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10 CFR 50.90

November 13, 2018

Serial: RA-18-0194

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

H.B. Robinson Steam Electric Plant, Unit 2
Renewed Facility Operating License No. DPR-23
Docket No. 50-261

Subject: Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt TSTF-425, Revision 3

References:

1. Duke Energy letter, *Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program (Adoption of TSTF-425, Revision 3)*, dated April 16, 2018 (ADAMS Accession No. ML18117A006).

2. Duke Energy letter, *Corrected Technical Specification Pages for TSTF-425 License Amendment Request*, dated September 25, 2018 (ADAMS Accession No. ML18269A009).

3. NRC E-Mail, *Robinson RAIs – LAR to Allow Implementation of the Provisions 10 CFR 50.69 (EPID L 2018-LLA-0095) and LAR to Adopt TSTF-425 (EPID L 2018-LLA-0104)*, dated October 12, 2018 (ADAMS Accession No. ML18288A019).

Ladies and Gentlemen:

By letter dated April 16, 2018 (Reference 1), as supplemented by letter dated September 25, 2018 (Reference 2), Duke Energy Progress, LLC (Duke Energy) submitted a license amendment request (LAR) for H.B. Robinson Steam Electric Plant, Unit 2 (HBRSEP2). The proposed amendment would modify HBRSEP2's Technical Specifications (TS) by relocating specific Surveillance Frequencies to a licensee-controlled program. The change would also add the Surveillance Frequency Control Program to TS Section 5, Administrative Controls.

By correspondence dated October 12, 2018 (Reference 3), the Nuclear Regulatory Commission (NRC) staff requested additional information from Duke Energy that is needed to complete the LAR review.

The enclosure to this letter provides Duke Energy's response to the NRC RAI regarding the HBRSEP2 application to adopt TSTF-425, Revision 3. The Attachment contains PRA implementation items which must be completed prior to implementation of the Surveillance Frequency Control Program at HBRSEP2.

The conclusions of the original No Significant Hazards Consideration and Environmental Consideration in the original LAR are unaffected by this RAI response.

There are no regulatory commitments contained in this letter.

In accordance with 10 CFR 50.91, Duke Energy is notifying the State of South Carolina of this LAR by transmitting a copy of this letter and enclosure to the designated State Official.

Should you have any questions concerning this letter and its enclosure, or require additional information, please contact Art Zaremba at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 13, 2018.

Sincerely,

Joseph W Donahue

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JLV

Enclosure: Response to NRC Request for Additional Information cc:

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Serial: RA-18-0194

H.B. Robinson Steam Electric Plant, Unit No. 2
Docket No. 50-261 / Renewed License No. DPR-23

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt TSTF-425, Revision 3

Enclosure

Response to NRC Request for Additional Information

NRC Request for Additional Information

By letter dated April 16, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18117A006), as supplemented by letter dated September 25, 2018 (ADAMS Accession No. ML18269A009), Duke Energy Progress, LLC, (Duke Energy, the licensee), submitted a license amendment request (LAR) for H.B. Robinson Steam Electric Plant, Unit 2 (HBRSEP2) to adopt Technical Specifications Task Force (TSTF) 425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control – RITSTF Initiative 5b." The proposed amendment would modify HBRSEP2's Technical Specifications (TS) by relocating specific Surveillance Frequencies to a licensee-controlled program. The change also would add the Surveillance Frequency Control Program to TS Section 5, Administrative Controls.

Regulatory Guide (RG) 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014) provides guidelines regarding the technical adequacy of PRA models used in risk informed applications. RG 1.200 describes a peer review process utilizing American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA standard RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to RA-S-2008," as one acceptable approach for determining the technical adequacy of the PRA once acceptable consensus approaches or models have been established for evaluations that could influence the regulatory decision.

To complete its review, the U.S. Nuclear Regulatory Commission (NRC staff) requests additional information (RAI) to the items below. The NRC staff notes that by letter dated April 5, 2018 (ADAMS Accession No. ML18099A130), Duke Energy submitted a LAR for HBRSEP2 to implement the provisions of Title 10 of the Code of Federal Regulations (10 CFR), Section 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Plants." RAIs regarding the HBRSEP2 LAR to implement the provisions of 10 CFR 50.69 are in Enclosure 1. The RAIs for this LAR, to adopt TSTF-425, are similar in nature to the RAIs for HBRSEP2 to implement 10 CFR 50.69. As such, the NRC staff requests separate responses to the RAIs for the HBRSEP2 LAR to implement TSTF-425 LAR and the HBRSEP2 LAR to implement 50.69, even though the responses may be similar.

PRA RAI 01 - Open/Partially Open Findings in the Process of Being Resolved:

Section 4.2 of RG 1.200 states that the LAR should include a discussion of the resolution of the peer review facts and observations (F&Os) that are applicable to the parts of the PRA required for the application. This discussion should take the following forms:

- A discussion of how the PRA model has been changed and
- A justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted (remained the same) by the particular issue.

