Nebraska Public Power District
Nebraska's Energy Leader

NLS2000113
December 20, 2000

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

Subject: Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

References:

1) Letter to J. H. Swailes (NPPD) from Mohan C. Thadani (USNRC) dated December 1, 2000, "Supplemental Request for Additional Information (RAI) Regarding Individual Plant Examination for External Events (IPEEE) Fire Analysis (TAC No. M83611)"

2) Letter NLS2000078 to USNRC Document Control Desk from John H. Swailes dated September 22, 2000, "Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)"

The purpose of this letter is to submit the Nebraska Public Power District's (District's) response to the Supplemental Request for Additional Information (RAI) dated December 1, 2000 (Reference 1). The attached response documents the information provided during a teleconference between District representatives and the Nuclear Regulatory Commission (NRC) Nuclear Reactor Regulation Project Manager for Cooper Nuclear Station and the NRC Technical Reviewer representative on November 30, 2000. During this teleconference it was agreed that the District had provided sufficient information to address the questions asked during the teleconference and that the District need only document its teleconference responses in responding to the RAI.

It should be noted that Question 3 of the RAI (Reference 1) requested additional information that was not discussed during the teleconference. To provide the newly requested information, the District would need additional time beyond the agreed upon response date of December 21, 2000. In view of the short turnaround time, the District has determined that it is appropriate to provide the information discussed in the teleconference on the agreed schedule. This determination is based on the District's view that the bounding condition described in its response to RAI

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Question 3, and discussed during the teleconference, may preclude the need to provide a more detailed response to this question. Should the NRC determine that the additional information for RAI Question 3 is still needed, the District will provide this information at a later date.

Please find attached the individual RAI questions and the District's documented response. Should you have any questions concerning this matter, please contact Sharon Mahler at 402-825-5236.

Sincerely,

John H. Swailes
Vice President of Nuclear Energy

/dnm
Attachments

cc: Regional Administrator
USNRC - Region IV w/attachments

Senior Project Manager
USNRC - NRR Project Directorate IV-1 w/attachments

Senior Resident Inspector
USNRC w/attachments

NPG Distribution w/attachments

Records w/attachments

STATE OF NEBRASKA)
)
NEMAHA COUNTY)

John H. Swailes, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this response on behalf of Nebraska Public Power District; and the statements contained herein are true to the best of his knowledge and belief.

John H. Swailes

Subscribed in my presence and sworn to before me this 20th day of December, 2000.

LuAnn Bray
NOTARY PUBLIC

GENERAL NOTARY-State of Nebraska
LUANN BRAY
My Comm. Exp. May 11, 2002

Nebraska Public Power District's Response to Second Supplemental Request for Additional Information Related to Postulated Fire Events

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Individual Plant Examination for External Events for the Cooper Nuclear Station

The following is the Nebraska Public Power District's (District's) response to three questions contained in the Nuclear Regulatory Commission staff's (NRC's) Second Supplemental Request for Additional Information (RAI), (Reference 1), concerning certain postulated fire scenarios discussed in the Individual Plant Examination for External Events (IPEEE) for the Cooper Nuclear Station (CNS) (Reference 2). The NRC's questions were a result of its review of the District's RAI response provided on September 22, 2000 (Reference 3).

The text of the Second Supplemental RAI (SRAI) is presented below (*identified in italic letters*), and is followed by the District's response. References cited in the NRC's request / District's response are listed at the end of this attachment.

Supplemental RAI 1:

Provide the details on how the core damage frequency of 2 x 10^{-7} per reactor-year reported in the response to SRAI 1 was obtained. Include details about the fire frequency, suppression probabilities, any severity factors used, and the probability of failure to shut down from the alternate shutdown panel in the description. Compare the values and means used to those given in the revised EPRI guidance on main control room evacuation scenario evaluations.

District Response to Supplemental RAI 1

For clarification purposes the value of 2.0E-07 per reactor-year does not represent the core damage frequency (CDF) value itself, but rather represents an increase in the CDF above the CNS base value. This CDF increase was obtained from a sensitivity calculation, as explained below.

