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New York Power Authority

Memorandum

October 15, 1984
JPN-84-67

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Equipment Qualification; Response to Questions

Reference: NRC letter, D. B. Vassallo to J. P. Bayne, dated August 20, 1984.

Dear Sir:

The referenced letter transmitted questions on equipment qualification efforts in progress at the FitzPatrick plant. Responses to these questions are enclosed as Attachment 1 and 2 to this letter.

If you have any questions, please do not hesitate to call Mr. J. A. Gray, Jr. of my staff.

Very truly yours,

Robt A Burns for

J. P. Bayne
First Executive Vice President
Chief Operations Officer

cc: Office of the Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 136
Lycoming, NY 13093

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Attachment I to JPN-84-67

Response to NRC Questions on Equipment Qualification

Dated

October 15, 1984

New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333

NRC Question 1. Submit all applicable justifications for continued operation (JCOs) that are currently being relied upon.

NYPA Response 1. Via Reference 1 the Authority transmitted JCOs for those equipment items requiring corrective action to establish environmental qualification. In Reference 2 we were notified that these JCOs were adequate except those for the junction boxes to protect terminal blocks and splices. The Authority provided a JCO for this equipment in Reference 3. In that letter, the Authority updated the status of the JCOs deleting those which were no longer necessary and adding new ones which had been found to be required. Enclosure 1 to this letter presents the currently applicable JCOs.

NRC QUESTION 2. For each JCO associated with equipment that is assumed to fail, provide confirmation that no significant degradation of any safety function or misleading information to the operator will occur as a result of failure of equipment under the accident environment resulting from a design basis event.

NYPA RESPONSE 2. In Attachment 2, each JCO associated with equipment that is assumed to fail, provides confirmation that no significant degradation of any other safety function, or misleading information to the operator (other than that resulting directly from the failed component), will occur as a result of failure of equipment under the environment resulting from a design basis accident.

NRC QUESTION 3. Confirm that in performing the review of the methodology to identify equipment within the scope of 10 CFR 50.49(b)(2) the following steps have been addressed:

a. A list was generated of safety-related electric equipment as defined in paragraph (b)(1) of 10 CFR 50.49 required to remain functional during or following design-basis Loss of Coolant Accident (LOCA) or High Energy Line Break (HELB) Accidents. The LOCA/HELB accidents are the only design-basis accidents which result in significantly adverse environments to electrical equipment which is required for safe shutdown or accident mitigation. The list was based on reviews of the Final Safety Analysis Report (FSAR), Technical Specifications, Emergency Operating Procedures, Piping and Instrumentation Diagrams (P&IDs), and electrical distribution diagrams;

b. The elementary wiring diagrams of the safety-related electrical equipment identified in Item "a" were reviewed to identify any auxiliary devices electrically connected directly into the control or power circuitry of the safety-related equipment (e.g., automatic trips) whose failure due to postulated environmental conditions could prevent required operation of the safety-related equipment and;

c. The operation of the safety-related systems and equipment were reviewed to identify any directly mechanically connected auxiliary systems with electrical components which are necessary for the required operation of the safety-related equipment (e.g., cooling water or lubricating systems). This involved the review of P&IDs, component technical manuals, and/or systems descriptions in the FSAR.

d. Nonsafety-related electrical circuits indirectly associated with the electrical equipment identified in Item "a" by common power supply or physical proximity were considered by a review of the electrical design including the use of applicable industry standards (e.g., IEEE, NEMA, ANSI, UL, and NED) and the use of properly coordinated protective relays, circuit breakers, and fuses for electrical fault protection.

NYPA RESPONSE 3.a.

This item was addressed in Enclosure 3 to Reference 2. That Enclosure provided the "Summary of Methodology for Identifying Electrical Equipment within the Scope of 10 CFR 50.49." The first item in the Enclosure refers to paragraph (b)(1) of 10 CFR 50.49 and states:

"The safety design basis of safety systems as described in the FitzPatrick FSAR were reviewed along with existing plant emergency operating procedures. Based on this review systems and components required to remain functional in order to mitigate postulated design basis events were identified. The environmental conditions at the specific location of the safety-related equipment was then determined. Safety-related equipment determined to experience postulated harsh environments as a result of these events were included in the 10 CFR 50.49 listing."

NYPA RESPONSE 3.b.

