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W3F1-2016-0010

March 3, 2016

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Responses to Request for Additional Information Regarding the Risk-Informed Surveillance Requirements License Amendment Request (LAR)
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy letter W3F1-2015-0006 "Application for Technical Specification Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program", Waterford Steam Electric Station, Unit 3 dated June 17, 2015 [ML15170A121]

2. NRC letter to Entergy dated January 22, 2016 "Request for Additional Information Regarding The Risk-Informed Surveillance Requirements: License Amendment Request" [ML16015A294]

Dear Sir or Madam:

By letter dated June 17, 2015 Entergy Operations, Inc. (Entergy), submitted a license amendment request (LAR) to adopt U.S. NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specifications Change traveler TSTF-425 Initiative 5b (Reference 1).

In letter dated January 22, 2016 (Reference 2), the NRC staff made a Request for Additional Information (RAI) needed to complete its review with responses due in 45 calendar days. Attachment 1 provides the responses to those questions. Attachment 2 contains revision markups of selected pages of the proposed revisions to the Technical Specifications impacted by the RAI responses. Attachment 3 presents updated clean versions of the proposed Technical Specifications affected by the RAI responses.

There are no new regulatory commitments contained in this submittal. If you require additional information, please contact the Regulatory Assurance Manager, John Jarrell at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 3, 2016.

Sincerely,

MRC/AJH

Attachments:
1. RAI Responses
2. Proposed Technical Specification Changes (Revised Markup)
3. Proposed Technical Specification Changes (Revised Clean)

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ATTACHMENT 1

W3F1-2016-0010

RAI Responses

RAI Responses

RAI-1

Several "gaps" identified in Table 3-1, "Status of Identified Gaps to Capability Category II of the ASME [American Society of Mechanical Engineers] PRA [Probabilistic Risk Assessment] Standard," of Attachment 2, "Documentation of PRA Technical Adequacy," of the LAR by letter dated June 17, 2015, are dispositioned as having "little/no impact on STI [surveillance test interval] evaluations" without explaining how the disposition was determined. As resolution of these "gaps" may be necessary inputs to determine the technical adequacy of the internal flooding probabilistic risk assessment (IFPRA) model, address the following:

A. Gap #1 states, in part:

> Although required by this SR [surveillance requirement], no evaluation of individual component failure modes, human-induced mechanisms, or other events that could release water into the area were identified.

The licensee's evaluation of this "gap" is that it has little or no impact on STI evaluations because the assumption of treating all flooding sources as a guillotine break is conservative. Treating all identified flooding sources conservatively does not address the gap issue and the LAR submittal does not appear to have sufficient information to indicate that all flooding sources have been identified and evaluated (e.g., human-induced over-filling of tanks or diverting flow through openings, inadvertent actuation of the fire-suppression system, etc.). Given that excluded flooding sources listed in Gap #1 contribute to underestimation of the internal flooding risk and could exclude risk contributors that impact the STI calculations, justify why the PRA is of sufficient quality for the STI application.

B. Gap #2 indicates that there are assumptions made in the IFPRA for which engineering calculations are not available. The licensee's evaluation of this "gap" is that these unsupported assumptions have little or no impact on the IFPRA. Given that unsupported assumptions should be evaluated to determine whether they contribute to underestimation of internal flooding risk, justify why the PRA is of sufficient quality for the STI application.

C. Gaps #3 and #6 state that some locations have been excluded from the modeling (i.e., the Fire Water pump house and condensate polisher building). Given that excluding these locations could contribute to underestimation of the internal flooding risk and could result in exclusion of risk contributors that impact STI calculations, justify why the PRA is of sufficient quality for the STI application.

D. Gap #4 states that a reduction factor was used inappropriately to convert the rupture flow rates to spray flow rates. Given that the use of the reduction factor could contribute to underestimation of the internal flooding risk and impact on STI calculations, justify why the PRA is of sufficient quality for the STI application.