LAR Attachment 2, Tables 1, 2, and 3, provide finding-level F&Os that are still open or only partially resolved after the F&O closure review. Also, F&O descriptions and their dispositions were previously provided to the NRC in the LAR to adopt for Technical Specification 5.5.16 Option B, "10 CFR 50, Appendix J, Integrated Leak Rate Test Interval and Type C Leak rate testing Frequency" (ADAMS Accession No. ML16201A195) and in the LAR to adopt National Fire Protection Association Standard 805 (ADAMS Accession No. ML16337A264). For a number of F&O dispositions there is insufficient information for NRC staff to conclude that the F&O is sufficiently dispositioned for this application.

a. <u>F&Os associated with Supporting Requirements (SR) AS-A5, AS-B3, LE-C4, and LE-D5</u> regarding thermally induced steam generator tube ruptures (SGTR):

   Open F&Os on SRs AS-A5, AS-B3, and LE-D5 in the LAR state, in part, that the thermally induced SGTR accident sequence was missing from the PRA. Separately, the LAR supplement indicates that the F&O associated with SR LE-D6 (SR LE-D6 directs that a thermally induced SGTR shall be modelled) was closed because a thermally induced SGTR accident sequence was developed and peer-reviewed with no subsequent F&Os. However, the resolutions for the open F&Os, associated with SRs LE-C4 and LE-D5 in the LAR also states, in part, that a sensitivity study demonstrates that un-modelled human failure events (HFEs) related to isolating a ruptured SG following an SGTR initiating event (i.e., apparently not a thermally induced SGTR) has a minimal impact on the PRA results and an acceptable impact of the adoption of TSTF-425.

   i. Clarify whether the evaluation of the impact of the un-modelled isolation HFE described in the F&O resolution for SRs LE-C4 and LE-D5 in the LAR includes the thermally induced SGTR accident sequence. If not, include the thermally induced SGTR accident sequence in the sensitivity study or otherwise evaluate its impact.

   ii. Provide clarification that the sensitivity study related to the exclusion of the SG isolation HFE demonstrated that there was no impact on the relocation of surveillance frequencies, or

   iii. Alternatively to Part ii, if the sensitivity study demonstrates that the exclusion of the operator action does impact any relocation of surveillance frequencies, then propose a mechanism to ensure incorporation of the operator action in the PRA model of record (MOR) prior to implementation of TSTF-425.

Duke Energy Response to PRA RAI 01.a.:

The evaluation of the impact of the un-modelled isolation HFE described in the F&O resolution for SRs LE-C4 and LE-D5 in the LAR does not apply to the thermally induced SGTR accident sequences. The referenced HFE addresses isolating feed flow to, and steam flow from, a SG that has had a tube rupture as the initiating event, in order to allow equalizing primary and secondary side pressure to stop flow out the break. It is only applicable to SGTR initiating events. Thermally induced SGTRs occur following core damage, given that the secondary side of the SG is faulted, such that there is little or no opportunity for operators to isolate the ruptured generator to prevent LERF. Therefore, credit for preventing LERF due to induced SGTRs by isolating a faulted SG is not taken.

A sensitivity study to demonstrate that the exclusion of this operator action for SGTR initiators does not impact any relocation of surveillance frequencies is not feasible since the number and duration of potential STI extensions is highly variable. Therefore, this operator action will be included in the RNP PRA model prior to implementing TSTF-425. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing 10 CFR 50.69.

b. F&O associated with SR IFEV-A7-01 regarding human-induced flood events:

One of the issues provided in the F&O description in LAR Attachment 2 concerns the proper screening of human-induced flood events to determine exclusion from the PRA MOR. The first part of the disposition states, "[t]he sensitivity study performed was overly conservative and attempted to apply all industry human induced failure events on a per piping frequency. This led to a largely over conservative value." There is no description or results for this sensitivity study provided in the LAR.

The second part of the disposition states, "[h]uman induced flooding events are not risk significant for this application as on the whole human induced flooding events in the industry have largely been occurring less often." The disposition makes reference to the period from 1971 to 2011, which appears to match the period used in the EPRI TR-3002000079, "Pipe Rupture Frequencies for Internal Flooding Probabilistic Assessments," Revision 3, which provides flood event probabilities including human-induced events. The NRC staff notes that the EPRI TR is an update of the 2006 TR data and would reflect the decreasing trend of events over that period.

The NRC staff has issued two information notices (IN) related to human-induced flooding events since 2007, IN 2007-01 and IN 2016-11.

Section 5.6 of EPRI TR-1019194, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment," provides specific methodology, including screening, for maintenance-induced flooding events.

