Nebraska Public Power District
Nebraska's Energy Leader

NLS2000078
September 22, 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 Lawrence J. Burkhart (USNRC) dated March 1, 2000, "Supplemental Request for Additional Information Regarding IPEEE Fire Analysis (TAC No. M83611)"

2) Letter NLS2000058 to USNRC Document Control Desk from John H. Swailes dated June 29, 2000, "Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)"

The purpose of this letter is to submit to the Nuclear Regulatory Commission (NRC) the Nebraska Public Power District's (District's) response to the Supplemental Request for Additional Information (RAI) dated March 1, 2000 (Reference 1). Attachment 1 addresses RAI Questions 1 and 2. The District's response to Question 3 was previously provided per Reference 2. However, the question and the District's response are included in Attachment 2 for information.

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

A011

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cc: Regional Administrator
USNRC - Region IV w/attachments

Senior Project Manager
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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 22nd day of September, 2000.

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

LuAnn Bray
NOTARY PUBLIC

NPPD Response to the NRC Staff's Supplemental Request for Additional Information Concerning Postulated Fire Scenarios Evaluated In the Individual Plant Examination for External Events for the Cooper Nuclear Station

The following is the Nebraska Public Power District's (District's) response to Questions 1 and 2 contained in the Nuclear Regulatory Commission's (NRC's) 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 District's response to Question 3 was previously submitted to the NRC per Reference 3. This question and the District's response are included in Attachment 2 for information.

The text of the Supplemental RAI 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:

Additional information addressing the concerns cited in question A.13 (see original RAI dated June 3, 1998, and the NPPD response dated January 28, 1999) should include consideration of the concerns raised in the original RAI questions A.2 and A.3. Additional information regarding the impact of fires in Main Control Room (MCR) "control panel 9-3" and MCR "Board C" is requested.

The fire assessment of the control room indicates that the analysis of fires in two panels were treated uniquely. Control panel 9-3 has no internal barriers and it was postulated that fires would only impact portions of the panel. Board C contains partial internal barriers that were assumed effective in preventing fire propagation from one section to another. Crediting partial barriers and separation for inhibiting fire growth is questionable and resulted in RAI question A.13.

The response to RAI question A.13 assumed that fires will remain limited to only a sub-section of each panel, per the original analysis. While the most likely fires may involve only a subsection of the panel, a less likely, but potentially more risk-significant fire involving the full panel, cannot be dismissed. The objective of this supplemental RAI is to obtain an assessment of the potential risk implications of fires involving these two panels that considers the impact of more severe fires than those identified thus far.

In reexamining scenarios associated with these panels, the licensee should also address the specific concerns identified in the original RAI question A.3 - the treatment of control systems interactions. That is, in assessing the core damage frequency (CDF) contribution of fires in

these two MCR panels, the licensee should verify that (1) the alternate shutdown capability is electrically independent of the MCR assuming a severe fire engulfs either of these two panels, (2) the loss of equipment before transfer will not compromise the operator's ability to control the plant from outside of the control room, and (3) spurious actuations that might result from a severe fire in either of the two cited panels will not compromise the safe shutdown capability. If these cannot be verified, they should be considered in the reanalysis discussed below.

In quantifying the risk contribution, use of a conditional probability of control room abandonment of 3.4E-3 given a fire is acceptable practice (per the Electric Power Research Institute (EPRI) revised guidance [1] associated with resolution of generic questions arising from the EPRI Fire PRA Implementation Guide). However, the response to RAI question A.2 indicated that an additional factor of 0.1 was applied to reflect the fraction of control room fires considered to be severe fires (sufficient to impact control room visibility). The application of this factor for general control room abandonment scenarios is inappropriate because the 3.4E-3 conditional probability value already reflects the probability that a small fire will remain unsuppressed and will grow into a severe fire. Using additional severity factors in combination with the 3.4E-3 conditional abandonment probability in quantifying the fire-induced CDF for these two panels is not appropriate. This assessment should include the human actions required for safe plant shut down in the event the postulated fires occur.