E. Gap #8 states that numerical uncertainties were not propagated as part of IFPRA quantification. In accordance with the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, estimates of the mean core damage frequency (CDF) should account for state of knowledge correlations (SOKC). Given that not incorporating the impact of SOKC could contribute to underestimation of the internal flooding risk and impact STI calculations, justify why the PRA is of sufficient quality for the STI application.

RAI-1 Response

A. Gap 1 from the PRA Adequacy section of the LAR is a Finding from a 2009 peer review. The review issued a Finding on Standard Supporting Requirement IF-B2. This finding is not an accurate reflection of the Waterford Internal Flood PRA analysis and is a result of an oversight by the peer review team. It remains a gap for the Waterford 3 PRA as it is an unresolved peer review finding, but it is a documentation issue , not a technical adequacy issue that would impact STI evaluations.

The Waterford 3 Internal Flooding Analysis (documented in PRA-W3-01-002) does consider non-pipe failure flooding sources. The following two statements are from the flooding document (underlined text to be noted):

Pipe breaks, tank and heat exchanger failures, valve and pump failures, and maintenance induced floods are potential flood sources. Valve and pump failures include ruptures of glands, seals, and gaskets. In this analysis, all causes of flooding were considered (Section 2 of PRA-W3-01-002).

The sizes of flood sources were first determined assuming guillotine breaks of lines or the catastrophic rupture of valves, tanks, gaskets, fittings and heat exchangers. (Section 3.1.2 'Flood Sources' PRA-W3-01-002)

Another example of a non-pipe flooding source consideration is from Section 4.2.1.4 of PRA-W3-01-002. This section evaluates a potential flooding scenario involving the Turbine Cooling Water system and considers failures of both the surge tank and valve TC-112 (TC MVAAA112 - TCW Surge Tank Outlet Isolation). Other examples of non-pipe flood source considerations are in the document.

The Waterford 3 flooding analysis does not evaluate human action/maintenance induced flood events "(e.g., human-induced over-filling of tanks or diverting flow through openings, inadvertent actuation of the fire-suppression system, etc.)". This is clearly documented in the analysis. The basis for the exclusion is the human element would allow for immediate response and heightened site awareness to the situation driving down the impact of such events. Excluding these events was not an oversight but an intentional technical decision (documented in the analysis).

This gap is the result of some imprecise wording in the flood analysis but does not represent a technical shortcoming. Relevant flood sources are considered and there is no underestimation of flood frequencies.

B. Gap 2 from the PRA Adequacy section of the LAR is a Finding from a 2009 peer review. The associated Finding was the result of technical references (including some plant specific calculations) used in the flooding analysis that were not available for the peer review team to complete the evaluation/review.

The supporting documents calculate many parameters performed for each pipe break scenario and were completed by the flood PRA vendor/contractor. Initially, the documentation was not included with the report and the peer review team did not have the supporting documents during the review. The reference material has since been obtained from the vendor and is no longer 'unavailable'. This is a documentation issue and no technical gap exists that would impact STI evaluations. The details and specific items listed in the finding are included in the obtained report.

C. Gaps 3 and 6 (from the PRA Adequacy section of the LAR are Findings from a 2009 peer review) are associated with how the Fire Pump House (3) and Condensate Polisher Building (6) are treated in the internal flooding analysis. These same issues were evaluated and dispositioned in RAIs associated with the Integrated Leak Rate Test submittal. The Findings corresponding to these gaps and the provided dispositions are as follows (From Attachment 2 to W3F1-2015-0021, [ADAMS Accession No. ML15124A946]):

Gap 3:

The Fire Water pump house has been excluded from evaluation on the basis that the failure of the fire pumps will not precipitate a reactor trip. This exclusion needs to be re-visited to determine if an internal flood in the fire water pump house has the potential to initiate a flood/spray event elsewhere in the plant due to spurious fire water valve actuations (e.g. look at potential for spray/submergence on a fire water control panel.), and if this inadvertent actuation could result in the need for a plant shutdown.