The event tree provided in Attachment 2 of this response shows the sequences for fires originating in panel 9-3 or board C in the Main Control Room (MCR). The sequences marked "a,""b," and "c" on the event tree are core damage sequences. Sequence "a" was included in the MCR fire analysis submitted with the original IPEEE (Reference 2). The CDF contribution from sequences "b" and "c," which are sequences for "fire not detected and suppressed within 15 minutes," amount to frequency f = (9.03E-08 + 1.00E-08) per year = 1.00E-07 per year. Thus, because this contribution may originate in either panel 9-3 or in board C, the total sensitivity contribution for this set of fire scenarios amounts to delta CDF = (2) (1.0E-07 per

year) = 2.0E-07 per year. This value represents essentially no change from the base CDF result for CNS.

Referring to the event tree, the initiator frequency f = 2.95E-04 per year is the value presented in the original IPEEE submittal (Reference 2) for a fire in a single MCR cabinet (Reference 2, Page 4-28, fire Zone 10B, "Control Room and SAS Corridor"). The conditional probability P = 3.40E-03 for "fire not detected and suppressed within 15 minutes" also is the same as in the original submittal for fire Zone 10B, "Control Room and SAS Corridor," (Reference 2, Page 4-28). The conditional probability of P = 1.00E-01 for "fire spreads to other panels" is a conservative screening value, based on guidance from Nuclear Safety Analysis Center, NSAC-181, "Fire PRA Requantification Studies," (Reference 4) which provides that fires originating in MCR panels generally do not propagate to other panels, but self-extinguish once the combustible material available within the panel of origin has been exhausted.

Whether the fire spreads to other panels or not, the CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," is invoked as applicable, prompting operators to evacuate the MCR and to shut down the plant from outside the MCR. Each of the core damage sequences "b" and "c" have a conditional probability of P = 1.00E-01 for "operators fail to shut down from outside of the main control room." This probability represents a conservative screening value, which envelopes both postulated equipment failure or human failure of the operators to implement the post-fire shutdown procedure, resulting in the core damage contributions marked on the attendant event tree branches.

The conditional probability value of P = 1.00E-01 is not affected by the revised Electric Power Research Institute (EPRI) guidance (Reference 5) on MCR evacuation scenario evaluations. The revised EPRI guidance has been reviewed, and no recommended numerical values for these specific scenario branches have been found. The value of P = 1.00E-01 is considered a "screening" value, (i.e., it is a conservatively enveloping value) assumed for the sake of analytical simplicity. If a more detailed analysis were performed, then smaller, best-estimate values would be utilized.

Supplemental RAI 2:

Page 4-27 of the Cooper IPEEE submittal indicates that failure to suppress a control room cabinet fire could result in the failure of circuits and controls in adjacent panels. Propagation of a fire in panel 9-3 to another panel in the control room containing controls related to offsite power could result in loss of offsite power. Without offsite power, the condensate and condensate booster pumps would not be operable. Table 4-3 of the Cooper individual plant examination submittal indicates that offsite power would be lost during fires in several panels (including panels BD-C and VBD-F). Indicate if fire propagation from panel 9-3 to adjacent panels could result in loss of offsite power or the loss of flow from the condensate and condensate booster pumps by other fire-induced failures and provide a quantitative evaluation of any identified propagation scenarios.

District Response to Supplemental RAI 2:

As stated in the above question, a fire in certain specific cabinets may cause offsite power to become unavailable. Thus, the total population of possible scenarios can be divided into two parts, namely scenarios with offsite power available and scenarios with offsite power not available.

For the majority of scenarios, offsite power is available. Therefore, CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," directs operators to leave one feedwater pump, one condensate booster pump, and one condensate pump running when evacuating the MCR. The bounding case for this scenario is where all eight safety relief valves (SRVs) go spontaneously fully open, based on postulated "smart" hot shorts. This causes maximum coolant loss from flashing. In this case, with the reactor depressurized, the feedwater pump is not available, while a condensate booster pump and condensate pump are available.

