Nebraska Public Power District

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NLS2016067
December 7, 2016

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

Subject: Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Revision 3
Cooper Nuclear Station, Docket No. 50-298, DPR-46

References:

1. Email from Thomas Wengert, U.S. Nuclear Regulatory Commission, to Jim Shaw, Nebraska Public Power District, dated October 27, 2016, "Cooper Nuclear Station - Formal Request for Additional Information Concerning License Amendment Request to Adopt TSTF-425, Revision 3" (CAC MF7498)

2. Letter from Oscar A. Limpias, Nebraska Public Power District, to the U.S. Nuclear Regulatory Commission, dated March 22, 2016, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program"

Dear Sir or Madam:

The purpose of this letter is for the Nebraska Public Power District to respond to the Nuclear Regulatory Commission's Request for Additional Information (RAI) (Reference 1) related to the Cooper Nuclear Station License "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a License Controlled Program" (Reference 2).

The response to the RAI is provided in Attachment 1 to this letter. Attachment 2 provides a revised mark-up and re-typed Technical Specifications (TS) page 3.3-12 and a revised mark-up of TS Bases page B 3.3-37.

This letter does not contain any new regulatory commitments.

If you have any questions concerning this matter, please contact Jim Shaw, Licensing Manager, at (402) 825-2788.

A006
NRR

COOPER NUCLEAR STATION
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I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 12/7/16
(Date)

Sincerely,

Ken Higginbotham
General Manager of Plant Operations

/dv

Attachments: 1. Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Revision 3

2. Revised Mark-up and Re-Typed Technical Specifications (TS) Page 3.3-12 and Revised Mark-up of TS Bases Page B 3.3-37

cc: Regional Administrator w/ attachments
USNRC – Region IV

Cooper Project Manager w/ attachments
USNRC – NRR Plant Licensing Branch IV-2

Senior Resident Inspection w/ attachments
USNRC – CNS

NPG Distribution w/o attachments

CNS Records w/ attachments

Attachment 1

Response to Nuclear Regulatory Commission Request for Additional Information Regarding License Amendment Request to Adopt TSTF-425, Revision 3

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

The Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) regarding the License Amendment Request to adopt Technical Specifications Task Force (TSTF) Traveler, TSTF-425, Revision 3, "Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specification Task Force Initiative 5B" is shown in italics. The Nebraska Public Power District (NPPD) response to the request is shown in normal font.

Technical Specifications Branch (STSB) RAI-1

The NRC's regulatory requirements related to the content of the Technical Specifications (TSs) Surveillance Requirements (SRs) are contained in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36(c)(3). Per 10 CFR 50.36(c)(3), SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

During the NRC staff's review of a change to ensure that the change is in accordance with 10 CFR 50.36, the staff uses the approved traveler, TSTF-425, Revision 3, as guidance. According to this guidance, the proposed change relocates all periodic surveillance frequencies from the TS and places the frequencies under licensee control in accordance with the new Surveillance Frequency Control Program, except for those meeting any of the four (4) exclusion criteria.

a. *SR 3.3.1.2.4 contains a two part frequency, "12 hours during CORE ALTERATIONS AND 24 hours" marked for replacement in its entirety with proposed "Insert 1." The first part of the frequency, "12 hours during CORE ALTERATIONS," appears to meet one of the four exclusion criteria of part 2.0 of the approved TSTF-425, Revision 3 traveler as a "frequency that is related to a specific condition ..." Propose a new markup that replaces only the second part of the frequency with proposed "Insert 1" or, alternately, explain why the exclusion does not apply and how the complete frequency will be addressed in the Surveillance Frequency Control Program.*

NPPD Response

NPPD has provided a revised mark-up and re-typed TS Page 3.3-12 and revised mark-up of TS Bases page B 3.3-37 in Attachment 2. The TS pages replace only the second part of the frequency with proposed "Insert 1."