Capability Category (CC) I/II for ASME/ANS 2009 PRA Standard for SR IFEV-A7 states, "[i]nclude consideration of human-induced floods during maintenance through application of generic data." SRs IFSN-A10 and IFSN-A15 provide flood event screening criteria. In light of these observations:

i. Describe the sensitivity study mentioned in the F&O disposition. Include in this discussion the purpose of the sensitivity study, what modifications to the PRA model were performed, the results of the study, and the insights from this sensitivity study.

ii. Provide justification, such as industry approved screening criteria, to exclude the remaining maintenance-induced internal flooding events, using industry generic data, from the PRA model, and provide justification that exclusion of these maintenance- induced events does not impact relocation of surveillance frequencies, or

iii. Alternatively to Part ii, propose a mechanism to ensure F&O IFEV-A7-01 will be resolved prior to implementation of TSTF-425. This mechanism should also provide an explicit description of changes that will be made to the PRA model and documentation to resolve the issue.

Duke Energy Response to PRA RAI 01.b.:

The sensitivity study referred to in the LAR apportioned the human induced flood frequency for all generic human induced flood scenarios by allocating the human induced flooding frequency by the amount of piping for that system in the flood area divided by the total amount of system piping in the plant. It then added this failure frequency to the model for the scenarios and quantified the result. It made no attempt in screening or qualifying whether maintenance induced flooding was possible in the specific system or flood area. This had the impact of dramatically overstating the impacts of human induced flooding and increasing CDF and LERF values (44% and 55% delta to CDF and LERF respectively), but not providing any real risk insight in terms of the impact from human induced flooding.

The generic human induced flooding data will be reviewed to determine whether any industry maintenance induced flooding events need to be added to the RNP IFPRA. This will be accomplished by reviewing the generic data from EPRI TR 3002000079 and screening out events that are not applicable to RNP. Following this, maintenance practices and activities at RNP will be reviewed to determine whether the generic maintenance event is applicable. Any of the generic human induced flood events that are not screened out based on this process will be included in the RNP IFPRA. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing TSTF-425.

c. <u>F&O associated with SR IFSN-A8-01</u> regarding door failure heights

The description of the finding in LAR Attachment 2 states the, “[u]se of EPRI door failure criteria of 1 ft [foot] / 3 ft may not be appropriate depending on the actual door attributes and flooding scenario.”

The disposition states, “[t]he current IFPRA assumes that the majority of the components would fail at or around 1 ft to 3 ft,” and concludes the effects, “minimal on modeling results and therefore will have no impact on the quantified values with regard to the RITSTF Initiative 5b application.” The disposition does not discuss how the application provides a bounding assessment for this assumption.

Appendix D of EPRI TR-1019194, “Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment,” provides methodology for determining door failure heights. In light of these observations:

i. Provide justification, such as a sensitivity study, that the exclusion of the correct door failure heights would not impact relocation of surveillance frequencies, or

ii. Alternatively, propose a mechanism that ensures F&O IFSN-A8-01 will be resolved prior to implementation of TSTF-425. This mechanism should also provide an explicit description of changes that will be made to the PRA model and documentation to resolve the issue.

Duke Energy Response to PRA RAI 01.c.:

The 1 ft or 3 ft door failure height is taken from the EPRI Internal Flooding Guidelines (EPRI TR-1019194). This value is used in lieu of a plant specific value. This is a simplifying assumption for the IFPRA. The only aspect of the IFPRA impacted by this assumption is the time available for operators to isolate the flood source prior to the door failing, which impacts the probability of isolation failure. The difference between the generic failure height and a door-specific failure height is expected to be very small since the doors at RNP are fairly typical of nuclear power plant doors. Small differences in failure height would lead to small differences in the time available for isolation which would then lead to a negligible difference in the calculated isolation failure probability. Therefore, the value used from the EPRI Internal Flooding Guidelines is reasonable.

As discussed in the response to PRA RAI 02, uncertainties and related assumptions that are key to the application (i.e., it cannot be quantitatively shown that they do not have the potential to impact the acceptance criteria) are being addressed by the STI evaluation process (application of a multiplicative factor and comparison to ΔCDF/ΔLERF) and performance monitoring program. The use of the STI evaluation process and performance monitoring required by NEI 04-10 is therefore appropriate to address the assumption of door failure heights, which cannot be shown to be conservative in all cases, for the TSTF-425 application.

d. <u>F&O associated with SR IFSN-A8-02</u> regarding door gap flooding propagation

The disposition in LAR Attachment 2 states that it identified one scenario where additional equipment would be impacted. In evaluating the additional failures the disposition states, "[c]rediting flow underneath door gaps would increase the time that operators would be able to potentially isolate the scenario. Therefore as it is currently modeled, scenarios for this flood area are conservative." The disposition concludes, "[t]he timing effects of this open F&O is minimal on modeling results and therefore will have no impact on the quantified values with regard to the RITSTF Initiative 5b application."