Disposition for ILRT RAI:

The Fire Pump control panel cannot affect suppression system actuation inside the plant. The Fire Protection Main Control Panel controls the operation of suppression within the Reactor Auxiliary building and Turbine building. Any malfunction in the Fire Pump House would not affect the main control panel and therefore could not cause a release to damage any risk significant equipment or cause a plant trip. This Finding has no impact on the quantified results of the IFPRA and therefore no impact on the ILRT application.

Gap 6:

The discussion for excluding the condensate polisher building from consideration based on the assumption that the operators would bypass the condensate polisher system in the event of a rupture/leak within the building is inadequate.
Disposition for ILRT RAI:

The worst case scenario from a flood in the condensate polisher building would be a loss of main feedwater (with a plant trip) and a loss of both 480V switchgears in the building. The Fire PRA developed a scenario with these impacts which had a CCDP of 4.68E-5 (PRA-W3-05-007) which bounds the potential effects of floods in this building. While specific flood scenarios should probably be developed for this building, it is evident that the contribution to CDF would be minor as the flood frequency still *needs to be considered as well. Therefore, addressing this finding has a negligible impact this application.*

In both cases, the ILRT related dispositions are valid for STI evaluations. Fire pump house flooding can only impact the local pumps and is correctly modeled (this is a documentation issue). Flood scenarios for the condensate polisher building can impact the plant, however, the fire PRA analysis shows the bounding risk impact of a flood in the building and it will have insignificant contribution to risk results.

D. Gap 4 is associated with application of reduction factors for flood events (initiating event frequencies). The gap is the result of imprecise wording in the flood analysis along with semi-redundant terminology (i.e. rupture events & flood events) in the analysis and in the source/methodology documentation.

The Waterford 3 Flood PRA analysis considers releases up to 100 gpm as being “spray” events to be consistent with the EPRI pipe break frequencies report. The Waterford 3 Flood PRA also treats releases from 100 gpm to 2000 gpm as “flood” events. The EGG-SSRE-9639 report considers “rupture” events as any leakage greater than 50 gpm. Note that there is overlap between ‘rupture’ and both ‘flood’ and ‘spray’. A small rupture (between 50 gpm and 100 gpm) would be a ‘spray’ but not a ‘flood’ per the EPRI pipe break report. The following statement is from the EGG-SSRE-9639 report.

“It should be kept in mind that the external rupture events include any leakage greater than 50 gpm. Therefore, most of the external ruptures identified…do not involve complete pipe severance or catastrophic failure of a valve or pump body. **The frequencies for such catastrophic rupture events should be lower than those presented in this report.”** The bold text is not bold in the report but is shown here to provide evidence the source of the flood data itself indicated that a reduction factor is warranted for more severe flooding incidents.

To adjust for the larger sized rupture events for Waterford 3 flooding scenarios involving one of the components from the EGG-SSRE-9639 report, the exponential correlation from Prugh (Prugh, R. W., "Leak Size/Frequency Distribution," Presented at 1992 Process Plant Safety Symposium, A.I.Ch.E, February 19, 1992) is used to relate it to the actual release size. The reduction factors applied are both appropriate and do not contribute to underestimation of flooding risk.

E. Gap 8 is associated with the inclusion of State of Knowledge Correlation and propagation of numerical uncertainties in the internal flooding analysis. The increase in probabilities associated with internal flooding due to SOKC would be minor. Though no formal uncertainty analysis has been performed on the internal flooding model/results, the flooding analysis is an application of the internal events model which had a thorough uncertainty analysis performed on it. The only unique unanalyzed uncertainty associated with this finding would be due to the initiating event frequencies in the IFPRA, that is, the pipe break frequencies. The associated error factors presented in the pipe break frequency basis document (EPRI TR-1013141) have more independence (as a data source) than the other inputs in the internal events model and are therefore less subject to the effects of correlation.

This excerpt is part of the definition of State of Knowledge Correlation from NUREG-1855, Revision 1 (Appendix 6-A).

