

## **NRR-PMDAPEm Resource**

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**From:** Wengert, Thomas  
**Sent:** Thursday, October 27, 2016 11:28 AM  
**To:** Shaw, Jim D.  
**Cc:** Van Der Kamp, David W.; Flaherty, James R.; Snyder, Pete; Driver, Adrienne  
**Subject:** Cooper Nuclear Station - Formal Request for Additional Information Concerning License Amendment Request to Adopt TSTF-425 Revision 3 (CAC MF7498)  
**Attachments:** Cooper Nuclear Station Draft RAI Regarding TSTF-425 LAR.pdf

On October 21, 2016, the U.S Nuclear Regulatory Commission (NRC) staff sent Nebraska Public Power District (NPPD or licensee) the draft Request for Additional Information (RAI) provided below and attached. This RAI relates to a license amendment request to adopt Technical Specifications Task Force (TSTF)–425, Revision 3, “Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specification Task Force Initiative 5b,” for the Cooper Nuclear Station (CNS).

NPPD subsequently informed the NRC staff that the information requested by the NRC staff was understood and that no additional clarification of the RAI was necessary. NPPD requested that the response date for this RAI be revised to 45 days from the date of this correspondence. The NRC staff informed NPPD that this response date is acceptable. The staff also informed NPPD that a publicly available version of this formal RAI would be placed in the NRC’s Agencywide Documents Access and Management System (ADAMS).

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By letter dated March 22, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16110A425), Nebraska Public Power District submitted a license amendment request to adopt Technical Specifications Task Force (TSTF)–425, Revision 3, “Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specification Task Force Initiative 5b,” for the Cooper Nuclear Station (CNS).

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information, as described in the attached request for additional information (RAI), is required for the staff to complete its review of the CNS TSTF-425 application.

This RAI is identified as draft at this time to confirm your understanding of the information that the NRC staff needs to complete the evaluation. If the request for information is understood, please respond to this request for additional information within 30 days of the date of this request. Please call me at 301-415-4037 if you would like to set up a conference call to clarify this request for information.

Regards,

Tom Wengert  
Project Manager – Cooper Nuclear Station  
NRR/DORL/LPL4-2  
(301) 415-4037

**Hearing Identifier:** NRR\_PMDA  
**Email Number:** 3128

**Mail Envelope Properties** (Thomas.Wengert@nrc.gov20161027112800)

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**Received Date:** 10/27/2016 11:28:00 AM  
**From:** Wengert, Thomas

**Created By:** Thomas.Wengert@nrc.gov

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Tracking Status: None

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Tracking Status: None

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**Options**

**Priority:** Standard

**Return Notification:** No

**Reply Requested:** No

**Sensitivity:** Normal

**Expiration Date:**

**Recipients Received:**

DRAFT REQUEST FOR ADDITIONAL INFORMATION

REGARDING LICENSE AMENDMENT REQUEST TO ADOPT TSTF-425, REVISION 3,

“RELOCATE SURVEILLANCE FREQUENCIES TO LICENSEE CONTROL– RISK INFORMED

TECHNICAL SPECIFICATION TASK FORCE INITIATIVE 5B”

COOPER NUCLEAR STATION

DOCKET NO. 50-298

By letter dated March 22, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16110A425), Nebraska Public Power District, (the licensee) submitted a license amendment request to adopt Technical Specifications Task Force (TSTF)–425, Revision 3, “Relocate Surveillance Frequencies to Licensee Control – Risk Informed Technical Specification Task Force Initiative 5b,” for the Cooper Nuclear Station (CNS).

The U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information, as described in the request for additional information (RAI) below, is required for the staff to complete its review of the CNS TSTF-425 application.

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.

## 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.

## 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.

### 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.

### 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.

#### 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.

#### 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.

#### References

1. Kenneth Higginbotham, Nebraska Public Power District, letter to U.S. Nuclear Regulatory Commission, “Cooper Nuclear Station, Docket No. 50-298, License No. DPR-46, Response to Request for Additional Information Regarding License Amendment Request to Adopt National Fire Protection Association Standard 805,” dated January 14, 2013 (ADAMS Accession No. ML13018A006).
2. NUREG-2169, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database,” dated January 2015 (ADAMS Accession No. ML15134A204)