The requantification of these scenarios, if required based on the results of the reexamination, should be based on the following factors: (1) the overall frequency of control room fires (values are cited in the fire-induced vulnerability evaluation (FIVE) methodology), (2) partitioning of the overall fire frequency to the specific panels of interest (guidance is available in the FIVE methodology), (3) a conditional probability of control room abandonment of 3.4E-3 (per the revised EPRI guidance), and (4) the reliability of the capability to shutdown the plant from outside of the control room including consideration of the control systems interactions concerns and human action assessment discussed above. Other factors should not be included in the requantification unless their applicability is fully justified.

(A) Please provide the results of a reexamination and a requantification, if necessary, of the CDF contribution for fires involving either the MCR "control panel 9-3" and MCR "Board C."

(B) Please provide the results of a reexamination of the fire risk associated with fires in these panels assuming that a fire might propagate to involve the entire panel.

(C) Please provide a detailed description of the functions, instrumentation, and control capabilities that are provided outside of the control room.

(D) Please provide the results of an assessment of the features identified in (C) above for adequacy and reliability in the specific context of a severe fire in either of the two cited panels and specify identified vulnerabilities, if any.

District Response to Supplemental RAI 1

The District's response to Supplemental RAI 1 consists of separating Supplemental RAI 1 into two parts. Part 1 addresses the three verification concerns expressed in the Supplemental RAI. The applicable portion is restated below in italics.

Part 2 addresses the use of the additional "0.1" factor described in the paragraph directly below the verification paragraph. The applicable portion is also restated below in italics. The District's response provides additional detail to demonstrate that the original submittal for fire compartment 10B (main control room (MCR) and security alarm station corridor) is sufficient, for the purpose of screening out fires in the MCR and subsequent shutdown from outside the MCR as potential vulnerabilities.

In regard to the portion of Supplemental RAI 1 which identifies four specific requests (A) through (D), the District has addressed the requests in the discussion below. Requests (A) and (B) are addressed in the discussion pertaining to Part 2. The information supporting request (C) is provided at the end of the discussion for Part 2. Request (D) is addressed in the discussion pertaining to Part 1.

The District has concluded, based on the reexamination discussed below, that no requantification is necessary for the fire compartment 10B (specifically MCR "control panel 9-3" and MCR "board C").

District Response to Supplemental RAI 1, Part 1:

In reexamining scenarios associated with these panels, the licensee should ... verify that (1) the alternate shutdown capability is electrically independent of the MCR assuming a severe fire engulfs either of these two panels, (2) the loss of equipment before transfer will not compromise the operator's ability to control the plant from outside of the control room, and (3) spurious actuations that might result from a severe fire in either of the two cited panels will not compromise the safe shutdown capability. If these cannot be verified, they should be considered in the re-analysis discussed below.

A reexamination of the CNS 10CFR50, Appendix R (Appendix R) related documentation shows that the alternate shutdown capability is both electrically and physically independent of the MCR for any fire in the MCR [item 1]. This independence is also verified by periodic surveillance testing. The Appendix R related documentation also supports that shutdown from outside the MCR can be accomplished even if the controls in the MCR were disabled by the fire [item 2]. Finally, the Appendix R documentation addresses spurious operation of equipment due to MCR fires and concludes that any single spurious operation will not compromise alternate shutdown capability. These items are supported by the CNS Safety Evaluation Report (SER) (Reference 4), issued April 16, 1984, concerning compliance to Items III.G.3 and III.L of Appendix R.

Appendix R did not cover multiple spurious actuations caused by the fire in the MCR [item 3]. In the case of panel 9-3, a worst case assumption can be made that all eight of the Safety Relief Valves (SRVs) spuriously open, causing Main Steam Isolation Valves to close (loss of condenser), low steam pressure (loss of High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling), and rapid loss of inventory. The low pressure Emergency Core Cooling System (ECCS) controls (Low Pressure Cooling Injection (LPCI) and Core Spray) are also assumed to be failed due to the postulated fire.

In this case, alternate shutdown can still be accomplished with equipment and controls located outside the MCR using manual control of LPCI. The operators will be afforded the time to do this because the alternate shutdown procedure directs the operators to leave one condensate booster pump and one condensate pump running when evacuating the MCR. The hotwell provides sufficient low pressure injection coolant to give the operators time to take manual control of LPCI. If the fire is in any other MCR panel, the resulting multiple spurious actuations would be much less challenging. Therefore, item 3 is also satisfied.

