Nebraska Public Power District
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NLS2008044
May 8, 2008

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
612 East Lamar Blvd, Suite 400
Arlington, TX 76011-4125

Subject: Additional Information for Consideration in Addressing Inspection Finding
Cooper Nuclear Station, Docket No. 50-298, DPR-46

Dear Sir:

The purpose of this correspondence is for the Nebraska Public Power District (NPPD) to provide additional information to the Nuclear Regulatory Commission (NRC) for consideration as an input to resolving issues associated with an inspection finding documented in Inspection Report 2008-007. NPPD will address these and other facets of the inspection finding during a Regulatory Conference scheduled for May 13, 2008.

Attachment 1 provides a summary of the information being provided in the enclosures. Enclosure 1 contains a Human Reliability Analysis that is an excerpt from CNS PSA-ES091, "Detailed PSA Study of Triennial Fire Protection Inspection." Enclosure 2 contains NEDC 08-032, "EPM Calculation 1906-07-06 Task 7.6 NFPA 805 Fire Ignition Frequencies, Revision 0." Additional studies and calculations are still being developed and processed, and will be provided at the Regulatory Conference on May 13.

NPPD looks forward to meeting with your staff and appreciates the NRC's consideration of the information provided.

If you have questions, you may contact me at (402) 825-2774.

Sincerely,

David W. Van Der Kamp

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Licensing Manager

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Attachment

Enclosures

cc: U.S. Nuclear Regulatory Commission w/ attachment and enclosures
Document Control Desk

Cooper Project Manager w/ attachment and enclosures
USNRC – NRR Project Directorate IV-1

Senior Resident Inspector w/ attachment and enclosures
USNRC – CNS

NPG Distribution w/o attachment and enclosures

CNS Records w/ attachment and enclosures

# Attachment 1

# Information Summary

# Attachment 1
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On June 12, 2007, during the Cooper Nuclear Station (CNS) Triennial Fire Protection Inspection conducted by the Nuclear Regulatory Commission, a walk down of manual operator actions used for 10 CFR 50 Appendix R compliance, identified some actions could not be executed as written in the procedure. In order to evaluate the potential risk increase associated with the inadequate procedure guidance with best available information, the following new information is being provided in support of the significance determination process.

**Evaluation of Recovery**

The at-power Human Reliability Analysis (HRA) is applicable to diagnosis of loss of core cooling and recovery of core cooling following failure of Residual Heat Removal (RHR) valve RHR-MO-25B to open as directed by Procedure 5.4Fire-S/D, Fire Induced Shutdown From Outside Control Room, Revision 14. The basis for recovery is by either opening RHR-MO-25B manually or restoring High Pressure Coolant Injection (HPCI) for long term hot shutdown mitigation.

The Modular Accident Analysis Program (MAAP 4.0.5) and CNS specific model were used to analyze the plant response to determine the available time for recovery from a failure to open RHR-MO-25B, while transitioning to "Alternate Shutdown Cooling" alignment.

Thermal-hydraulic analysis was performed with initial success of HPCI and RHR Suppression Pool Cooling as directed by Procedure 5.4Fire-S/D from the Alternate Shutdown Panel (ASD). In order to determine time available for recovery, the limiting time interval used is based on a plant cool-down rate of 90°F/hr. Then upon entry to Section 3, the Preferred Method is used, where HPCI injection is secured independent of Reactor Pressure Vessel (RPV) level. The analysis conservatively assumed Safety Relief Valves (SRVs) were opened 3 hours following event initiation and HPCI injection was secured at the time SRVs were opened. The limiting time interval to the onset of core damage (1800°F) is more than 2 ½ hours after the SRVs are opened.

Due to the 2 ½ hours available, diagnosis and restoration of RPV level is considered viable. Using guidance contained in Procedure 5.4FIRE-S/D and engrained training associated with key RPV parameter maintenance (i.e. RPV water level, RPV pressure, and power); the expected human failure probability (HEP) for recovery of RPV injection is low. The combined non-recovery probability of either "Alternate Shutdown Cooling" mode of RHR by manual opening of RHR-MO-25B or restoration of level using HPCI injection is calculated to be less than 4.E-3.

**Conclusion**

The recovery of RPV injection at the ASD panel is considered likely. This is supported by a detailed HRA that reflects the time available, expected time of execution and diagnosis of lowering RPV water level that leads to recovery of core cooling. It is recommended that a non-recovery probability on the order of 4.E-03 be applied to these sequences.

The detailed HRA is provided as Enclosure 1 and is documented in Attachment 2 of CNS PSA-ES091, "Detailed PSA Study of Triennial Fire Protection Inspection."

**Evaluation of Shutdown Cooling and Other Assumptions**

The CNS SPAR model was used to evaluate the risk significance of the loss of RHR Shutdown Cooling (SDC). Sensitivity studies were performed for fire zones 3A, 3B, 2D, 7A, 8A, 8B, 8G, 8H, 9A, 9B and 10B. The baseline risk for specific source fires in each area was determined using the equipment identified in the CNS Fire Compartment Close-out Strategy document. In each case the delta conditional core damage probability (ΔCCDP) for loss of SDC is minimal, <1.E-05 (or zero). This confirms that the loss of SDC resulting from the subject inadequate procedure guidance is not risk significant. This evaluation supports our previous conclusion that cold shutdown state (as modeled with SDC) is not risk significant.

The inadequate procedure guidance related to RHR-MO-25B contained in Procedure 5.4POST-FIRE was confirmed to be limited to the operation of SDC mode of RHR. The risk significance as evaluated in CNS PSA-ES083 shows that failure to operate RHR-MO-25B to support SDC is very low. There is no reduction of CCDP related to loss of Low Pressure Coolant Injection (LPCI) function of RHR due to procedure errors when using Procedure 5.4POST-FIRE.