Probabilistic Risk Assessment Licensing Branch (APLA) RAI-1

NRC Regulatory Guide (RG) 1.177 Revision 1, Section 2.3.3.2 recommends that initiating events resulting from support system failure (e.g., service water, component cooling water, and instrument air) be modeled explicitly in the logic model (i.e., fault tree models developed in the Probabilistic Risk Assessment (PRA)). Any TS changes for these systems will affect the corresponding initiating event frequency as well as the system unavailability and availability of other supported systems. The effect of TS changes on these initiating event frequencies should be considered.

The CNS Internal Events Probabilistic Risk Assessment (IEPRA) Peer Review identified Facts & Observations (F&Os) associated with supporting requirement QU-F5-01 related to American Society of Mechanical Engineers (ASME)/ American Nuclear Society (ANS) RA-Sa-2009 standard element QU-F5 due to the method used for quantifying initiating event frequencies where quantification was performed separately, and a point estimate value for initiating event frequency was inserted into the PRA top event model, rather than quantifying the entire logic model as single top event models for core damage frequency (CDF) and large early release frequency (LERF).

a. *Clarify how the CNS PRA models address this concern in Step 8 of the Nuclear Energy Institute (NEI) 04-10, Revision 1, guidance to assure accuracy in calculations of net change in CDF and LERF for evaluations of STIs.*

NPPD Response

Maintenance of the Cooper Nuclear Station (CNS) IEPRA through the routine update process has addressed this observation through incorporation of the supporting system initiating events fault trees into the single top model for CDF and LERF [Reference 7, page ES-5]. Therefore, a point estimate value for these initiating event frequencies is no longer required to be inserted into the PRA top event model because the support system initiating event fault trees are an integral part of the CDF and LERF quantifications.

Because the support system initiating event fault trees are now an integral part of the quantification of CDF and LERF, accuracy of the net change to CDF and LERF for Surveillance Test Intervals (STI) is assured in the area of use of support system initiator modeling.

APLA RAI-2

NRC RG 1.177, Revision 1, Section 2.3, recommends that the licensee demonstrate that its PRA is valid for assessing the proposed TS changes and identify the impact of the TS change on plant risk.

The CNS fire PRA peer review identified F&O HR-G7 related to ASME/ANS RA-Sa-2009 standard. In review of SR HR-G7, post-initiator Human Reliability events do not appear to have been evaluated (or at least documented). It is unclear whether sequence cutsets involving

multiple human errors are treated as dependent vs. independent events, for which human error probabilities are adjusted accordingly. For example, additional review of material provided in the CNS "Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association Standard 805," dated January 14, 2013, states in part, "none of the HFEs created for the Fire PRA required the use of the 1E-06 floor" (Reference 1).

ASME/ANS RA-Sa (2009) standard elements HR-G7 and QU-C1 may not preclude treatment of human errors as being "independent" provided: (a) cutsets containing multiple human errors are not screened out, and (b) that justification is properly documented why certain human error probabilities are treated as independent.

a. *Identify if any of the joint human factors events (HFEs) for the FPRA use the floor value of 1.0E-05 or less. Provide justification and a sensitivity analysis for any of the HFEs if a value of less than 1.0E-05 has been used.*

NPPD Response

As required by the ASME/ANS RA-Sa (2009) [Reference 3] standard element HR-G7, and QU-C1, the CNS fire PRA sequence cutsets involving multiple human errors were evaluated for dependency when fire PRA results were analyzed.

Discussions detailed in the Office of Nuclear Regulatory Regulation's Safety Evaluation for transition to National Fire Protection Association 805 [Reference 2, page 78] detail the CNS fire PRA human reliability dependency analysis and application of floor values. This safety evaluation concludes that the human reliability dependency analyses along with floor value assignments are acceptable and apply a method endorsed by NUREG 1921 [Reference 1]. This endorsed method will continue to be used for the fire PRA in support of surveillance test interval change evaluations. Therefore, sensitivity analysis is not warranted.