> *In the ideal situation, each plant initiating event, structure, system, or component or operator action modeled as a basic event in the PRA would have its own database associated with it and thus would be statistically independent (i.e., their parameter values would be based on independent data that is not pooled or correlated in any way). If this were the case, the propagation of the basic event mean values in the analysis would lead to point estimates of the risk metrics that would themselves be true mean values. However, in general, this ideal situation is not realized in practice, and the data used for like components within a cut set of the analysis often has some common element, is pooled, or is correlated in some way. For example, the generic knowledge of the failure rate of one particular pump (such as a low pressure coolant injection pump) for a given failure mode is typically based on experience with all "similar" pumps. Therefore, the various basic events that involve this failure mode of a pump are all in fact being estimated from a single state-of-knowledge distribution, and the data used for the pumps is not independent but is correlated. If this correlation, i.e., the SOKC, is not accounted for, the point estimate of a cut set containing two or more basic events involving failures of these pumps will differ from the true mean value*.

The flood frequencies (pipe break frequencies) are from a data source with its own database and are statistically independent of component failure rates and initiating event frequencies in the PRA model. This fact significantly limits the potential impact of SOKC on the flood results. The SOKC has been evaluated for the internal events PRA model and results. The impact of uncertainties associated with the flooding model and results are judged to be minor due to the independence of the flooding data. Any increase in probabilities associated with internal flooding due to SOKC would be minor and this gap has a negligible impact on STI evaluations using the Waterford 3 PRA model.

RAI-2

The licensee utilizes the Waterford PRA model, which was peer reviewed in an Internal Events PRA peer review from August of 2009. The Facts and Observations (F&O) from this review, along with their dispositions from this review, are summarized in Attachment U of the LAR to adopt National Fire Protection Association (NFPA) Standard 805, dated November 11, 2011 (ADAMS Accession No. ML113220230). F&O IE-C6 01, for Event Fault Tree Modeling states:

> The initiating event fault tree [IE FT] modeling for these systems considers multiple failures, CCF [common-cause failure] events and routine system alignments. These IE FTs exclude many failures that are included in the systems analysis (failures of valves, breakers, etc. in redundant paths to transfer open or transfer closed; component failure rates less than 1% of the pump active failures such as sensors and transmitters; and flow diversion paths).

The disposition presented for this F&O states that these failures are omitted from the initiating event fault trees because they are passive failures that affect one path only. However, this disposition appears to apply only in the context of fire protection. Although excluding certain failures from the initiating events fault tree models does not impact fire risk, they could contribute to underestimation of internal events risk and impact STI calculations.

Address each of the excluded failures cited in F&O IE-C6-01 in the context of the current license amendment request. If this F&O is not completely resolved, explain why the incomplete modeling does not impact the STI calculations.

RAI-2 Response

This RAI has been drafted based on the content of NFPA 805 LAR Attachment U (ADAMS Accession No. ML113220230). The response is based on the content of a more recent submittal of Attachment U. The Peer Review Finding (F&O IE-C6-01) was explicitly addressed during a PRA model update and a disposition was provided in Table U-1 (Attachment U – Internal Events PRA Quality) of the Waterford 3 NFPA 805 LAR Supplement documented in W3F1-2013-0048. [ADAMS Ascension # ML13365A325]

That table states:
The current IE fault tree logic is more thorough than past models. The current model is documented in PSA-WF3-01-IE. The current IE fault trees include items in redundant paths (including valves and breakers).

RAI-3

Section 3.3, "External Events Considerations," in Attachment 2 of the LAR submittal, by letter dated June 17, 2015, states, in part, that WF3 has a state-of-the-art Fire PRA model and:

> Any STI related parameter changes evaluated by the internal events model can also be evaluated using the Fire model.

The submittal further states, in part, that:

> The Fire PRA model will be exercised to obtain quantitative fire risk insights when a qualitative or a bounding analysis is not deemed sufficient…

The use of a qualitative or bounding analysis when a system, structure, or component is explicitly modeled and evaluated in the fire PRA is inconsistent with Section 4, "Surveillance Frequency Control Program Change Process," Step 10, "Perform Qualitative or Bounding Risk Analysis," of the NEI 04-10, Revision 1, methodology. Explain how and when the fire PRA will be used in performing the STI calculations to ensure consistency with the NEI 04-10 methodology.