The layout of the CNS MCR shows the Emergency Core Cooling System (ECCS) equipment controls (panel 9-3) on the far left of the room, while the controls for maintaining offsite power (board C) are on the far right of the room. The controls for Condensate and Feedwater systems are nearly in the center, but somewhat closer to board C. For a fire in panel 9-3 to affect offsite power availability, it would have to propagate through four intervening panels. It would have to propagate through two intervening panels to affect the Feedwater and Condensate systems.

This fire propagation, if it were to occur, would take an appreciable amount of time, because the propagation would be in the horizontal rather than in the vertical direction. Therefore, it is reasonable to expect that for a fire in panel 9-3, the Feedwater and Condensate systems would remain unaffected for several minutes into the worst case scenario. For a fire in board C, it is reasonable to expect that the ECCS equipment would remain unaffected for several minutes into the worst case scenario.

Sensitivity calculations indicate that even with no coolant supply, the boiloff time to incipient fuel damage is about 29 minutes, and the time to localized fuel damage in the hot channel is about 40 minutes. If condensate is supplied for a duration of 60 seconds, the boiloff time is increased from about 29 minutes to about one hour. If condensate is supplied for five minutes, the boiloff time is increased from about 29 minutes to about two hours.

With the condenser hotwell being used for coolant makeup, permitted by CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," the water volume typically available in the condenser hotwell would last for about 1.1 hours, assuming the water level control loop is operating and all eight SRVs stay open. The reactor vessel would be kept near normal operating water level, from which a boiloff time to incipient fuel damage of 4.9 hours was calculated, after postulated depletion of condenser hotwell water inventory. The District concluded that the

calculated boiloff times provide more than sufficient time for the operators to initiate coolant injection from outside the MCR, even in cases where water from the condenser hotwell would not be available.

If offsite power is not available (e.g., the fire is initially in board C), the analysis simply amounts to retracing the 10CFR50 Appendix R analysis. In that analysis, loss of offsite power is postulated, so that one diesel generator must be started and loaded using the alternate shutdown procedure. It also assumes the spontaneous opening of one SRV concurrent with the initiator, even though the fire would need to propagate through several intervening panels to affect the SRVs.

As stated above, a fire originating from panel 9-3 would have to propagate through at least two panels in order to result in a loss of offsite power or loss of condensate. Any amount of time taken for this to occur provides significant additional time for implementation of the shutdown procedure, identified above.

Supplemental RAI 3:

The amount of water in the condenser hotwell is limited, as is the normal makeup capacity to the hotwell. Thus, the period in which condensate could be injected into the vessel would be limited. Provide the sequence of events (including operator actions) and the timing of the events. Specify the time available for the operators to establish LPCI flow from the alternate shutdown panel considering this limited capacity and assuming opening of all eight SRVs.

Begin the sequence of events where all of panel 9-3 is engulfed in fire and the operators decide to abandon the control room. Address the impact of different numbers of spuriously opened SRVs, the time at which they occur in relationship to the time the control room is abandoned, and any delay in abandoning the control room has on the provided sequence of events and its timing. Justify the human error probability used in the sensitivity evaluation for failure to control the reactor from the alternate shutdown panel in light of the identified timing of events.

District Response to Supplemental RAI 3:

As stated in the above question, the amount of coolant available from the condenser hotwell is limited, as is the makeup capacity to the condenser hotwell. As discussed in the response to Second SRAI 2, however even without any coolant supply after spontaneous opening of all eight SRVs, the boiloff time to incipient fuel damage is about 29 minutes, with localized fuel damage occurring at about 40 minutes. Additionally, sensitivity calculations indicate that the boiloff time is increased to about one hour, if condensate is supplied for a duration of 60 seconds, and this time is increased to about two hours if condensate is supplied for five minutes.

The District assumes an estimated time of ten minutes for the operators to evacuate the MCR and start taking actions in the alternate shutdown room. The operator actions necessary to establish

low pressure or high pressure coolant flow at the alternate shutdown panel require only positioning of valve control switches. To preclude propagation of postulated electrical faults, CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," directs that specific fuses be removed and specific breakers be opened. No disconnecting and reconnecting of cables are required to establish coolant flow. Installation of jumpers, using prearranged connection points and wires with quick-connect clips, is required if relief valves are to be opened from the alternate shutdown panel.