In accordance with the SR IFSN A10 (ASME/ANS 2009 PRA Standard), each developed flood scenario includes, "giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs [structures, systems, and components]." The NRC staff notes the exclusion of SSC(s) impacts from initiating events reduces their contribution to risk and can therefore impact their importance measures, thus potentially impacting the importance measures of other SSC(s) as well. In light of these observations:

i. Provide justification, such as a sensitivity study, that the exclusion of the additional PRA SSC impacts from the door gap propagation has no impact on the relocation of surveillance frequencies, or

ii. Alternatively, propose a mechanism that ensures F&O IFSN-A8-02 will be resolved prior to implementation of TSTF-425. This mechanism should also provide an explicit description of changes that will be made to the PRA model and documentation to resolve the issue.

Duke Energy Response to PRA RAI 01.d.:

The only important flood area in the RNP IFPRA that has substantial door gaps found during the walkdown is a large open area that would not allow for water to accumulate to a large depth. In addition, the flood area is abutted by double doors that open out from the flood area. This in effect would preclude the accumulation of water and for a large driving hydrostatic head to develop. As shown in the IFPRA analysis RNP door gaps are in general small and a large depth of water would need to develop to induce a significant amount of water via inter door propagation. Treatment of door gaps is a model uncertainty in any IFPRA. Door gap treatment in the RNP IFPRA is found to be reasonable and in line with industry practice on the evaluation and subsequent treatment.

As discussed in the response to PRA RAI 02, uncertainties and related assumptions that are key to the application (i.e., it cannot be quantitatively shown that they do not have the potential to impact the acceptance criteria) are being addressed by the STI evaluation process (application of a multiplicative factor and comparison to ΔCDF/ΔLERF) and performance monitoring program. The use of the STI evaluation process and performance monitoring required by NEI 04-10 is therefore appropriate to address the assumption of door gap flow, which cannot be shown to be conservative in all cases, for the TSTF-425 application.

PRA RAI 02 – Identifying Key Assumptions and Uncertainties that Could Impact the Application:

Section 1.3 of RG 1.200 describes the level of detail of a PRA required and states, "[i]n general, the level of detail for the base PRA needs to be consistent with current good practice." Current good practices are those practices that are generally accepted throughout the industry and have shown to be technically acceptable in documented analyses or engineering assessments.

LAR Attachment 2, Section 6.0 contains three key assumptions and sources of uncertainty from three PRA models, whereas industry guidance documents such as NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," March 2017 (ADAMS Accession No. ML17062A466), and EPRI TR-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," (2012) address a large number of potential assumptions and uncertainties. For example, one key source of fire PRA modeling assumptions/uncertainty are provided in the LAR, compared to the 2012 EPRI document which identifies 71 potential sources of uncertainty. There appear to be no uncertainties or assumptions associated with large early release (LERF) and internal flooding.

The LAR continues, "[t]he list below represents the modeling assumptions and uncertainties that are considered to have the greatest impact on the HBRSEP PRA results if different reasonable alternative assumptions are utilized."

The NRC staff notes that Stage C, D, E, and F of NUREG-1855 (Revision 1) provides guidance on how to identify key sources of uncertainty relevant to the application.

To address the observations above, the staff requests the following additional information:

a. Provide a detailed summary of the process used to determine the impact of each of the

71 potential sources of uncertainty in the EPRI documents and describe how this process resulted in the final set of three key assumptions and sources of uncertainty presented in LAR Attachment 2, Section 6.0. Include in this discussion an explanation of how the process is in accordance with NUREG-1855, Rev. 1, or another NRC-accepted method.

Duke Energy Response to PRA RAI 02.a.:

Step E-1 (section 7.2) of NUREG 1855, Revision 1 provides guidance for identifying and characterizing those sources of model uncertainty and related assumptions in the PRA required for the application.

Substep E-1.1 of the NUREG recommends using the detailed guidance and a generic list of sources of model uncertainty and related assumptions in EPRI 1016737 for the internal event hazard group (including LERF), and using the examples of sources of model uncertainty for the internal fires, seismic, Low Power Shutdown and Level 2 hazard groups in EPRI 1026511. For RNP, this process was performed by reviewing PRA documentation for generic issues identified in Table A.1 of EPRI 1016737, as well as identifying plant-specific assumptions and uncertainties, and is therefore consistent with step E-1.1 of the NUREG. EPRI 1026511 was not explicitly used to identify generic uncertainties in models other than the internal events model. However, of the models addressed by EPRI 1026511, only the RNP fire PRA is being used to support the current application. The response to part b of this RAI further addressed this.

Substep E-1.2 of NUREG 1855, Revision 1 involves identifying those sources of model uncertainty and related assumptions in the base PRA that are relevant to an application. Those that are irrelevant can be screened from further discussion. However, since this application uses the internal events, internal flood, and fire PRA models for both CDF and LERF, all model uncertainties and related assumptions identified for these models are considered relevant. The original process screened some based on other factors, which is not consistent with the latest version of the NUREG.