RAI-3 Response

Waterford will consistently follow NEI 04-10 guidance when performing STI calculations. Application of the Fire PRA model will follow steps in section 4.0 of NEI 04-10 (SURVEILLANCE FREQUENCY CONTROL PROGRAM CHANGE PROCESS) in evaluating all STI changes.

Most changes are expected to be straight forward and similar to the process for the internal events PRA. If the structure, system, component (SSC) is in the model and meets the criteria of Step 8, then Steps 12 (and 12A-1) will be completed to evaluate the total and cumulative effects on CDF and LERF (and delta CDF/delta LERF).

It is likely that there will be cases were application associated with the fire PRA is not as clear and will differ. From NEI-04-10 Section 4.0 Step 10:

If in Step 8, the SSC is determined to be explicitly modeled and evaluated in the fire PRA, then the fire PRA evaluation process is used to determine the fire risk metric inputs associated with the STI change as depicted in Step 12. However, if it is determined that the SSC is only implicitly modeled, then there is a choice of performing either a bounding analysis as described in Step 10b or a detailed analysis as described in Step 11.

If the SSC is determined to be only implicitly modeled in the FIVE methodology process, then there is a choice of performing either a bounding analysis as described in Step 10b or a detailed analysis as described in Step 11. Because FIVE is a conservative screening analysis, care should be exercised in adding the risk increase values from FIVE evaluation to the total increase from all other PRA results.

If the SSC is not evaluated in either the fire PRA or FIVE evaluations, (either explicitly or implicitly, and it is judged to have no impact on the PRA results), then the SSC can be qualitatively screened with the information summarized in Step 15 for presentation to the IDP. This initial screening is from the standpoint of fire events as not having an impact on the CDF and LERF metrics. The evaluation is continued with seismic risk.

Cases that are not clearly applicable to Step 12 treatment will be treated on a case by case basis and some qualitative or bounding assessments may be necessary (though few cases of these are expected). In all cases, NEI 04-10 guidance will be used to ensure compliance with accepted methods is maintained.

RAI-4

The LAR submittal, does not discuss how the surveillance frequency control program will assess the risk of shutdown events when evaluating STIs. Guidance on the assessment of shutdown events is provided in NEI 04-10 Section 4, Step 10. In addition, shutdown risk impact is listed as a line item in Section C of the Sample Surveillance Test Frequency Evaluation form, which references TSTF-425. Describe how the risk of shutdown events will be assessed as part of the WF3 Surveillance Frequency Control Program (SFCP) and whether and how a shutdown PRA model will be used in these assessments.

RAI-4 Response

Waterford 3 has a shutdown PRA model. The model is based on EPRI shutdown PRA guidance and has shutdown event trees as well as fault tree logic with shutdown specific system configuration (Shutdown Cooling for example). The shutdown risk model will be used in accordance with the process outlined in NEI 04-10 Section 4.0 to determine the risk metric changes associated with the STI evaluation.

If the Structures, Systems, and Components (SSC) are not involved in the shutdown PRA, then the SSC can be screened qualitatively with the information presented to the Independent Decision making Panel (IDP). This initial screening is from the standpoint of shutdown risk as not having an impact on the CDF and LERF metrics.

RAI-5

Section 3.3 in Attachment 2 of the LAR submittal, states that a qualitative or bounding approach will be utilized in most cases for STI change evaluations. The submittal does not explain how the external hazards evaluated in the WF3 Individual Plant Examination of External Events are updated to reflect new information when used in performing a qualitative or bounding analysis. Discuss the process for incorporating new information in these qualitative or bounding analyses and how this process adequately supports implementation of the SFCP.

RAI-5 Response

Entergy fleet procedure EN-DC-151 ("PSA Maintenance and Update" revision 5, November 2013.) contains requirements to maintain and update the site PRA models. The stated purpose of the procedure is to establish a process for maintaining active Entergy PSA internal and external events Level 1 and LERF models, current with the as-built and as-operated plants and to ensure that current industry standards, experience, and technology are incorporated appropriately into the models.

