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May 7, 1984

Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, D.C. 20555

Prairie Island Nuclear Generating Plant
Docket No. 50-282 License No. DPR-42
Docket No. 50-306 License No. DPR-60

Resolution of Safety Evaluation Report for Environmental Qualification of Safety-Related Electrical Equipment

Gentlemen:

On April 13, 1984, a conference telephone discussion was held between members of the NRC Staff and NSP to discuss questions the Staff had in order to complete the SER on Environmental Qualification of Safety-Related Electrical Equipment for Prairie Island Nuclear Generating Plant, Units 1 and 2. The discussion specifically concerned our January 16, 1984 submittal of meeting minutes of the December 1, 1983, NRC/NSP meeting. The December 1 meeting was held to resolve qualification deficiencies identified in the SER and TER received by NSP on April 25, 1983.

This letter addresses the Staff's questions and supplements our January 16, 1984 submittal.

The following additional information was requested by the NRC Staff to supplement the NSP January 16, 1984 submittal:

Item

1. The methodology for compliance with 10CFR50.49(b)(2) must be expanded to address the 4 items in the Point Beach example provided by Point Beach.

2. All JCO's which are currently being used including any (b)(3) [R.G. 1.97 Category 1 or 2] equipment which is installed and operational and not qualified must be provided.

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3. JCO's that rely on fracture mechanics are not acceptable without statement that continuous temperature monitoring can be achieved. The NRC Staff has requested that this JCO be clarified to identify leak detection methods and provide an updated schedule for completion.

Our response to these items is as follows:

Item 1

Enclosure I is our methodology for compliance with 10CFR50.49(b)(2). This description clarifies the items discussed on April 13, 1984.

Item 2

Enclosure II updates each JCO provided in our January 16, 1984 response. Note that items 2 and 3 of Enclosure II are now fully qualified. The JCO's provided in the January 16, 1984 response and updated by Enclosure II include (b)(3) [R.G. 1.97 Category 1 or 2] equipment which is installed and operational and not fully qualified. For each item, the expected schedule for completion is also provided.

Item 3

The specific JCO utilizing a fracture mechanics analysis is contained in Item 6 of Enclosure II. This item pertains to the relocation of MCC's for the steam supply motor valves to the Turbine Driven AFW Pumps. This JCO has been clarified to address leak detection methods and schedule for qualification.

During the discussion we indicated that the Auxiliary Building HELB environmental profile had been re-evaluated. The result was a somewhat higher peak temperature. The revised profile is included in Enclosure III. We have re-evaluated the impact of the higher peak temperature on equipment and have concluded that all equipment is qualified except as noted in Enclosure II.

Please contact us if you require any additional information.

David Musolf
Manager
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cc: Regional Administrator-III, NRC
NRR Project Manager, NRC
Resident Inspector, NRC
G. Charnoff

Attachments

# ENCLOSURE I

## METHODOLOGY TO IDENTIFY EQUIPMENT WITHIN THE SCOPE OF 10CFR50.49(b)(2) PRAIRIE ISLAND NUCLEAR PLANT, UNITS 1 & 2

Paragraph (b)(2) of 10CFR50.49 requires that licensees identify non-safety related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in paragraph (b)(1). The methodology used to identify such equipment is summarized below:

1. A list was generated of safety-related electric equipment as defined in paragraph (b)(1) of 10CFR50.49 required to remain functional during or following design-basis Loss of Coolant Accident (LOCA) or High Energy Line Break (HELB) Accidents. The equipment was identified through a review of the accident analysis provided in the PINGP FSAR, a review of the PINGP emergency procedures, a review of safety system flow diagrams and Q-List, and a review of the installed equipment locations with respect to postulated harsh environmental zones.

2. The wiring diagrams of safety related electrical equipment as defined in paragraph (b)(1) of 10CFR50.49 are being reviewed to identify any auxiliary devices, electrically connected directly into the control or power circuitry, whose failure due to postulated environmental conditions could prevent the required operation of the safety-related equipment. This review also addresses the potential failure of safety-related electrical equipment after its qualified operating time but before the end of the postulated accident.