APLA RAI-3

The CNS fire PRA (FPRA) Peer Review identified F&Os associated with supporting requirements (SRs) SY-A2, SY-C2, SY-A3, and DA-C2. The peer review team provided comment that significant system modelling had been performed. Furthermore, the CNS disposition to address SY-A3 states, "new components and fire-induced impacts should be considered." It is unclear, for the FPRA in which the system modelling has been modified and/or new components have been introduced into the FPRA, that the updates have been appropriately considered and, if necessary, incorporated into the internal events PRA (IEPRA) to reflect the as-built and as-operated systems.

a. *Confirm that the enhanced modeling of the feedwater system for the fire PRA has been incorporated into the IEPRA. If not, provide a discussion to justify why the feedwater system modelling enhancements were not incorporated into the IEPRA to support future*

surveillance test interval (STI) evaluations, including why the exclusion will have no more than a negligible effect on the STI evaluations.

b. *Confirm, as appropriate, for the new components listed in NEDC 09-079, if any have been incorporated into the IEPRA model. If not, provide a discussion to justify why these components have been excluded from the IEPRA to support future STI evaluations, including why the exclusion will have no more than a negligible effect on the STI evaluations.*

NPPD Response

APLA RAI-3(a)

Modeling of the feedwater system contained in the fire PRA has been previously reviewed against the IEPRA. The recommendation from the fire peer review resulted in a model change request to document and track review of the fire PRA feedwater modeling. This review was performed as part of the routine PRA update process to apply the recommendation made in the fire PRA peer review. The update review found that the elements of the fire PRA feedwater system modeling were contained in the current internal events model [Reference 8]. Therefore, use of the internal events PRA to support future surveillance test interval evaluations is acceptable, and has negligible effect.

APLA RAI-3(b)

New components listed in Nuclear Engineering Design Calculation (NEDC) 09-079 [Reference 9, Attachment D, Table D-1] have been evaluated for applicability with respect to the IEPRA. This evaluation is summarized in the following Table 1. Evaluation determined that the new components added by the fire PRA addressed failure modes specific to fire events, and did not apply when evaluated for IEPRA modeling.

Acronyms used in Table 1:

CC	Common Cause	MV	Motor Valve
CKV	Check Valve	PC	Primary Containment
CRD	Control Rod Drive	PCI	Primary Containment Isolation
CS	Core Spray	PRA	Probabilistic Risk Assessment
CSS	Core Spray System	RCI	Reactor Core Isolation Cooling
CV	Check Valve	RCIC	Reactor Core Isolation Cooling
LCS	Low Pressure Core Spray	RHR	Residual Heat Removal
LK	Leak	SO	Solenoid Operated
LOCA	Loss of Coolant Accident	SO	Spurious Operation
MOV	Motor Operated Valve	SOV	Solenoid Operated Valve