The actions identified in the CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," can be accomplished well within the time limits for the boiloff scenarios discussed. Additionally, because the amount of liquid flashed into steam is maximized for the case of eight open SRVs, this case is enveloping in terms of minimizing the amount of coolant available. Therefore, cases with fewer than eight SRVs opening are not considered to be limiting.

A more detailed description of the actions directed by CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," is provided below in order to generally address sequence of events following MCR abandonment. One of the actions taken by the operators following MCR evacuation is to de-energize the SRV control circuit. This will cause the SRVs to close, terminating the loss of coolant and causing the reactor to re-pressurize, so that the High Pressure Coolant Injection (HPCI) system can be operated.

Once the operators arrive at the alternate shutdown panel, in accordance with CNS procedure, "Post-Fire Shutdown to Mode 4 Outside Control Room," they will confirm either: (a) they do have sufficient reactor pressure to run the HPCI System, or (b) they do not have sufficient reactor pressure to run the HPCI system, in which case they initiate a low pressure coolant injection. Both cases are accounted for in the procedure.

With regard to the portion of the NRC question requesting the justification of the human error probability used in the sensitivity evaluation, please refer to the District's response to Second SRAI 1, above.

<u>References Cited in the NRC's Request</u>

-- Letter NLS2000078 to USNRC Document Control Desk from John H. Swailes dated September 22, 2000, "Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)"

-- "Guidance for Development of Response to Generic Request for Additional Information on Fire Individual Plant Examination for External Events (IPEEE)," prepared by Data Systems & Solutions, LLC, Final Report, May 1999

-- Letter NLS960143 to USNRC Document Control Desk from G. R. Horn (NPPD), dated October 30, 1996, "Individual Plant Examination for External Events (IPEEE) Report - 10 CFR 50.54(f)"

References Cited in the District's Response

1. Letter to J. H. Swailes (NPPD) from Mohan C. Thadani (USNRC) dated December 1, 2000, "Supplemental Request for Additional Information (RAI) Regarding Individual Plant Examination for External Events (IPEEE) Fire Analysis (TAC No. M83611)"