Substep E-1.3 of NUREG 1855, Revision 1 involves characterizing the identified sources of model uncertainty and related assumptions. This characterization involves understanding how the identified sources of model uncertainty and related assumptions can affect the PRA. For the RNP uncertainty analysis, this was performed for all identified uncertainties/assumptions.

Substep E-1.4 is a qualitative screening process that involves identifying and validating whether consensus models have been used in the PRA to evaluate identified model uncertainties. As stated in NUREG 1855, Rev. 1, the use of a consensus model eliminates the need to explore an alternative hypothesis. For the RNP uncertainty analysis, some uncertainties/assumptions were screened based on their use of a consensus method, however, others were screened based on additional criteria, which again is not entirely consistent with NUREG-1855, Rev. 1.

Once all relevant uncertainties/assumptions are identified in Step E-1, Step E-2 (section 7.3) of NUREG 1855, Rev. 1 provides guidance for identifying those sources of model uncertainty and related assumptions that are key to the application. The input to this step is the list of the relevant sources of model uncertainty identified in Step E-1. These sources of model uncertainty and related assumptions are then quantitatively assessed to identify those with the potential to impact the results of the PRA such that the application's acceptance guidelines are challenged. This assessment is made by performing sensitivity analyses to determine the importance of the source of model uncertainty or related assumption to the acceptance criteria or guidelines. In the RNP uncertainty analysis, this step was performed qualitatively to arrive at the list of uncertainties, not quantitatively, and therefore is not entirely consistent with the NUREG.

For those uncertainties and related assumptions that are key to the application (i.e., it cannot be quantitatively shown that they do not have the potential to impact the acceptance criteria), Stage F (section 8) of NUREG 1855, Rev. 1, provides guidance on justifying the strategy used to address the key uncertainties that contribute to risk metric calculations that challenge application-specific acceptance guidelines. This portion of the NUREG was not addressed in the original RNP uncertainty analysis. This is further addressed in the response to RAI-02.b.

b. If the process of identifying key sources of uncertainty or assumptions for these PRA models cannot be justified, provide the results of an updated assessment of key sources of uncertainty or assumptions.

Duke Energy Response to PRA RAI 02.b.:

The process for identifying sources of uncertainty and assumptions is described and compared to the process outlined in NUREG-1855 rev. 1, in the response to item a. This comparison shows that the initial RNP identification of sources of model uncertainties and related assumptions was consistent with Substep E-1.1 of the NUREG, with the exception that generic sources of uncertainties for the fire PRA identified in EPRI 1026511 were not explicitly reviewed. However, the process to assess the identified uncertainties/assumptions was not entirely consistent with all portions the latest revision of the NUREG. As such, an updated assessment was performed, as described below.

The first step in the updated evaluation was to identify the risk metrics used as acceptance guidelines for the application. As stated in step 12 of NEI 04-10, the effects on CDF and LERF of surveillance test interval (STI) extensions are considered from all applicable PRAs. The acceptance guidelines are based on the ΔCDF/ΔLERF results. Therefore, individual uncertainties/assumptions in the RNP PRA models (and the 71 generic sources of uncertainties for the fire PRA identified in EPRI 1026511) were assessed based on their ability to impact the ΔCDF/ΔLERF results from STI extensions, as calculated in accordance with NEI 04-10.

Many uncertainties/assumptions in the RNP PRA models are conservative with respect to a particular system, structure or component (SSC), and therefore are conservative with respect to ΔCDF/ΔLERF from an STI extension for that SSC. For example, a 24-hour mission time is assumed for high head safety injection (HHSI) system for all accident scenarios. However, for sequences where a cooldown and depressurization occurs, the actual mission time would be much shorter, and the run failure probability of the HHSI pumps would be reduced. Therefore, the ΔCDF/ΔLERF values calculated for an increased STI for the HHSI pumps would be reduced if the assumption was eliminated by reducing the run failure probability of the HHSI pumps for appropriate scenarios. Another case of conservatism would be where an SSC that could potentially be a backup for the SSC whose STI is being extended (because it performs the same function) is not included in the PRA model. This is typically done when it is determined that only minimal reduction in CDF/LERF would be achieved by modeling the backup system. This is conservative because modeling of backup SSC would reduce the ΔCDF/ΔLERF values calculated for the SSC whose STI is being extended. An example of this would be the RNP

Station Air system which can be used as a backup to the Instrument Air system, but which was not included in the PRA model. Modeling of the Station Air system would reduce the contribution of the Instrument Air system to CDF and LERF, and therefore reduce the ΔCDF/ΔLERF values calculated for any Instrument Air SSC whose STI is being extended. Therefore, assumptions that are shown to be conservative do not need to be addressed further for the TSTF-425 application since they do not challenge the acceptance criteria.

Other assumptions exist in which the assumption would not be characterized as conservative. The treatment of these assumptions in regard to the TSTF-425 application is detailed below.