3. Included in the review of safety-related systems and equipment described in Step 2 is the identification of auxiliary equipment with electrical components which are necessary for the required operation of the safety-related equipment.

4. Non-safety related electrical circuits associated with equipment identified in Step 1 are being reviewed for proper fuse and breaker coordination such that failure of non-safety related equipment will not affect safety related equipment.

Our May 19, 1983 response to 10CFR50.49 included equipment identified by the above methodology, steps 2, 3 and 4, pertaining to (b)(2) equipment. We are not aware of any unqualified non-safety related equipment whose failure would prohibit accomplishment of the safety functions of (b)(1) equipment. Review as discussed above for (b)(2) equipment is continuing. If additional equipment items are identified they will be qualified in accordance with the provisions of 10CFR50.49.

ENCLOSURE II

SCHEDULE FOR COMPLETION OF OPEN ITEMS,
JUSTIFICATION FOR CONTINUED OPERATION

All equipment qualification open items within the scope of 10 CFR 50.49 are discussed below. For each item, the expected schedule for completion is provided. In addition, a Justification for Continued Operation (JCO) is provided. In all cases, we conclude that there is sufficient assurance that the consequences of any Design Basis Accident can be mitigated, and the safety of the general public will be preserved until the open items are resolved.

1. Limitorque Motor Valve Actuators

   Plant IDs MV-32016, 32017, 32195, 32196
   [32019, 32020, 32197, 32198]

   A project was initiated to re-evaluate the environmental qualification of these motor valve actuators based on the concerns raised in the TER. As a result of this evaluation, we have concluded that modifications or replacements to these installed actuators will be required. These modifications or replacements will be performed prior to March 31, 1985.

   The identified valves perform two functions as described below.

   MV-32016, 32017, [32019, 32020] - Main Steam isolation valve on the steam supply line to the turbine driven auxiliary feedwater pumps. These valves are installed in the auxiliary building.

   MV-32195, 32196, [32197, 32198] - Pressurizer Power operated relief isolation motor valve (block valve). These valves are located inside containment and are redundant to fully qualified air operated relief valves.

   Considerable type testing has been performed by Limitorque; however, the testing does not envelope the Prairie Island conditions for temperature and chemical spray (for inside containment actuators). The actuators are equipped with Class B insulation and were tested to a peak temperature of 250°F as documented in Limitorque qualification report B0003.

   For inside containment actuators, the peak specified temperature is 290°F. A review of the derivation of the 290°F value will reveal its conservatisms. In the SER dated May 12, 1981, the NRC required Prairie Island to use the saturation temperature corresponding to the posutlated accident pressure profile. This resulted in an increase of the 268°F peak accident temperature specified in the Prairie Island FSAR to 289°F. In this respect, the type testing applicable to these actuators falls only 18°F short of the postulated temperature values specified in the FSAR. The post-accident temperature values derived in the FSAR naturally contain inherent conservatisms.

Valve actuators installed inside containment will also be exposed to chemical spray conditions. Although these specific actuators and motors have not been tested to chemical sprays, we have confidence that they could survive exposure for the short duration of their required operating time. Their general construction and materials are reasonably similar to those tested to chemical spray conditions for which successful operation has been demonstrated. As noted earlier, the unqualified valve actuators located inside containment are back-up valves to qualified air operated valves Failure of these motor valves would not adversely affect other safety-related equipment or mislead the operator.

Limitorque valve actuators installed in the auxiliary building may be exposed to peak temperature of 300°F. The analysis on which this is based contains many conservatisms in the development of the thermal-hydraulic model which could likely be reduced through a more detailed analysis. No chemical spray is used in the auxiliary building.

IEEE Standard 117-1974, "IEEE Standard Test Procedure for Evaluation of System of Insulating Materials for Random Wound AC Electric Machinery," provides a temperature and exposure time guide for various classes of motor insulation. For Class B insulation, which is the minimum grade of insulation that the motors in this category contain, the guide states the motor to be operable for 32 days at 150°C (320°F). The longest required operating time for any of these actuators is 1 day. IEEE 117-1974 is an industry standard that most motor manufacturers follow as a guide in preparing motor application guides and is based on considerable motor performance history data. This does provide reasonable assurance of operability in post-accident conditions and forms a sound engineering basis for a JCO.