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
CRD-SOV-SO140A	TRIP SYSTEM A BACKUP SCRAM VALVE SO140A	CRD	CRD-SOV-CC-SO140A	NORMALLY CLOSED SOV CRD-SOV-SO140A FAILS TO TRANSFER	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE IN THE REACTOR PROTECTION SYSTEM, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS ABILITY TO EMPLOY ALTERNATE SCRAM CAPABILITIES, AND SCRAM VOLUME ISOLATION. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS NON-FIRE FAILURE MODES ARE JUDGED NOT CREDIBLE, AND SUBSUMED.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
CRD-SOV-SO140B	TRIP SYSTEM B BACKUP SCRAM VALVE SO140B	CRD	CRD-SOV-CC-SO140B	NORMALLY CLOSED SOV CRD-SOV-SO140B FAILS TO TRANSFER	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE IN THE REACTOR PROTECTION SYSTEM, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS ABILITY TO EMPLOY ALTERNATE SCRAM CAPABILITIES, AND SCRAM VOLUME ISOLATION. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS NON-FIRE FAILURE MODES ARE JUDGED NOT CREDIBLE, AND SUBSUMED.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
CS-CV-18CV	CSS A TESTABLE CHECK	CS	LCS-CKV-LK-18CV	CHECK VALVE 18CV INTERNAL LEAKAGE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE FOR THE REACTOR PRESSURE BOUNDARY, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS FIRE INDUCED SPURIOUS OPENING OF A PRIMARY CONTAINMENT VALVE. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE INCLUDED AS PART OF THE INTERFACING LOCA INITIATING EVENT FREQUENCY IN THE INTERNAL EVENTS MODEL.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
CS-CV-19CV	CSS B TESTABLE CHECK	CS	LCS-CKV-LK-19CV	CHECK VALVE 19CV INTERNAL LEAKAGE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE FOR THE REACTOR PRESSURE BOUNDARY, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS FIRE INDUCED SPURIOUS OPENING OF A PRIMARY CONTAINMENT VALVE. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE INCLUDED AS PART OF THE INTERFACING LOCA INITIATING EVENT FREQUENCY IN THE INTERNAL EVENTS MODEL.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
PC-MOV-305MV-PASSIVE	BYPASS VALVE AROUND PC-MOV-230MV	PC	PCI-MOV-SO-305MV	PC-MOV-305MV SPURIOUSLY OPENS DUE TO FIRE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE FOR PRIMARY CONTAINMENT, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS FIRE INDUCED SPURIOUS OPENING OF A PRIMARY CONTAINMENT VALVE. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE JUDGED NOT CREDIBLE AND SUBSUMED.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
PC-MOV-306MV-PASSIVE	BYPASS VALVE AROUND PC-MOV-231MV	PC	PCI-MOV-SO-306MV	PC-MOV-306MV SPURIOUSLY OPENS DUE TO FIRE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE FOR PRIMARY CONTAINMENT, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS FIRE INDUCED SPURIOUS OPENING OF A PRIMARY CONTAINMENT VALVE. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE JUDGED NOT CREDIBLE AND SUBSUMED.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
RCIC-CV-18CV	CONDENSATE SUPPLY TO RCIC SYSTEM - PRESSURE MAINTENANCE	RCIC	RCI-CKV-LK-18CV	CHECK VALVE RCIC-CV-18CV INTERNAL LEAKAGE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE NFPA 805 FAILURE MODE ASSOCIATED WITH RCIC PRESSURE BOUNDARY. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE JUDGED NOT CREDIBLE AND SUBSUMED.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
RCIC-CV-19CV	CONDENSATE SUPPLY TO RCIC SYSTEM - PRESSURE MAINTENANCE	RCIC	RCI-CKV-LK-19CV	CHECK VALVE RCIC-CV-19CV INTERNAL LEAKAGE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE NFPA 805 FAILURE MODE ASSOCIATED WITH RCIC PRESSURE BOUNDARY. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE JUDGED NOT CREDIBLE AND SUBSUMED.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
RHR-CV-26CV	INJECTION CHECK	RHR	RHR-CKV-LK-26CV	RHR-CV-26CV INTERNAL LEAKAGE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE FOR THE REACTOR PRESSURE BOUNDARY, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS FIRE INDUCED SPURIOUS OPENING OF A PRIMARY CONTAINMENT VALVE. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE INCLUDED AS PART OF THE INTERFACING LOCA INITIATING EVENT FREQUENCY IN THE INTERNAL EVENTS MODEL.

APLA RAI 3(b), Table 1, New Components and/or New Random Failure Modes Added to Support the Fire PRA

CIC	COMPONENT DESCRIPTION	SYSTEM	BASIC EVENT	BASIC EVENT DESCRIPTION	COMMENTS
RHR-CV-27CV	INJECTION CHECK	RHR	RHR-CKV-LK-27CV	RHR-CV-27CV INTERNAL LEAKAGE	THIS EVENT IS NOT APPLICABLE, AND DOES NOT REQUIRE INCORPORATION INTO THE INTERNAL EVENTS MODEL. THIS EVENT ADDRESSES A SPECIFIC FIRE FAILURE MODE FOR THE REACTOR PRESSURE BOUNDARY, AND IS REQUIRED BY THE FIRE PRA TO ADDRESS FIRE INDUCED SPURIOUS OPENING OF A PRIMARY CONTAINMENT VALVE. THIS IS NOT REQUIRED BY THE INTERNAL EVENTS MODEL AS THE ASSOCIATED NON-FIRE FAILURE MODES ARE INCLUDED AS PART OF THE INTERFACING LOCA INITIATING EVENT FREQUENCY IN THE INTERNAL EVENTS MODEL.