<u>District Response to Supplemental RAI 1, Part 2:</u>

In quantifying the risk contribution, use of a conditional probability of control room abandonment of 3.4E-3 given a fire is acceptable practice (per the Electric Power Research Institute (EPRI) revised guidance [1] associated with resolution of generic questions arising from the EPRI Fire PRA Implementation Guide). However, the response to RAI question A.2 indicated that an additional factor of 0.1 was applied to reflect the fraction of control room fires considered to be severe fires (sufficient to impact control room visibility). The application of this factor for general control room abandonment scenarios is inappropriate because the 3.4E-3 conditional probability value already reflects the probability that a small fire will remain unsuppressed and will grow into a severe fire.

In the original IPEEE submittal (Reference 2) for the MCR fire scenario, it was not intended that the factor of 0.1, applied to "severe fires," be interpreted as an additional severity factor for those fires that were not extinguished within 15 minutes. The 0.1 factor used in the IPEEE submittal reflects the Nuclear Safety Analysis Center, NSAC-181 (Reference 5) guidance that nearly all fires originating within the MCR panels do not propagate to other panels.

Based on this guidance, it is assumed in the CNS IPEEE that MCR panel fires, which are not extinguished within 15 minutes, have a 10 percent probability of propagating through barriers (sometimes multiple barriers) and affecting panels containing safe shutdown equipment. Thus, this factor represents a "conditional propagation factor." This 10 percent value for the propagation factor was considered appropriate because the majority of MCR panels are not located near panels containing safe shutdown equipment.

The topic being addressed in this Supplemental RAI question involves fires that originate within the two panels (namely panel 9-3 or board C) that were apportioned differently than the approach

suggested by the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) methodology (Reference 7) utilized for the CNS IPEEE. In these cases, a fire that is not promptly extinguished may potentially have a greater than 10 percent probability for affecting the entire panel.

For demonstration purposes, the District performed a sensitivity evaluation for fires initiated in Panel 9-3 or board C that were not extinguished within 15 minutes, i.e., fire engulfing the entire panel. If all of these scenarios become severe fires that require shutdown from outside the MCR, and the previously reported screening probability of alternate shutdown failure probability is used, the maximum increase in core damage frequency (CDF) due to fires in these panels would be no more than $2x10^{-7}$ per year. This represents essentially no change in the results reported in the IPEEE submittal (Reference 2). No additional vulnerabilities or insights were identified.

(C) Please provide a detailed description of the functions, instrumentation, and control capabilities that are provided outside of the control room.

The alternate shutdown capability at CNS consists of circuit isolation devices, remote and local controls, and instrumentation for the components necessary to achieve cold shutdown in a postulated fire scenario. Centralized remote controls are available on the three control panels in the alternate shutdown (ASD) room. Local controls for diesel generator 2 and for manual operation of selected valves and circuit breakers are located in their respective locations.

The alternate shutdown capability has been designed to replace the redundant safe shutdown functions which could be lost to a fire in either the MCR, cable spreading room (CSR), cable expansion room, auxiliary relay room, or the 903' elevation of the Northeast corner of the reactor building. The three control panels in the alternate shutdown room are: (1) HPCI system, (2) automatic depressurization system (ADS)/ reactor equipment cooling (REC) system loop B (electrical division 2), and (3) residual heat removal (RHR) system loop B (electrical division 2).

Although the shutdown systems are limited to one electrical division, redundancy is provided in that high pressure and low pressure cooling are both available. The HPCI system is available, three power operated relief valves, REC pumps C and D, RHR heat exchanger B, RHR pumps B and D [RHR loop B] are also available at the alternate shutdown room panel. The RHR loop can be used in the suppression pool cooling mode and in the shutdown cooling mode.

Instrumentation on the HPCI board consists of indicators for turbine speed and steam inlet pressure, pump suction pressure, discharge pressure and flow, suppression chamber water level, reactor water level, reactor shroud level, and water level in emergency condensate storage tank 1B. RHR instrumentation consists of indication of loop B flow. The ADS board has four temperature indicators, each for a different torus region.

Additional details regarding functions, instrumentation, and control capabilities provided outside the MCR can be found in the District's December 2, 1983, submittal (Reference 8) utilized to support the NRC's issuance of the SER (Reference 4), concerning compliance to Items III.G.3 and III.L of Appendix R.