On the basis of the system requirements and motor specific discussion proved above, we believe a sound engineering basis exists for the continued operation of the plant until motor actuator modifications or replacements can be made. As stated earlier, modification or replacement will be completed prior to March 31, 1985.

2. Westinghouse Motors

   Plant IDs 15-1, 15-4, 16-4, 16-5
   [25-4, 25-5, 26-3, 26-4]

   Resolution of all qualification deficiencies identified in the January 16, 1984 response have been completed. Westinghouse has concluded that existing type testing enveloping the postulated conditions is applicable to the installed motors. This item is qualified.

3. Barton Flow Transmitters

   Plant IDs 23073, 23074, [23075, 23076]

   Our January 16, 1984, response identified that these transmitters would be replaced with qualified transmitters. Replacement has been completed. This item is qualified.

4. Endevco Accelerometer
Unholtz-Dickey Charge Amp

Plant IDs 1EQ-443, 1EQ-444, 1EQ-445, 1EE-443, 1EE-444, 1EE-445,
[2EQ-443, 2EQ-444, 2EQ-445, 2EE-443, 2EE-444, 2EE-445]

These equipment items are part of the pressure relief valve detection systems that have been installed in response to NUREG 0737 paragraph II.D.3, Direct Indication of RCS Relief-and-Safety-Valve Position. An owners group is conducting a qualification test program on this equipment. The testing has been completed and some equipment modifications will be necessary. Any necessary modifications will be completed prior to March 31, 1985.

In addition to this newly installed system, other diverse instrumentation is available to the control room operator that provides indication that the relief or safety valves have lifted. These other indications include PORV position, relief line temperature, pressurizer relief tank pressure and level, and pressurizer pressure and level. PORV position and pressurizer pressure and level instruments are fully qualified.

On the basis of the multiple diverse and qualified equipment that could be used to monitor relief valve position, we believe there is a sound engineering basis for the continued operation of the plant until the qualification testing and any modifications that may be necessary, are completed. Failure of this system would not adversely affect other safety related equipment or mislead the operator. As stated earlier, this will be completed prior to March 31, 1985.

5. Gould-Century unit cooler motors

Plant IDS 113-53, 113-54, 113-55, 123-54, 123-55, 123-56,
[213-52, 213-53, 213-54, 223-53, 223-54, 223-55]

These unit cooler motors are exposed to radiation ($1.8 \times 10^6$ - $3.3 \times 10^7$ Rads) during the recirculation phase. Motors of this type and insulation class are generally qualified to these levels based on other Class B motors installed at Prairie Island that have been radiation tested. However, because specific type testing or documentation of the materials of construction is not available, these motors will be replaced with fully qualified motors manufactured by Reliance Electric. Replacement motors have been ordered and will be installed prior to March 31, 1985. It is important to note that replacement is being performed because qualification documentation does not exist, rather than having qualification information that shows these motors to be unqualified for their service.

Unit cooler motors provide cooling to rooms in which the containment spray (CS), residual heat removal (RHR), and safety injection (SI) pump motors are located. There are no postulated High Energy Line Breaks (HELBs) in the vicinity of these motors that would result in a high

temperature or steam environment. In the event of common-mode unit cooler motor failure as a result of radiation exposure during the recirculation phase, the affected rooms would experience a gradual increase in ambient temperature due to heat generated by the motors. Some room cooling will be provided by the Auxiliary Building Special Ventilation System (ABSV). An increase in ambient temperature as a result of common-mode failure of the unit cooler motors will not lead to failure of the CS, SI and RHR functions. While it is certainly desirable to maintain continuous service of the unit cooler, the inherent conservatisms in the system design and, specifically, the motor design, combined with evaluation of the motor duty requirements, shows that the CS, SI and RHR safety functions can still be accomplished.