APLA RAI-4

NRC RG 1.200, Revision 2, provides staff guidance to ensure that the PRA Technical Adequacy reflects the plant as-built and as-operated. RG 1.200, Revision 2, also directs that the risk perspective used in a risk-informed application be based on a consideration of the total risk, which includes contributions from initiating events whose causes are attributable to both internal and external hazards. CNS did not explain how it plans to use the latest available external hazard information as a part of its STI evaluation, (e.g., revised seismic hazard frequencies from U.S. Geological Survey (USGS) 2008-1128, "Documentation for the 2008 Update of the U.S. Regional Seismic Hazard Maps," or, as an alternative, the results from the NRC "Safety Assessment Results for GI-199" (ADAMS Accession No. ML100270582)). CNS' submittal states that the IPEEE program was a "one-time review" and, therefore, has not been updated since it was performed in 1996. Hazard characteristics can change over time due to physical changes and changes in the available information.

a. *Summarize what will be considered for CNS' external hazards evaluation in support of future STI evaluations. Describe how updated information pertaining to all external event hazards will be incorporated.*

NPPD Response

As indicated in Section 3.0 of Attachment 2 in the submittal [Reference 4], it is recognized that the use of the available external event information from the Individual Plant Examination for External Events (IPEEE) is limited, but the NEI 04-10 [Reference 5] methodology allows a qualitative screening or bounding analysis to provide justification for acceptability of proposed surveillance frequency changes. Therefore, the intent is not to directly use any numerical results from the IPEEE, but to qualitatively assess any available information to determine the impact on proposed surveillance interval changes consistent with the NEI 04-10 methodology. The qualitative information and/or the bounding analysis from the internal events analysis as described in Section 1.0 of the submittal must be acceptable and reflective of current plant configurations and applicable industry information on external hazards. This information is documented for each STI change provided to the Integrated Decision Panel. Therefore, if the qualitative assessment is used it will be acceptable and reflective of the current plant configuration and applicable industry information on external hazards.

By following Steps 10a and 10b of the NEI 04-10 guidance, the evaluation of seismic risk and other external events risk supporting this application will reflect and consider current plant configurations and applicable industry information on external hazards. The IPEEE is not a living document and has not been updated. As a result, the external event risk information from the IPEEE is limited to qualitative insights. For the STI change evaluations, the intent is not to directly use any numerical results from the IPEEE, but to qualitatively assess any available information to determine the impact on the proposed surveillance interval changes, consistent with Step 10a of the NEI 04-10 methodology. This qualitative assessment of external event risk will include a review of applicability to the current plant configuration and applicable industry information on external hazards. Additionally, for some STI change evaluations, per Step 10b of

the NEI 04-10 methodology, qualitative reasoning and very low changes to core damage frequency (ΔCDF) and large early release frequency (ΔLERF) results from the internal events analysis may be sufficient to support the STI change evaluation where Step 10b reads in part:

> *"Alternative evaluations for the impact from external events and shutdown events are also deemed acceptable at this point. For example, if the ΔCDF and ΔLERF values have been demonstrated to be very small from an internal events perspective based on detailed analysis of the impact of the* [Structures, Systems, Components] *SSC being evaluated for the STI change, and if it is known that the CDF or LERF impact from external events (or shutdown events as applicable) is not specifically sensitive to the SSC being evaluated (by qualitative reasoning), then the detailed internal events evaluations and associated required sensitivity cases (as described in Step 14) can be used to bound the potential impact from external events and shutdown PRA model contributors. "*

Qualitative evaluation of external events risk in support of Step 10b would also include consideration of applicability to the current plant configuration and applicable industry information on external hazards.