Supplemental RAI 2:

Additional information is needed with respect to the response to the original RAI question A.5. The fire analysis for the cable spreading room (CSR) apparently assumed that fires in closed electrical panels cannot escape the panel of origin. Further, the response to the original RAI question A.5 would also be impacted by the concerns identified in the original RAI question A.4 (cabinet fire heat release rates) and the original RAI question A.10 (cable thermal damage limits).

Assessments of fire propagation from closed electrical cabinets should include consideration of the revised EPRI guidance [1]. In particular, EPRI has provided revised guidance in the areas of electrical panel fires that might escape from the panels, and in the selection of cabinet fire heat release rates. Both factors will impact the assessment of CSR fires for CNS. The current assessments are not consistent with the revised guidance. The assessment should include consideration of the concerns raised in Supplemental RAI 3 below (relating to cable thermal damage limits) in the response, namely the use of appropriate cable thermal damage limits.

(A) Please provide the results of a reassessment of the CDF contribution of fires in the CSR, including electrical panel fires.

District Response to Supplemental RAI 2:

The revised EPRI guidance (Reference 6) referred to in Supplemental RAI 2, recommends in Section 3.11 that certain electrical cabinets not be screened out solely on the basis that they are fully enclosed. The guidance suggests that such cabinets be treated as if they were vented. Additional details are provided in Section 4.11 of the revised EPRI guidance.

This guidance addresses a case where a high-energy cabinet (i.e., switchgear cabinet or motor control center) has been screened out solely on the basis of being fully enclosed, disregarding the fact that short circuits above a certain energy level may have explosive type consequences with potential consequential damage to other cabinets in the immediate vicinity.

The District has verified, through walkdowns and documentation review, that the CSR does not contain any switchgear cabinets or motor control centers. Electrical cabinets in the CSR are low-power cabinets, such as control cabinets containing relays, signal processors and other components typically related to instrumentation functions in the plant. Consequently, the concern regarding high-energy short circuits is not applicable to cabinets in the CSR at CNS.

The concern regarding thermal damage limit ratings of qualified cable has been resolved in a prior submittal, by the District, in response to Supplemental RAI question 3, dated June 29, 2000 (Reference 3). Consequently, no reassessment has been performed. The District has not identified any additional potential vulnerability during the walkdown or in the course of the documentation review activities performed for this Supplemental RAI.

References Cited in the NRC's Request

-- Letter to G. R. Horn (NPPD) from James R. Hall (USNRC), dated June 3, 1998, "Request for Additional Information Related to the Individual Plant Examination of External Events (IPEEE) for the Cooper Nuclear Station (TAC No. M83611)"

-- Letter NLS990008 to USNRC Document Control Desk from John H. Swailes (NPPD), dated January 28, 1999, "Response to 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

-- EPRI TR-100370 "Fire-Induced Vulnerability Evaluation (FIVE)," prepared by Professional Loss Control, Final Report, April 1992

References Cited in the District's Response

1. Letter to J. H. Swailes (NPPD) from Lawrence J. Burkhart (USNRC), dated March 1, 2000, "Cooper Nuclear Station - Supplemental Request for Additional Information Regarding 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 NLS2000058 to USNRC Document Control Desk from John H. Swailes (NPPD), dated June 29, 2000, "Response to Supplemental Request for Additional Information - Individual Plant Examination for External Events (IPEEE)"

4. Letter to J. M. Pilant (NPPD) from Domenic R Vassallo (USNRC), "Safety Evaluation for Appendix R to 10 CFR Part 50, Items III.G.3 and III.L, Alternate or Dedicated Shutdown Capability"

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

6. "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

7. EPRI TR-100370 "Fire-Induced Vulnerability Evaluation (FIVE)," prepared by Professional Loss Control, Final Report, April 1992

8. 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"

Attachment 2 to
NLS2000078
5 Pages Total

Nebraska Public Power District's Response to NRC Staff's Supplemental Request for Additional Information Question 3.

This information was forwarded as an attachment (4 Pages Total) to Letter No. NLS2000058 to USNRC Document Control Desk from John H. Swailes (NPPD), dated June 29, 2000, (Reference 3).