There are only two fire sources which adversely effect HPCI high level trip logic in fire zone 2A/2C, those are Motor Control Center (MCC) MCC-Y and ASD Panel. Further review of 2B and 2D reveal there are no cables related to HPCI high level trip logic in these zones and therefore no adverse impact to HPCI.

The failure to isolate valve RR-MO-53A is confirmed not to defeat the LPCI function of RHR for non-LOCA sequences. Therefore, these failures should not represent a LPCI failure state when evaluating the plant response to fire events.

The alternate shutdown cooling mode of operation at the ASD panels does not rely on valve RHR-MO-17. The RHR valves used to operate alternate shutdown cooling at the ASD room are listed in Procedure 5.4Fire-S/D and does not include RHR-MO-17.

**Fire Ignition Frequencies**

Revised fire ignition frequencies were calculated for each of the ten fire zones contributing to delta Core Damage Frequency using industry approved methods as specified in NUREG 6850. The fire ignition frequency results are listed below by fire zone and compared with the corresponding values listed in the CNS IPEEE. The screening of the FIVE methodology in the CNS IPEEE produces results that over-estimate the importance of fire areas evaluated for the subject condition. The CNS specific NUREG 6850 analysis results represent a detailed evaluation of source contributions within each area using best available techniques.

| CNS IPEEE Fire Area | CNS IPEEE Ignition Frequency | Fire PRA (NUREG 6850) Ignition Frequency NEDC 08-032 | IPEEE Multiplier |
|---|---|---|---|
| Area 1C, RHR Pump 1A and 1C Room | 2.94E-03 | 1.089E-03 | 0.3704 |
| Area 1D/1E, RHR Pump 1B and 1D, And HPCI Room | 4.28E-03 | 2.582E-03 | 0.6033 |
| Area 2B, RHR Heat Exchanger 1A Compartment | 6.70E-04 | 4.552E-04 | 0.6794 |
| Area 2D, RHR Heat Exchanger 1B Compartment | 6.70E-04 | 4.552E-04 | 0.6794 |
| Area 2A/2C, Reactor Building 903 | 2.32E-02 | 5.607E-03 | 0.2417 |
| Area 2A-1, Reactor Building 903 NE | 6.34E-03[1] | 1.432E-03 | 0.2259 |
| Area 8A, Auxiliary Relay Room | 4.02E-03 | 1.415E-03 | 0.3520 |
| Area 9A, Cable Expansion Room | 6.39E-03 | 4.267E-03 | 0.6678 |
| Area 9B, Cable Spreading Room | 6.89E-04 | 1.687E-04 | 0.2448 |
| Area 10B, Main Control Room | 1.09E-02 | 6.880E-03 | 0.6312 |

Notes: 1) Contribution to the IPEEE frequency is MCC K, Panel AA3, Panel BB3 and RCIC Starter Rack.

The above Fire Probablistic Risk Assessment (PRA) ignition values accurately reflect the sources in each defined fire area. These values are considered best estimate for purposes of Fire PRA modeling and meets the guidance of NUREG/CR-6850. Therefore, these values are provided for use in the significance determination of the subject condition. The detailed analysis is provided as Enclosure 2, NEDC 08-032, EPM Calculation 1906-07-06 Task 7.6 NFPA 805 Fire Ignition Frequencies, Revision 0 (refer to Table A-2, Fire Zone Ignition Frequencies).

**Fire Modeling**

Fire modeling is being performed for the ignition sources in the Control Room (CR), Cable Spreading Room, Cable Expansion Room, Auxiliary Relay Room and Reactor Building 903 level and documented in CNS calculations. These calculations will be used to determine CR abandonment frequencies. Calculation NEDC 08-041 is being performed to determine forced CR (Fire Zone 10B) abandonment due to habitability. NUREG/CR-6850, Task 11 provides criteria for habitability that is listed as follows:

1. The room or hot gas layer temperature reaches 200°F (93°C).

2. The heat flux to the CR floor is above 1 $kW/m^2$,

3. The hot gas layer is 6 ft (or lower) above the floor and has an optical density of 3.0 1/m. Such an optical density will prevent operators to see through smoke.

Fire Dynamics Simulator (FDS) fire model was used to determine the CR conditions. Preliminary results of the fire modeling indicate that the CR mechanical ventilation will not isolate when a fire starts in the CR. The reason for this is due to the unique design of the Control Room Ventilation. The Control Room ventilation system is shared with the Cable Spreading room. The Control Room ventilation system will isolate on the activation of smoke detector SD-1001 or thermal detectors located inside floor level fire dampers in the CR. Smoke detector SD-1001 is located in the cable spreading room in a return air inlet vent. The fire modeling results show that the SD-1001 does not detect enough smoke to cause the isolation of ventilation system. The contribution of a CR forced abandonment would be minimal with the current ventilation configuration.

Based on the above results, the dominant contribution to determine the probability of forced CR abandonment due to habitability would be a random ventilation system unavailability and limited fire scenarios that would cause a spurious isolation of the ventilation system. Preliminary results from NEDC 08-041 shows that this will result in a reduction in the CR forced abandonment frequency in the range 3.E-02 from that reported in NRC IR 2008007.

# Enclosure 1

# Human Reliability Analysis

# Enclosure 2

# EPM Calculation 1906-07-06 Task 7.6
# NFPA 805 Fire Ignition Frequencies

ATTACHMENT 3 LIST OF REGULATORY COMMITMENTS©

Correspondence Number: NLS2008044

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the Licensing Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

| COMMITMENT | COMMITMENT NUMBER | COMMITTED DATE OR OUTAGE |
|---|---|---|
| None | | |
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