Therefore, by following Steps 10a and 10b of the NEI 04-10 guidance, the evaluation of external events will reflect and consider the current plant configuration and applicable industry information on external hazards.

APLA RAI-5

Fire ignition frequencies and non-suppression probabilities were previously developed in NUREG/CR-6850/EPRI 1011989 and revised in Supplement 1 to NUREG/CR-6850/EPRI 1019259. For SR QU-E3, the CNS disposition states, in part, "generic fire frequencies are directly based on assumptions in NUREG/CR-6850 (including FAQ-48 enhancements)." NUREG 2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database [(FEDB)]" (Reference 2), dated January 2015, provides updated fire ignition frequency estimates using the most current FEDB data while applying methodology enhancements.

a. *Explain what updated fire ignition frequencies will be used for future STI evaluations. If the updated fire ignition values from the NRC-endorsed guidance NUREG-2169 will not be used, provide discussion to justify why not.*

NPPD Response

The CNS PRA Program has included the updated fire ignition values from the NRC-endorsed guidance NUREG-2169 [Reference 10] through application of the PRA model change request process, as entered by model change request (MCR) CN2014-088 [Reference 11]. This formal change process ensures that these updated fire ignition frequencies are included during a future fire PRA update.

MCR CN2014-088 was approved within the CNS PRA maintenance update process that was developed to meet the PRA standards/guidance [References 3 and 6] requirements for PRA model maintenance/update. This model change request will provide interim measures to ensure that future applications of the fire PRA for STI evaluations address the impact from the NUREG-2169 updated fire ignition values. The Cooper Nuclear Station STI evaluation process will require identification of any open model change requests and evaluate them for applicability.

APLA RAI-6

For the disposition of the F&O related to PRM-B9 of the FPRA Peer Review, the licensee states, "PRAQuant solves each fire scenario by setting all internal events initiators to 0.0 and setting fire initiators and those basic events representing components impacted by the fire to 1.0." It is unclear when new basic events have been added to the FPRA logic, if both their random (non-fire) and fire-induced failure probabilities are logically modelled to ensure both failure probabilities propagate through the logic for quantification when the internal events initiators have been set to 0.0.

a. *Provide a discussion to confirm for basic events involving both their random (non-fire) and fire-induced failure probabilities that both are included in the quantification, when appropriate. If replacing a random failure probability with that for a fire-induced failure on a particular basic event will have no more than a negligible impact on the internal event risk (e.g., the fire-induced failure probability is always so much greater than that for the random failure that any cutsets involving the same basic event will always be dominated by the fire-induced scenario), justify and confirm.*

NPPD Response

The CNS fire PRA quantification of CDF and LERF includes both random (non-fire) and fire-induced failure probabilities. This was achieved through using the internal events PRA structure as a foundation to develop the fire PRA. The development of the fire PRA model using the internal events structure was done such that both appropriate random event (non-fire) and fire-induced failure probabilities propagate when the internal events initiators have been set to 0.0 to allow for quantification of fire events.

General practice during the modeling of fire failure events and probabilities included modeling the fire-induced equipment failures as separate inputs. This allowed the fire PRA to preserve the applicable random failures from the internal events model. Therefore, there is negligible impact on the internal event portion of risk from use of fire failure events when the fire PRA is quantified.