Nebraska Public Power District's Response to the NRC Staff's Supplemental Request for Additional Information Concerning Postulated Fire Scenarios Evaluated in the Individual Plant Examination for External Events for the Cooper Nuclear Station

The following is the Nebraska Public Power District's (District's) response to Question 3 contained in the Nuclear Regulatory Commission's (NRC's) Supplemental Request for Additional Information (RAI), dated March 1, 2000 (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). References cited in the NRC's request and/or in the District's response are listed at the end. As per discussions between the NRC Nuclear Reactor Regulation Project Manager for CNS and the CNS Assistant Licensing Manager, the individual responses to RAI Questions 1 and 2 are still being developed. The District plans to submit the responses to the remaining questions no later than September 22, 2000.

The CNS IPEEE (Reference 2) was submitted to the NRC on October 30, 1996, in response to Generic Letter 88-20, Supplement 4 (Reference 3). The CNS IPEEE contains a systematic probabilistic assessment of potential plant vulnerabilities to postulated external events, including but not limited to internal fire scenarios, utilizing the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) methodology (Reference 4).

Supplemental RAI 3:

Additional information is needed with respect to the response to the original RAI question A.10. The concern is associated with the assumption that the cables at CNS are as good or better than IEEE-383 qualified cables. The licensee cited previous tests and NRC licensing documents as supporting this assumption. A review of the licensing documents indicates that the only tests performed were associated with cable flammability. The IEEE-383 standard includes both a flammability test and aging and thermal performance tests associated with loss-of-coolant accidents. The flammability tests are acceptable in the context of dismissing self-ignited cable fires (this was one part of the question). However, flammability tests alone do not demonstrate that a given cable has the same thermal damage thresholds that one normally associates with fully qualified cables (this was the second part of the question).

The selection of thermal damage temperatures should consider the specific cable insulation materials associated with the cables at CNS or should bound the lower limit of thermal damage for unqualified cables.

(A) It is not clear that cables are qualified equivalent to IEEE-383 (including thermal damage threshold criteria). Demonstrate that cables are qualified to a standard equivalent to IEEE-383 or provide the results of an assessment of the impact of using

thermal damage limits associated with unqualified cables (i.e., lower damage threshold than used in the original assessment) on quantitative screening of fire areas.

(B) If any damage scenarios in quantitatively screened compartments are impacted, please provide a reassessment of the CDF contribution for those compartments.

District Response to Supplemental RAI 3:

In the District's response to Question A.10 of the original RAI (Reference 5), the District established a general position that qualification of Class 1E cables, as described in the CNS IPEEE submittal, is equivalent to IEEE-383 (Reference 6) qualified cables. The information below provides additional detail as to how this position was reached along with information that demonstrates the qualification status of these cables. In summary, IEEE-383 equivalency at CNS is based on the flammability tests and on aging and post accident performance tests.

In order to demonstrate full equivalency to IEEE-383, 1974, it may be stated that a cable is an "IEEE-383 qualified cable" if it meets the acceptance criteria of the flammability test, the accelerated thermal and radiation aging test, and the harsh environment test [LOCA simulation with specified environmental parameter profiles], in accordance with accepted test practices as standardized in IEEE-383.

Regarding the flammability threshold part of the original RAI Question A.10, the response previously provided (Reference 7) is restated here, for completeness. It states, in part: "Cooper Nuclear Station was constructed before the Standard "IEEE-383, 1974" had become a consensus standard. In accordance with Branch Technical Position 9.5-1, Appendix A, CNS submitted a letter on December 17, 1976, which detailed the specifications and testing performed on cable types present in safety related areas of CNS, in order to demonstrate the technical basis for equivalency to IEEE-383, 1974. This letter was accepted by the NRC in the Safety Evaluation Report for Fire Protection, associated with Technical Specification Amendment 56, dated May 23, 1979. Thus, the established equivalency to IEEE-383, 1974 was the basis for utilizing the qualified cable data in the IPEEE fire analysis."