This item was addressed in the second paragraph of Enclosure 3 to Reference 4. That paragraph states:

"A review of the electrical elementary diagrams for the safety-related eqipment identified under paragraph (b)(1) was performed. This review confirmed the application of the original plant design criteria for electrical separation of safety-related electrical equipment and circuit coordination/protection schemes. As a result, no non-safety related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions were identified."

NYPA RESPONSE 3.c.

The operation of the safety-related electrical equipment identified in Item "a" was reviewed to identify any directly mechanically connected auxiliary systems which have electrical components which are necessary for the required operation of the safety-related equipment. As a result, no such equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions was identified.

<u>NYPA RESPONSE</u> 3.d.	The response to this question is the same as A3.b
<u>NRC QUESTION</u> 4.	Provide confirmation that all design basis events which could potentially result in a harsh environment, including flooding outside containment, were addressed in identifying safety-related electrical equipment within the scope of 10 CFR 50.49 (b)(1).
<u>NYPA RESPONSE</u> 4.	This question was answered in Reference 4 which states: "The Authority has identified all design-basis events which could result in a potentially harsh environment, including flooding due to a high-energy line break (HELB) outside containment."
<u>NRC QUESTION</u> 5.	Confirm that the electrical equipment within the scope of 10 CFR 50.49(b)(3) is all R.G. 1.97 Category 1 and 2 equipment or that justification has been provided for any such equipment not included in the environmental qualification program.
<u>NYPA RESPONSE</u> 5.	This question was answered in Reference 4, Enclosure 3. The third item in the Enclosure refers to paragraph (b)(3) of 10 CFR 50.49 and states: "Regulatory Guide 1.97, Rev 2 "Instrumentation ... to Assess Plant and Environments During and Following an Accident" was reviewed as it applies to Boiling Water Reactors (BWR). Instruments were then identified which were presently installed in the FitzPatrick Plant and which meet the required design criteria. If those instruments required environmental qualification (categories 1 and 2), the associated components were included in the 10 CFR 50.49 component listing."

References

1. PASNY letter, J.P. Bayne to T.A. Ippolito, dated September 29, 1981 (JPN-81-78).

2. NRC letter, D.B. Vassallo to J.P. Bayne, dated April 19, 1983.

3. NYPA letter, J.P. Bayne to D.B. Vassallo, dated June 6, 1983 (JPN-83-52).

4. NYPA letter, J. P. Bayne to D. B. Vassallo, dated June 15, 1984 (JPN-84-36).

Attachment 2 to JPN-84-67

Response to NRC Questions on Equipment Qualification

Dated

October 15, 1984

New York Power Authority
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: NUCLEAR BOILER (02)

COMPONENT I.D.: 02VMY-71A-L SRV ACOUSTICAL MONITORING
02VME-71A-L SYSTEM

The Safety Relief Valve (SRV) Acoustical Monitoring System was installed in January 1981 in response to NUREG-0578. NRC direction provided to the Authority required immediate installation of equipment to be followed by a qualification program.

The acoustical monitoring system has completed a qualification test program which was commissioned by a utility group. During the test program, the equipment was exposed to test conditions more severe (temperature, pressure, radiation) than the maximum postulated accident conditions at JAFNPP. A comparison of the test program results to the JAFNPP plant specific requirements is currently being performed. If required, the JAFNPP valve monitoring system will be modified to bring the installed system into the tested configuration.

In addition, the part of the system which is located in a harsh environment is fully redundant. Each SRV is equipped with two acoustical sensors and associated preamplifiers. If one sensor channel were to fail, the other sensor channel can be connected to the system cabinet in the Relay Room. In addition, each SRV discharge line is also equipped with temperature sensors which indicate an open SRV. Temperature readouts and alarms are provided in the Control Room.

Based on the redundant equipment used in the system (two acoustical sensors/preamplifiers and temperature sensors) to accomplish the required safety function, and pending the assessment of the test program results, continued operation is justified.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: NUCLEAR BOILER VESSEL INSTRUMENT (02-3)

COMPONENT I.D.: 02-3AU-278 (A-D) - REACTOR HIGH PRESSURE ANALOG TRIP UNIT (RPS) ROSEMOUNT 510 DU

Following postulated Reactor Building HELB's, it is unlikely that reactor trip would be required based on high reactor pressure. If required for this postulated accident, this unit would perform its intended function of providing a trip signal to the normally energized (fail-safe) protection logic. Redundant trip units are provided and are located at different instrument racks locations.