References:

1. United States Nuclear Regulatory Commission, NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines," dated July 2012

2. Safety Evaluation by the Office of Nuclear Reactor Regulation, "Cooper Nuclear Station - Issuance of Amendment Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c)," dated April 29, 2014

3. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated February 2, 2009

4. NLS2016010, "Application for Technical Specifications Change Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee Controlled Program," dated March 22, 2016

5. NEI 04-10, "Risk-Informed Technical Specifications Initiative 5b, Risk-Informed Method for Control of Surveillance Frequencies, Industry Guidance Document," Revision 1, dated April 2007

6. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009

7. CNS PSA-013, "PRA Summary Notebook," Revision 3, dated February 2015

8. CNS PRA Model Change Request Number CN2013-028, "Transfer of Enhance Feedwater Modeling in Fire PRA to Internal Events PRA," dated June 2013

9. NEDC 09-079, "Scientech Calculation 17712-002 Task 7.5 Risk Model Development," Revision 2, dated October 2014

10. United States Nuclear Regulatory Commission, NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009," dated January 2015

11. CNS PRA Model Change Request Number CN2014-088, "New EPRI Report on Fire Frequencies," dated December 2014

Attachment 2

Revised Mark-up and Re-Typed Technical Specifications (TS) Page 3.3-12 and Revised Mark-up of TS Bases Page B 3.3-37

Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.4	----------NOTE---------- Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ---------- Verify count rate is ≥ 3.0 cps with a signal to noise ratio ≥ 2:1.	12 hours during CORE ALTERATIONS AND ~~24 hours~~ INSERT 1
SR 3.3.1.2.5	Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	~~7 days~~ INSERT 1
SR 3.3.1.2.6	----------NOTE---------- Not required to be performed until 12 hours after IRMs on Range 2 or below. ---------- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	~~31 days~~ INSERT 1
SR 3.3.1.2.7	----------NOTES---------- 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. ---------- Perform CHANNEL CALIBRATION.	~~24 months~~ INSERT 1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2.4	----------NOTE---------- Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies in the associated core quadrant. ---------- Verify count rate is ≥ 3.0 cps with a signal to noise ratio ≥ 2:1.	12 hours during CORE ALTERATIONS <u>AND</u> In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2.5	Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2.6	----------NOTE---------- Not required to be performed until 12 hours after IRMs on Range 2 or below. ---------- Perform CHANNEL FUNCTIONAL TEST and determination of signal to noise ratio.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.2.7	----------NOTES---------- 1. Neutron detectors are excluded. 2. Not required to be performed until 12 hours after IRMs on Range 2 or below. ---------- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program

BASES

SURVEILLANCE REQUIREMENTS (continued)

Table 3.3.1.2-1, footnote (b), only the a. portion of this SR is required. Note 2 clarifies that more than one of the three requirements can be met by the same OPERABLE SRM. ~~The 12 hour Frequency is based upon operating experience and supplements operational controls over refueling activities that include steps to ensure that the SRMs required by the LCO are in the proper quadrant.~~ INSERT 2

SR 3.3.1.2.4

This Surveillance consists of a verification of the SRM instrument readout to ensure that the SRM reading is greater than a specified minimum count rate with the detector full-in, which ensures that the detectors are indicating count rates indicative of neutron flux levels within the core. With few fuel assemblies loaded, the SRMs will not have a high enough count rate to satisfy the SR. Therefore, allowances are made for loading sufficient "source" material, in the form of irradiated fuel assemblies, to establish the minimum count rate.

To accomplish this, the SR is modified by a Note that states that the count rate is not required to be met on an SRM that has less than or equal to four fuel assemblies adjacent to the SRM and no other fuel assemblies are in the associated core quadrant. With four or less fuel assemblies loaded around each SRM and no other fuel assemblies in the associated core quadrant, even with a control rod withdrawn, the configuration will not be critical. This SR does not require determination of the noise ratio.

The Frequency is based upon channel redundancy and other information available in the control room, and ensures that the required channels are frequently monitored while core reactivity changes are occurring. When no reactivity changes are in progress, the Frequency is ~~relaxed from 12 hours to 24 hours.~~ controlled under the Surveillance Frequency Control Program.

SR 3.3.1.2.5 and SR 3.3.1.2.6

Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated channel will function properly. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. SR 3.3.1.2.5 is required in MODE 5 and ~~the 7 day~~