The procedure has requirements (EN-DC-151 section 5.5) to help ensure External Event improvements are considered during periodic model updates. During these model updates, "[o]ther separate models such as; seismic, fire, internal flooding, winds and tornadoes etc., should be considered for updating as well." Also, "[d]uring each periodic PSA model update, recent industry experience, new PRA technology, new research, and changes to industry standards should be reviewed for possible impact on the PSA models. The checklist in Attachment 9.2 should be used to assist in this review. If a potential PSA model change is identified, a Model Change Request (MCR) shall be initiated. The MCR should be graded … at that time and the MCR evaluated for inclusion in the scope of the Periodic PSA model update."

The Waterford 3 PRA follows the listed procedure and routinely evaluates new information and technology and considers the impact on the external events models. The current external event models are judged to bound the sites risk from external events. The available analyses will be used to help inform/support the qualitative evaluations developed to support the STI changes. New relevant information will be considered on a case by case basis. As with all other aspects of the STI change process, NEI 04-10 methodology will be followed for external events considerations.

RAI-6

In Surveillance Requirement (SR) 4.1.3.6 of Attachment 3, "Proposed Technical Specification Changes," of the LAR submittal, WF3 proposes to relocate the fixed periodic surveillance frequency into the surveillance frequency control program:

1. for when each regulating control element assembly (CEA) group and CEA group P are within the Transient Insertion Limits, and

2. for determination of the accumulated times during which regulating CEA groups or CEA group P are inserted beyond the long term steady state insertion limits but within the transient insertion limits.

However, in SR 4.1.3.6 of Attachment 4, “Revised Technical Specification Pages,” on page 3/4 1-26 of the LAR submittal, WF3 has replaced the surveillance frequencies for the above, and for verification of individual CEA positions during time intervals when the power dependent insertion limit (PDIL) auctioneer alarm circuit is inoperable. This part of SR 4.1.3.6 appears to be part of a surveillance related to a specific condition (“…when the PDIL Auctioneer Alarm Circuit is inoperable …”) which should be excluded in accordance with Section 2.0, “Proposed Changes,” of TSTF-425, Revision 3.

If WF3 intends to relocate the surveillance frequency for verification of individual CEA positions during time intervals when the PDIL auctioneer alarm circuit is inoperable then:

1. explain why this does not meet the exclusion, and

2. submit a revised SR 4.1.3.6 mark-up, page 3/4 1-26 of Attachment 3 to reflect the relocation.

RAI-6 Response

The surveillance frequency associated with the verification of individual CEA positions during time intervals when the PDIL auctioneer alarm circuit is inoperable meets the criteria for exclusion. The proposed markup of Technical Specification SR 4.1.3.6 in Attachment 3, “Proposed Technical Specification Changes,” of the Waterford 3 submittal is correct. A revised SR 4.1.3.6 is included in Attachment 3 of this RAI response letter and matches the corresponding markup in Attachment 3 of the LAR submittal (Entergy letter W3F1-2015-0006).

RAI-7:

The licensee proposes to relocate the surveillance frequency for SR 4.3.2.3, which is similar to SR 4.3.1.3 in that both SRs test response times and both include an N term where N is the total number of redundant channels in a specific function. However, in the LAR submittal, Attachment 3, for SR 4.3.1.3, WF3 includes removal of the language defining the N term (i.e., “… at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function …”). The Attachment 3 mark-ups of the LAR submittal for SR 4.3.2.3 do NOT include removal of the language defining the N term. The Attachment 4, of SR 4.3.2.3 on page 3/4 3-1 does include removal of the N term language.

If WF3 intends to remove the language defining the N term in SR 4.3.2.3, submit a revised SR 4.3.2.3 mark-up, page 3/4 3-1 of Attachment 3, showing this change.