It should be noted that NEI 04-10 has a specific and conservative process to address SSCs which are not explicitly modeled in the PRA. This process is mainly for assumptions in which potential SSCs could be omitted due to simplifying or level of detail assumptions. In Step 8 of NEI 04-10, it must be established that the PRA modeled components sufficiently represent the SSCs uniquely impacted by the proposed STI change. If not, NEI 04-10 provides two options. The first option is to use qualitative information to provide confidence that the net impact of the STI change would be negligible or zero, from a CDF and LERF perspective (Step 10a). This is conservative since it requires demonstrating (to the satisfaction of the Integrated Decision-making Panel (IDP)) that the STI extension essentially has no impact on CDF/LERF. The second option is to perform a bounding analysis using surrogate basic events in the PRA model (Step 10b). This is conservative since the acceptance criteria for ΔCDF/ΔLERF for the bounding analysis are an order of magnitude lower than for explicitly modeled SSCs. Based on this, uncertainties/assumptions that are used to exclude SSCs from the PRA model are either conservative or are adequately addressed by the NEI 04-10 process (as described above) for non-modeled components.

There are other uncertainties/assumptions in the RNP PRA models which have a minimal impact on overall CDF/LERF results, but which cannot be conclusively shown to be conservative with respect to all SSCs. As discussed in Step E-2 (section 7.3) of NUREG-1855, rev. 1, these uncertainties/assumptions are then quantitatively assessed to identify those with the potential to impact the results of the PRA such that the application's acceptance guidelines are challenged. Without detailed (and sometimes very complex) sensitivity studies for each proposed STI extension for each uncertainty/assumption, it cannot be shown that they will not challenge the acceptance guidelines. If these uncertainties/assumptions are shown to challenge the acceptance guidance for TSTF-425, an appropriate method for dealing with these uncertainties and related assumptions must be developed as stated in Stage F of NUREG-1855 rev. 1 (section 8.1).

Section 8.5 of the NUREG states that performance monitoring is one option that can be used to demonstrate that, "following a change to the design of the plant or operational practices, there has been no degradation in specified aspects of plant performance that are expected to be affected by the change. This monitoring is an effective strategy when no predictive model has been developed for plant performance in response to a change". An example of a performance monitoring approach to address key uncertainties/assumptions is given in section 8.5 of the NUREG for 10CFR 50.69 applications. The NUREG states:

> One example of such an instance is the impact of the relaxation of special treatment requirements (in accordance with 10 CFR 50.69) on equipment unreliability. No consensus approach to model this cause-effect relationship has been developed. Therefore, the approach adopted in NEI 00-04 as endorsed in Regulatory Guide 1.201,

"Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," [NRC, 2006a] is to:

• Assume a multiplicative factor on the SSC unreliability that represents the effect of the relaxation of special treatment requirements

• Demonstrate that this degradation in unreliability would have a small impact on risk. Following acceptance of an application which calls for implementation of a performance monitoring program, such a program would have to be established to demonstrate that the assumed factor of degradation is not exceeded.

The degradation in specified aspects of plant performance that may be expected to be affected by the change due to extension of STIs is the same as that for the 10CFR 50.69 program (i.e., a potential increase in unreliability of SSCs). That is, extension of a STI could potentially result in an increase in the unreliability of the tested components. Since no predictive model of the increase in unreliability following the extension of an STI exists, a similar performance monitoring approach to that taken for 10CFR 50.69 is appropriate for the TSTF-425 application.

The process used to extend surveillances in NEI 04-10 under the TSTF-425 program contains all three of the elements in the example above, as described below:

1) The determination of acceptability for each STI extension is made by first applying a multiplicative factor (proportional to the STI extension factor) to the SSC unreliability that represents the effect of the STI extension. That is, the failure rate of each SSC affected by the STI extension is increased by the factor of the extension. This is considered very conservative in that a potentially significant portion of the failure rate is unrelated to the time between tests. This process is described in Step 12-A1 of NEI 04-10.

2) It then must be demonstrated that this assumed degradation in unreliability would not result in an increase in CDF or LERF of 1.0E-06 or 1.0E-07, respectively. This process is described in steps 12-A1, 12-B3, and 12-A2 of NEI 00-04. If the acceptance criteria for the STI change are met, an additional sensitivity is required (per Step 14 of NEI 04-10) which re-performs all of the ΔCDF and ΔLERF determinations assuming that the standby failure rate of the basic events impacted by the STI change is 3 times larger than that used in the base case assessment. These results are then compared to the 1.0E-06/1.0E-07 ΔCDF/ΔLERF limits. If these limits are exceeded, then revised frequencies should be considered. Although it is acceptable to proceed with the STI extension even if the results of the sensitivity studies are above the limits (provided the base case results are below the limits), in order to do so, qualitative considerations must be developed and provided to the IDP to provide confidence that proceeding with the STI change is still acceptable even though sensitivity studies indicate that the change could exceed the RG 1.174 limits for the individual STI change.