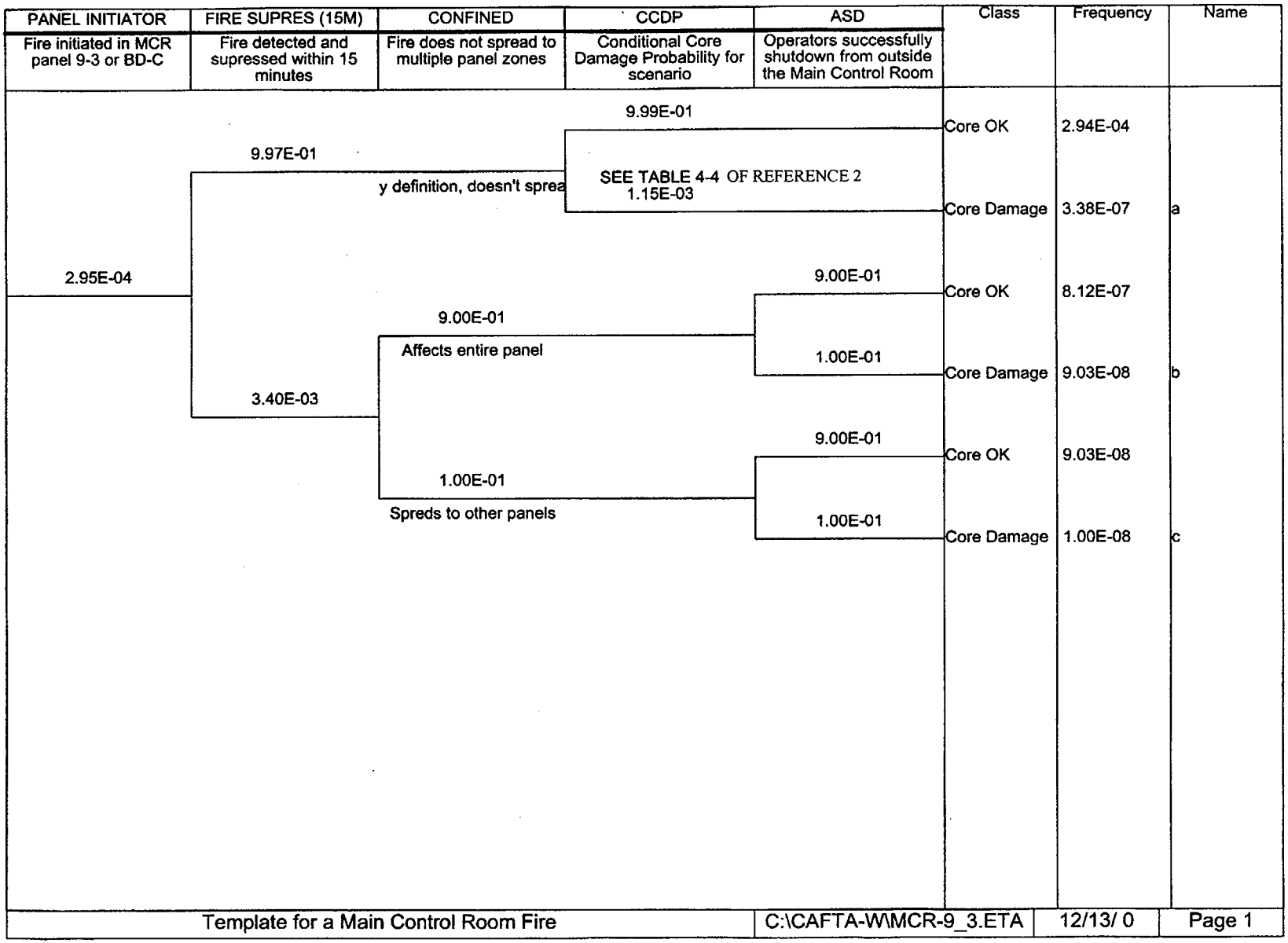
2. Letter NLS960143 to USNRC Document Control Desk from G. R. Horn (NPPD), dated October 30, 1996, "Individual Plant Examination for External Events (IPEEE) Report - 10 CFR 50.54(f)"

3. Letter NLS2000078 to USNRC Document Control Desk from John H. Swailes dated September 22, 2000, "Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)"

4. NSAC-181 "Fire PRA Requantification Studies," prepared by Science Applications International Corporation, Final Report, March 1993

5. "Guidance for Development of Response to Generic Request for Additional Information on Fire Individual Plant Examination for External Events (IPEEE)," prepared by Data Systems & Solutions, LLC, Final Report, May 1999

6. Letter LQA8300256 to Director of Nuclear Reactor Regulation (USNRC) from Jay M. Pilant (NPPD), dated December 2, 1983, "Response to 10CFR50, Appendix R, Fire Protection of Safe Shutdown Capability - Volume III"


PANEL INITIATOR
Fire initiated in MCR panel 9-3 or BD-C
FIRE SUPRES (15M)
Fire detected and supressed within 15 minutes
CONFINED
Fire does not spread to multiple panel zones
CCDP
Conditional Core Damage Probability for scenario
ASD
Operators successfully shutdown from outside the Main Control Room
Class
Frequency
Name
2.95E-04
9.97E-01
3.40E-03
y definition, doesn't sprea
9.00E-01
Affects entire panel
1.00E-01
Spreds to other panels
9.99E-01
SEE TABLE 4-4 OF REFERENCE 2
1.15E-03
9.00E-01
1.00E-01
9.00E-01
1.00E-01
Core OK
2.94E-04
Core Damage
3.38E-07
a
Core OK
8.12E-07
Core Damage
9.03E-08
b
Core OK
9.03E-08
Core Damage
1.00E-08
c
Template for a Main Control Room Fire
C:\CAFTA-W\MCR-9_3.ETA
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Page 1

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The following table identifies those actions committed to by the District in this document. Any other actions discussed in the submittal represent intended or planned actions by the District. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	N/A