RAI-7 Response

The language defining the N term should have been removed from SR 4.3.2.3 located in Attachment 3, “Proposed Technical Specification Changes,” (mark-ups) of the LAR submittal. The SR 4.3.2.3 provided in Attachment 4 (clean copy) of the LAR submittal is correct. The marked-up version of SR 4.3.2.3 has been revised to remove the language defining the N term; thereby, matching the SR 4.3.2.3 clean version of Attachment 4 of the LAR submittal (Entergy letter W3F1-2015-0006). The corrected markup of SR 4.3.2.3 is included in Attachment 2 of this RAI response letter.

RAI-8

The NRC staff cannot find any justification in the LAR application for the removal of the "… within the previous 62 days" language from the end of table note 3 of Table 4.3-2 on page 3/4 3-27 of Attachment 3. This part of the table note appears to be part of a surveillance related to a specific condition, which should be excluded per Section 2.0 of TSTF-425, Revision 3. Provide adequate justification for the removal of this language from the note by explaining why the words immediately before (i.e., "… during each COLD SHUTDOWN condition …") do not make this an excluded frequency related to a specific condition (i.e., "COLD SHUTDOWN")?

Submit a revised markup of page 3/4 3-27 of Attachment 3 of the LAR application if this part of the table notation does indicate a surveillance related to a specific condition without striking the "… within the previous 62 days."

RAI-8 Response

Note 3 of Table 4.3-2 notation was erroneously marked up to delete "… within the previous 62 days." This portion of Note 3 meets the exclusion criteria specified by TSTF-425, Revision 3. Corrected mark-up and clean versions of the TS are provided in Attachments 2 and 3 of this RAI response letter respectively.

RAI-9

WF3 proposes to relocate the surveillance frequency for SR 4.7.6.3, Part b according to page 3/4 7-18a of Attachment 3 of the LAR submittal. How can this be done while allowing the language "… if not performed within the last quarter …" to remain in the SR?

Can the surveillance frequency be potentially changed (from quarterly) in the future in the licensee-controlled document per the accepted guidance in NEI 04-10, Revision 1 with the language identified above remaining in the SR?

Either submit a markup removing the language mentioned above with adequate justification for removal or explain how the language can remain as marked in the June 17, 2015, submittal.

RAI-9 Response

SR 4.7.6.3.b was erroneously marked up to delete "At least quarterly, if not performed within the last quarter, by verifying that each…" This portion of the SR meets the exclusion criteria specified by TSTF-425, Revision 3. A corrected mark-up and clean version of the TS that returns this exclusionary portion of SR 4.7.6.3.b to its original state has been added to Attachments 2 and 3 of this RAI response letter respectively.

ATTACHMENT 2

W3F1-2016-0010

Proposed Technical Specification Changes (Revised Markup)

Page 3/4 3-13
Page 3/4 3-27
Page 3/4 7-18a

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

1

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE ~~at least once per 18 months~~ during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

Add INSERT 1a

4.3.2.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit ~~at least once per 18 months~~. Each test shall include at least one channel per function such that all channels are tested ~~at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function~~ as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	~~R~~	1, 2, 3
b. SG Level (1/2)-Low and ΔP (1/2) - High	~~S~~	~~R~~	~~Q~~	1, 2, 3
c. SG Level (1/2) - Low and No Pressure - Low Trip (1/2)	~~S~~	~~R~~	~~Q~~	1, 2, 3
d. Automatic Actuation Logic (except subgroup relays)	N.A.	N.A.	~~Q~~(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A.	~~M~~(1) (3)	1, 2, 3
e. Control Valve Logic (Wide Range SG Level - Low)	~~S~~	~~R~~	~~SA~~(5)	1, 2, 3

Replace each marked through surveillance frequency with 'SFCP'.

Add INSERT 1a

TABLE NOTATION

(1) Each train or logic channel shall be tested ~~at least every 62 days on a STAGGERED TEST BASIS~~.

(2) Testing of Automatic Actuation Logic shall include energization/deenergization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays K109, K114, K202, K301, K305, K308 and K313 are exempt from testing during power operation but shall be tested ~~at least once per 18 months~~ and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

Add INSERT 1a

(4) Using installed test switches.