3) Per Step 18 of NEI 04-10, a performance monitoring and feedback program must be implemented which is capable of adequately tracking the performance of equipment that, when degraded, could alter the conclusions that were key to supporting the acceptance of revised surveillance frequencies. The performance monitoring process should have the following attributes:

- Enough tests are included to provide meaningful data.

- The test is devised such that incipient degradation can reasonably be expected to be detected.

- Appropriate parameters are trended as necessary, to provide reasonable assurance that the component will remain operable over the test interval.

This performance monitoring program has been implemented at Duke Energy through procedure AD-EG-ALL-1216. This procedure, in addition to addressing the above attributes requires that the program ensure that component failure rates should not be allowed to rise to unacceptable levels (e.g., significantly higher than the failure rates used to support the change) before detection and corrective action take place.

The use of the STI evaluation process and performance monitoring required by NEI 04-10 is therefore appropriate to address key uncertainties and assumptions, which cannot be shown to be conservative, for the TSTF-425 application.

Based on the above discussion, an updated assessment of sources of uncertainty and assumptions was performed. All uncertainties and assumptions identified in the original RNP process consistent with Substep E-1.1 of NUREG-1855 rev. 1 (i.e., all identified internal events, internal flood, and fire, uncertainties/assumptions), and the 71 generic sources of uncertainties for the fire PRA identified in EPRI 1026511 were reviewed to identify any that are not conservative with respect to TSTF-425 and which are not adequately addressed by NEI 04-10 STI evaluation process and performance monitoring required by NEI 04-10. No uncertainties/assumptions that are not addressed were identified. Therefore, the NEI 04-10 STI evaluation process and performance monitoring required by NEI 04-10 is adequate to address all relevant uncertainties.

PRA RAI 03 – Very Early Warning Fire Detection Systems (VEWFDS) Utilized in the PRA:

In the HBRSEP2 LAR to implement 10 CFR 50.69 (ADAMS Accession No. ML18099A130), the NRC staff notes that Assumption/Uncertainty No. 4 in Attachment 6 states, "[t]he HBRSEP2 Fire PRA assumes Incipient Detection System functions as outlined in NUREG 2180." The disposition to this uncertainty states, "[t]he current method of crediting Incipient Detection at RNP is similar to NUREG 2180 with more credit for operators to prevent fires based upon actual plant experience and plant procedures." It is not clear to the NRC staff how much actual plant experience with fires has been collected and what differences exist between NUREG 2180 and the licensee's approach.

LAR Attachment 2, Section 4.2 states "[t]he HBRSEP fire PRA model was developed using the guidance in NUREG/CR-6580, and the HBSREP fire PRA meets the requirements of the ASME/ANS PRA standard as clarified by Regulatory Guide 1.200, Revision 2." However, in a letter dated July 1, 2016, "Retirement of National Fire Protection Association 805 Frequently Asked Question 08-0046 "Incipient Fire Detection Systems" (ADAMS Accession No. ML16167A444), FAQ 08-0046 that was previously accepted by the NRC was retired. In this letter it was requested of licensees to evaluate the impact of the new guidance on their PRA in accordance with their licensing basis. In light of these observations, address the following:

a. Provide justification, such as a sensitivity study, that the use of FAQ 08-0046 VEWFDS methodology, which is not endorsed by the NRC, has no impact on the relocation of surveillance frequencies. If determined that the use of VEWFDS has an impact, provide a detailed description of the method used for crediting Incipient Detection and technical justification for why it is acceptable for use in the relocation of surveillance frequencies.

Duke Energy Response to PRA RAI 03.a.:

The methodology used for crediting incipient detection at HBRSEP2 is NUREG-2180 *Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities,* Final Report Published December 2016.

b. Alternatively, propose a mechanism that ensures the VEWFDS methodology will be updated to be consistent with NUREG-2180, or other current NRC acceptable methodology prior to implementation of TSTF-425. If this update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, include in this mechanism a process for conducting a focused-scope peer review and ensure any findings are closed by using an approved NRC process.

Duke Energy Response to PRA RAI 03.b.:

The HBRSEP2 Fire PRA model was reviewed against NUREG-2180 guidance. There are no differences between the method used at HBRSEP2 and the method described in NUREG-2180. The wording in the LAR with regards to additional credit for operator actions is not implemented in the model. Thus, treatment of incipient detection at HBRSEP2 fully aligns with NUREG-2180 guidance.

PRA RAI 04 – Incorporation of FLEX into the PRA Model(s):

The NRC memorandum dated May 30, 2017, “Assessment of The Nuclear Energy Institute 16-06, ‘Crediting Mitigating Strategies in Risk-Informed Decision Making,’ Guidance for Risk-Informed Changes to Plants Licensing Basis” (ADAMS Accession No. ML17031A269), provides the NRC’s staff assessment of challenges to incorporating FLEX equipment and strategies into a PRA model in support of risk-informed decision making in accordance with the guidance of RG 1.200. The LAR does not state whether or not the licensee has incorporated FLEX mitigating strategies and associated equipment into the PRA models at Robinson.