Electric motors in general, and specifically the CS, SI and RHR pump motors, are typically built to industry standards, e.g., IEEE, NEMA, etc. These industry standards contain various design specifications for electric motors including operating characteristics in high temperature environments. For example, IEEE Standard 117-1974, "IEEE Standard Test Procedure for Evaluation of Systems of Insulating Materials for Random Wound AC Electric Machinery," provides a temperature/exposure time guide for various classes of insulating materials. For Class B insulation, this standard states that the motor should remain operable for 32 days at 150°C (302°F). Although this is a generic standard and is not sufficient to establish qualification on its own, it does provide some insight to the conservatisms inherent in the design of the motors. NEMA Standard MG-1, Part MG 1-20.40 shows allowable temperature rise for Class B insulated motors to be 90°C by embedded detector method for a motor operating in an ambient temperature of 90°C at a motor service factor of 1.0. Motors of the CS, SI and RHR variety would typically have a temperature rise of less than 80°C during normal motor duty.

A study was performed to determine the ambient temperature in the RHR pits in the event that safeguards chillers (unit coolers) were inoperable. The temperatures calculated in this study would evelope CS and SI pump room conditions because of the relative size of the rooms and the individual motors. The calculated temperature is 160°F during the recirculation phase (180°F fluid temperature). This study has also assumed no cooling by the ABSV System.

The SI and RHR pump motors are manufactured by Westinghouse (W). W has performed an environmental qualification evaluation of these motors and has concluded that they are environmentally qualified. The SI and RHR motors are virtually identical from a qualification standpoint.

The SI pump motor System Component Evaluation Worksheet (SCEW) states a 30 day operating time for these motors; however, post LOCA cooldown and depressurization procedures are in place and SI operation should

not be required after approximately 24 hours. It will take approximately one year to reach a total integrated dose (TID) of $1 \times 10^6$ in the SI pump area, which is likely a conservative threshold for the common-mode failure of the unit cooler motors. Thus, it is highly probable that SI operation would no longer be required by the time the unit cooler motors become inoperable. In the unlikely event that unit cooler motors become inoperable prior to conclusion of SI, SI pump motor operability can be expected. In addition to the generic standards discussed earlier, Westinghouse has performed testing on the stators to a temperature of 210°C for 168 hours with no adverse degradation. The motor lubricant and bearings would also be capable of performing their function in elevated ambient temperatures for the duration of any potential SI operation in the absence of unit coolers. Any failure of the unit cooler motors during or after SI operation would not, therefore, lead to the failure of any other equipment, mislead the reactor operator, or prevent the successful completion of SI function.

RHR system operation could be required to provide long term cooling for up to one year. One train of cooling is capable of providing cooling for any Design Basis Event. Unit cooler motors could reach their postulated radiation threshold within a few days, thus the potential for requiring RHR pump motor operability in an elevated ambient environment is high. Based on testing performed by W, supplemented with additional motor-specific analyses, we can conclude that sufficient RHR cooling can be assured for one year post-accident. The analysis shows that the motor type testing has simulated sufficient qualification for one-year post-accident operation considering the following assumptions:

1. Post accident ambient temperature will not exceed 160°F if unit cooler motors were to exhibit a common-mode failure,
2. Continuous RHR motor operation at the full load of the motor for one year at a conservative motor temperature rise of 70°C (per discussions with W),
3. 500 hours of RHR pump motor operation per year through March, 1985.

RHR motor lubricants and bearings can also be expected to remain operable for this postulated peak temperature because it is below the lubricant service rating. Therefore, based on the motor specific type testing and analysis, we can conclude that any failure of the unit cooler motors would not lead to failure of any other equipment, mislead the reactor operator, or prevent the successful completion of RHR function. Additionally, the SI system and normal charging are alternative equipment which would be available to provide heat removal in the unlikely event that RHR was not available.