Additionally, the NRC Safety Evaluation Report associated with License Amendment 56 (Reference 8) states, on Page 4-8, in Section 4.8, in part: "Flame tests conducted on the electrical cables at Cooper plant were comparable to the combustibility tests set forth in IEEE-383 . . . Accordingly, we find the electrical cables used at the Cooper plant acceptable." This addresses the aspect of qualification with regard to a minimum demonstrated flammability threshold and confirms the equivalency of flammability tests performed on CNS cables to those prescribed by IEEE-383, 1974.

To demonstrate equivalency to IEEE-383, 1974 with respect to aging and post accident tests, CNS cables have equivalent qualification to IEEE-323, 1971, which is the parent to IEEE-383. IEEE-323, 1971 describes the basic requirements for the qualification of Class I electrical

equipment (Section 1). The qualification requirements include confirmation of the adequacy of the equipment design under normal service conditions (Section 4.3) and special conditions, such as large signals, extreme power supply voltages, fire, water, seismic forces, radiation, chemical sprays, etc. The standard requires the range, sequence, and combinations of environment to simulate the design basis event conditions (Sections 5.2.3.5 and 5.2.3.6). IEEE-383 addresses "significant environmental conditions" specifically for Class 1E cable, such as "1.3.3.1 Atmosphere" (moisture content, temperature, pressure, etc.); "1.3.3.2 Radiation" (normal dosage, design basis event dose rate, total design basis event dosage); "1.3.3.3 Chemicals" (e.g., type, spray, temperature of exposure). At the time of the cable purchases, equivalent qualification to IEEE-323, 1971 was accomplished by exposing test specimens to thermal aging, radiation aging (40-year equivalency of normal radiation and thermal conditions, plus post accident radiation exposure) and exposure to LOCA test profiles, which included steam environment, increased pressures, chemical spray, and elevated temperatures. This addresses the aspect of qualification with regard to thermal and radiation aging and harsh environment tests performed on CNS cables to those prescribed by IEEE-383, 1974.

Thus, based on the equivalency of flammability tests as discussed above, and on the tests for harsh environment and aging, it is concluded that the essential cables at CNS are de facto "IEEE-383 qualified cables."

Regarding Question 3, Part (A), the thermal damage threshold value of 700^0 F [371^0 C], per the EPRI FIVE Methodology as being applicable to "IEEE-383 qualified cables," has been assigned correctly to the Class 1E cables at CNS. Regarding Question 3, Part (B), based on there being no impacted damage scenarios in quantitatively screened compartments, no reassessment for Core Damage Frequency contribution, based on the thermal damage threshold value, is required.

<u>References</u>

1. Letter to J. H. Swailes (NPPD) from Lawrence J Burkhart (USNRC), dated March 1, 2000, "Cooper Nuclear Station - Supplemental Request for Additional Information Regarding IPEEE Fire Analysis (TAC No. M83611)"

2. Letter NLS960143 to USNRC Document Control Desk from G. R. Horn (NPPD), dated October 30, 1996, <u>CNS PSA - IPEEE</u> CNS Probabilistic Safety Assessment - "Individual Plant Examination for External Events"

3. Generic Letter No. 88-20, Supplement 4, dated June 28, 1991, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)"

4. <u>EPRI TR-100370</u> "Fire-Induced Vulnerability Evaluation (FIVE)," prepared by Professional Loss Control of Vernon Hills, Illinois, for the Electric Power Research Institute in Palo Alto, California. Final Report, April 1992

5. Letter to G. R. Horn (NPPD) from James R. Hall (USNRC), dated June 3, 1998, "Request for Additional Information Related to the Individual Plant Examination of External Events (IPEEE) for the Cooper Nuclear Station (TAC No. M83611)"

6. IEEE Std 383-1974: "IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations". The Institute of Electrical and Electronics Engineers, New York City, New York, February 1974.

7. Letter NLS990008 to USNRC Document Control Desk from John H. Swailes (NPPD), dated January 28, 1999, "Response to Request for Additional Information - Individual Plant Examination for External Events (IPEEE) - Cooper Nuclear Station"

8. Letter to NPPD from USNRC dated May 23, 1979, "Fire Protection Safety Evaluation Report in the Matter of Nebraska Public Power District, Cooper Nuclear Station - Unit 1, Docket No. 50-298"

9. IEEE Std 323-1971: "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations". The Institute of Electrical and Electronics Engineers, New York City, New York, 1971

Correspondence Number: NLS2000078

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