than the consequences of an accident prior to adoption. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Criterion 2—The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated

The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously analyzed. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3—The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety

TSTF-475, Revision 1 will: (1) revise the TS SR 3.1.3.2 frequency in TS 3.1.3, "Control Rod OPERABILITY", (2) clarify the requirement to fully insert all insertable control rods for the limiting condition for operation (LCO) in TS 3.3.1.2, "Source Range Monitoring Instrumentation" (NUREG-1434 only), and (3) revise Example 1.4-3 in Section 1.4 "Frequency" to clarify the applicability of the 1.25 surveillance test interval extension. The GE Nuclear Energy Report, “CRD Notching Surveillance Testing for Limerick Generating Station,” dated November 2006, concludes that extending the control rod notch test interval from weekly to monthly is not expected to impact the reliability of the scram system and that the analysis supports the decision to change the surveillance frequency. Therefore, the proposed changes in TSTF-475, Revision 1 are acceptable and do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Dated at Rockville, Maryland, this 9th day of August, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Carl Schulten, Acting Chief
Technical Specifications Branch
Division of Inspection & Regional Support
Office of Nuclear Reactor Regulation

"CRD Notching Surveillance Testing for Limerick Generating Station," dated November 2006, concludes that extending the control rod notch test interval from weekly to monthly is not expected to impact the reliability of the scram system and that the analysis supports the decision to change the surveillance frequency. Therefore, the proposed changes in TSTF-475, Revision 1 are acceptable and do not involve a significant reduction in a margin of safety.

Based upon the reasoning presented above and the previous discussion of the amendment request, the requested change does not involve a significant hazards consideration.

Dated at Rockville, Maryland, this 9th day of August, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Carl Schulten, Acting Chief
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Office of Nuclear Reactor Regulation

Accession Number: ML072040123 **ADAMS Template:**

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DATE	07/23/2007	07/24/2007	07/26/2007	08/8/2007	08/01/2007

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