Provide the following information separately for internal events PRA, external hazard PRAs, and external hazard screening as appropriate:

a. If FLEX mitigating strategies and associated equipment have not been incorporated into the base PRA and the external hazard evaluations, confirm that FLEX equipment is not modelled.

Duke Energy Response to PRA RAI 04.a.:

FLEX mitigating strategies and associated equipment have not been incorporated into the current Robinson PRA models of record for internal events or external hazards. There is no FLEX equipment in the current model.

b. If FLEX mitigating strategies and associated equipment have been incorporated into the base PRA and the external hazard evaluations but do not impact the relocation of surveillance frequencies, summarize the evaluation supporting the conclusion that there is no impact on the relocation of surveillance frequencies.

Duke Energy Response to PRA RAI 04.b.:

The scenario described in RAI 04.b. is not applicable to HBRSEP2.

c. If FLEX mitigating strategies and associated equipment have been incorporated into the base PRA and the external hazard evaluations and do impact the relocation of surveillance frequencies, provide the following information:

   i. A discussion detailing the extent of incorporation, i.e. summarize the supplemental equipment and compensatory actions, including FLEX strategies that have been quantitatively credited for each of the PRA models used to support this application.

   ii. A discussion detailing the methodology used to assess the failure probabilities of any modeled equipment credited in the licensee's mitigating strategies (i.e., FLEX). The discussion should include a justification explaining the rational for parameter values, and whether the uncertainties associated with the parameter values are considered in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200.

   iii. A discussion detailing the methodology used to assess operator actions related to FLEX equipment and the licensee personnel that perform these actions. The discussion should include:

      (1) A summary of how the licensee evaluated the impact of the plant-specific HEPs and associated scenario-specific performance shaping factors listed in (a)-(j) of supporting requirement HR-G3 of the ASME/ANS RA-Sa-2009 PRA standard.

      (2) Whether maintenance procedures for the portable equipment were reviewed for possible pre-initiator human failures that renders the equipment unavailable during an event, and if the probabilities of the pre-initiator human failure events were assessed as described in HLR-HR-D of the ASME/ANS RA-Sa-2009 PRA standard.

      (3) If the licensee's procedures governing the initiation or entry into mitigating strategies are ambiguous, vague, or not explicit, a discussion detailing the technical bases for probability of failure to initiate mitigating strategies.

   iv. The ASME/ANS RA-Sa-2009 Standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard.

      (1) Provide an evaluation of the model changes associated with incorporating mitigating strategies, which demonstrates that none of the following criteria are satisfied: (1) use of new methodology, (2) change in scope that impacts the

significant accident sequences or the significant accident progression sequences, (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences, OR

(2) Propose a mechanism to ensure that a focused-scope peer review is performed on the model changes associated with incorporating mitigating strategies, and associated F&Os are resolved to CC II prior to implementation of TSTF-425.

Duke Energy Response to PRA RAI 04.c.:

Items i.-iv. in RAI 04.c. are not applicable to HBRSEP2.

Serial: RA-18-0194

H.B. Robinson Steam Electric Plant, Unit 2
Docket No. 50-261 / Renewed License No. DPR-23

Response to NRC Request for Additional Information (RAI) Regarding Application to Adopt TSTF-425, Revision 3

Attachment

HBRSEP2 PRA Implementation Items

The table below identifies the items that are required to be completed prior to implementation of the Surveillance Frequency Control Program at H.B. Robinson Steam Electric Plant, Unit 2 (HBRSEP2). The issues identified below will be addressed and any associated changes made, focused scope peer reviews performed on changes that are PRA upgrades as defined in the PRA standard (ASME/ANS RA-Sa-2009, as endorsed by RG 1.200, Revision 2), and findings resolved and reflected in the PRA of record prior to implementation of the Surveillance Frequency Control Program.

Robinson TSTF-425 PRA Implementation Items	
Description	Resolution
i. The HBRSEP2 internal flood model does not account for generic human induced flooding data as described in response to RAI 1.b in Duke letter dated November 13, 2018. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing the Surveillance Frequency Control Program.	Duke Energy will update the HBRSEP2 internal flood model to account for generic human induced flooding events using an industry accepted methodology described in the response to RAI 1.b. in Duke letter dated November 13, 2018.
ii. Human Failure Events (HFEs) related to isolating a ruptured SG following a thermally induced steam generator tube rupture (SGTR) are not represented in the internal events model as described in response to RAI 1.a. in Duke letter dated November 13, 2018. If this update is determined to be a PRA model upgrade per the 2009 ASME/ANS PRA standard, then a focused scope peer review will be conducted. Any findings from the focused scope peer review will be resolved and closed per an NRC approved process prior to implementing the Surveillance Frequency Control Program.	Duke Energy will update the HBRSEP2 internal events model to include these operator actions.