CS system operation could be required at containment pressures above 10 psig; it is expected that CS system operability would not be required beyond 24 hours. Similar to the SI discussion provided above, system

functional requirements are expected to be completed prior to common-mode failure of the unit cooler motors. In addition, an evaluation was performed using the motor specific duty cycle and a conservative 90°C motor temperature rise. This evaluation shows that the CS pump motor qualification has sufficient margin to account for any gradual ambient temperature increase in this 24 hour period. Therefore, we can conclude that any failure of the unit cooler motor will not lead to failure of any other equipment, mislead the reactor operator, or prevent the successful completion of CS function.

On the basis of the discussion provided above, we believe that there is a sound engineering basis for continued operation of the plant until the installed unit cooler motors are replaced with fully qualified motors. As stated earlier, this will be completed prior to March 31, 1985.

6. General Electric Motor Control Centers (MCCs)

Plant IDs 1LA1, 1LA2, 1MA1, 1MA2 [2LA1, 2LA2]

These motor control centers are potentially exposed to harsh environments as a result of breaks in the main steam line or in the steam supply line to the turbine driven auxiliary feedwater (AFW) pumps. New MCCs have been purchased for the safety-related loads in the MCCs identified above. The new MCCs will be located in a mild environment.

The safety-related loads in these MCCs are for the safeguards chillers and the main steam supply valve to the turbine driven AFW pumps.

In the event of a steam line break rendering the safeguards chillers inoperable, the effects will be gradual heat-up of the affected areas. The heat-up of the rooms will be due primarily to the lack of the plant's chilled water cooling system rather than a result of a High Energy Line Break. The safety impact of nominal heat rise in these rooms is minor. In addition, the heat-up will be gradual and over a sufficient period of time that alternative cooling methods can be employed, e.g., portable fans.

Failure of the safeguards chiller MCC's would not adversely affect other safety related equipment or mislead the operator. Relocation of safeguards chiller loads will be completed by September 1, 1984.

The main steam supply valve to the turbine driven AFW pumps would be required during a postulated break in the steam supply line to isolate the break. This valve would not be required to function for a main steam line break. Because the MCC for this motor valve would be exposed to a break in the steam supply line, operation of the motor valve and isolation of the line is not assured. To evaluate this potential scenario in more detail, a fracture mechanics evaluation of the steam supply line was performed.

Linear elastic/elastic-plastic fracture mechanics analyses were used to demonstrate that small stable cracks will develop and be detected and repaired before a catastrophic break occurs that would render the MCC inoperable. The defense in depth approach consisted of the following evaluations:

a. The maximum allowable in-service flaw according to ASME, Section XI was postulated and the crack growth was shown to be acceptable throughout the life of the plant.

b. The leakage through a crack twice the wall thickness (2t) in both longitudinal and circumferential directions was shown to be detectable under normal operating conditions (Level A).

c. A crack four times the wall thickness (4t) in both longitudinal and circumferential directions was shown to be stable locally and globally under faulted conditions (Level D).

d. The safety margin of a 4t crack in item "c" was determined.

e. The flow through the crack in item "c" was shown to have no impact on the safety-related equipment in the room.

f. Subcritical crack development was shown to prove that cracks are likely to break through the pipe wall and leak before they propagate around the pipe and cause a break.

The analyses demonstrated that small postulated cracks in the steam supply line to the turbine driven AFW pumps are acceptable. The following specific conclusions can be drawn:

a. The double ended guillotine breaks that were previously postulated are overly conservative.

b. Small cracks in the piping would be stable, i.e., they would not grow, even under twice the Level D loads.

c. Leakage from cracks would be visually detectable during routine Auxiliary Building inspections by operating personnel; and continuous monitoring provided by the steam exclusion system, would detect a degrading environment and alarm in the control room.

On the basis of the functional requirements of the safety-related loads within the affected MCCs, combined with the defense in depth fracture mechanics evaluation of the steam supply line to the turbine driven AFW pumps, we believe there is a sound engineering basis for the continued operation of the plant until the affected loads are relocated. Leakage from a crack would be detected, and would not affect other safety related equipment or mislead the operator. Relocation of AFW pump steam supply loads will be completed by September 1, 1984.

ENCLOSURE III

Auxiliary Building Temperature Transient