(5) To be performed during each COLD SHUTDOWN if not performed in the previous 6 months.

(6) Each train shall be tested, with the exemption of relays, K110, K410 and K412, ~~at least every 62 days on a STAGGERED TEST BASIS.~~ Relays K110, K410 and K412 shall be tested ~~at least every 62 days but will be exempt from the STAGGERED TEST BASIS.~~

PLANT SYSTEMS

CONTROL ROOM AIR TEMPERATURE - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.6.3 Two independent control room air conditioning units shall be OPERABLE.

APPLICABILITY*: MODES 1, 2, 3, and 4.

ACTION:

a. With one control room air conditioning unit inoperable, restore the inoperable unit to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two control room air conditioning units inoperable, return one unit to an OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each control room air conditioning unit shall be demonstrated OPERABLE:

a. ~~At least once per 12 hours~~ by verifying that the operating control room air conditioning unit is maintaining average control room air temperature less than or equal to 80°F.

Add INSERT 1b

b. At least quarterly, if not performed within the last quarter, by verifying that each control room air conditioning unit starts and operates for at least 15 minutes.

*During load movements with or over irradiated fuel assemblies, TS 3.7.6.4 is also applicable.

ATTACHMENT 3

W3F1-2016-0010

Proposed Technical Specification Changes (Revised Clean)

Page 3/4 1-26
Page 3/4 3-27
Page 3/4 7-18a

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With the regulating CEA groups or group P CEAs inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 5 EFPD per 30 EFPD interval or greater than 14 EFPD per calendar year, either:

   1. Restore the regulating CEA groups or group P CEAs to within the Long Term Steady State Insertion Limits within two hours, or

   2. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group and CEA group P shall be determined to be within the Transient Insertion Limits in accordance with the Surveillance Frequency Control Program except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups or CEA group P are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined in accordance with the Surveillance Frequency Control Program.

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	SFCP	1, 2, 3
b. SG Level (1/2) - Low and _P (1/2) - High	SFCP	SFCP	SFCP	1, 2, 3
c. SG Level (1/2) - Low and No Pressure - Low Trip (1/2)	SFCP	SFCP	SFCP	1, 2, 3
d. Automatic Actuation Logic (Except subgroup relays)	N.A.	N.A.	SFCP(2)	1, 2, 3
Actuation Subgroup Relays	N.A.	N.A	SFCP(1) (3)	1, 2, 3
e. Control Valve Logic (Wide Range SG Level - Low)	SFCP	SFCP	SFCP(5)	1, 2, 3

TABLE NOTATION

(1) Each train or logic channel shall be tested in accordance with the Surveillance Frequency Control Program.

(2) Testing of Automatic Actuation Logic shall include the energization/deenergization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays K109, K114, K202, K301, K305, K308 and K313 are exempt from testing during power operation but shall be tested in accordance with the Surveillance Frequency Control Program and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

(4) Using installed test switches.

(5) To be performed during each COLD SHUTDOWN if not performed in the previous 6 months.

(6) Each train shall be tested, with the exemption of relays, K110, K410 and K412, in accordance with the Surveillance Frequency Control Program. Relays K110, K410 and K412 shall be tested in accordance with the Surveillance Frequency Control Program.

PLANT SYSTEMS

CONTROL ROOM AIR TEMPERATURE - OPERATING

LIMITING CONDITION FOR OPERATION

3.7.6.3 Two independent control room air conditioning units shall be OPERABLE.

APPLICABILITY*: MODES 1, 2, 3, and 4.

ACTION:

a. With one control room air conditioning unit inoperable, restore the inoperable unit to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two control room air conditioning units inoperable, return one unit to an OPERABLE status within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.3 Each control room air conditioning unit shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that the operating control room air conditioning unit is maintaining average control room air temperature less than or equal to 80°F.

b. At least quarterly, if not performed within the last quarter, by verifying that each control room air conditioning unit starts and operates for at least 15 minutes.

*During load movements with or over irradiated fuel assemblies, TS 3.7.6.4 is also applicable.