Qualification type test data on file supports qualification for harsh environment parameters for HELB and LOCA with the exception of thermal aging. Harsh environment parameters for temperature, pressure, radiation, and humidity are enveloped by the test data. The aging concern can be partially resolved by the periodic surveillance testing which is performed to verify trip unit operation and calibration.

The function of this component is performed in the initial phases of postulated design basis accidents. The type-testing noted above provides a high degree of confidence that this component will perform its intended design function. In addition, redundant trip units are located at different instrument rack locations. There is also diverse instrumentation which can initiate a reactor trip during a postulated Reactor Building HELB.

Based on the validity of the partial test data in support of qualification, continued operation is considered justified pending replacement of these items.

References

a. Rosemount Report 3768A - "Qualification Test Summary for the Trip/Calibration System Rosemount Model 510DU", dated March 10, 1976.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: NUCLEAR BOILER VESSEL INSTRUMENT (02-3)

COMPONENT I.D.: 02-3PT-178 (A-D) REACTOR HIGH PRESSURE TRANSMITTER (RPS) ROSEMOUNT 1151GP

This pressure transmitter provides a trip signal on high reactor pressure to the Reactor Protection System. The Rosemount 1151GP transmitter has been fully type tested to IEEE 323-1971 for harsh environment parameters of pressure, radiation, temperature, and humidity at more severe levels than experienced in the specific JAF locations. The only outstanding qualification issue is aging of the transmitter's electronics. This concern can be partially resolved due to the periodic surveillance testing which is performed to verify transmitter operation and calibration.

The function of this component is performed in the initial phases of postulated design basis accidents. It will experience a harsh environment following postulated Reactor Building HELB's. The type testing noted above provides a high degree of confidence that this unit will perform its intended function. In addition to redundant sensors located at different instrument locations, there is also diverse instrumentation which can initiate a reactor trip for this postulated event.

Based on the validity of the partial test data in support of qualification, continued operation is considered justified pending replacement of these items.

References

a) Rosemount, Inc. Report 127227, Rev. B - "Nuclear Service Qualification Testing Interim Report, Model 1151DP - Differential Pressure Transmitter"

b) Rosemount, Inc. Report 117415, Rev. A - "Qualification Tests for Rosemount Pressure Transmitter, Model 1152"

c) EDS Assessment Report 0900-001-005 - "Evaluation of Class 1E Transmitter (Rosemount Model 1151GP)"

d) General Electrical Environment Qualification Record-DV File 145C3240 (Rosemount Model 1151DP)

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: NUCLEAR BOILER VESSEL INSTRUMENT (02-3)

COMPONENT I.D.: 02-3LITS-73 REACTOR WATER LEVEL INDICATING SWITCH
YARWAY 4418EC

This unit provides a reactor water level permissive signal to the RHR System and provides indication of water level to a Control Room indicator. This unit is qualified to perform its short term function of providing reactor water level permissive signal to the RHR System. However, test data does not support full qualification for providing reactor water level indication over an extended post-accident time frame after a postulated HELB in the Reactor Building.

Type test data of identical equipment at elevated temperatures and humidity in conjunction with a radiation threshold analysis of internal components supports qualification for its Reactor Building location following postulated LOCA's inside containment. Type test data supports short term operation following postulated Reactor Building HELB's. This is considered acceptable based on other water level indications available in the Control Room and the ability to access the Reactor Building for repairs within a short time frame following a postulated Reactor Building HELB.

Based on the validity of partial test data and the availability of alternate equipment to perform the safety function, continued operation is considered justified pending replacement of this item.

References

a) Yarway Report No. 3232-3155
b) Yarway Report No. 5628-3509

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: CONTROL ROD DRIVE (CRD) (03)

COMPONENT I.D.: 03SOV-117; -118; -140A, B; -31A, B
SCRAM AIR PILOT SOLENOID VALVES

The solenoid valve performs its safety related function immediately upon initiation of the postulated accident and prior to failure if any. In addition, the solenoid valve is required to de-energize in the performance of its safety related function., i.e., the valve is "fail safe". Based on the completion of the solenoid valve's safety related function prior to significant exposure to the accident environment, failure of this device would not cause degradation of any safety function or provide misleading information to the operator. An ongoing qualification program has identified type test data for identical solenoid valves for high temperature and humidity conditions.

Based on the validity of partial test data obtained for the solenoid valves and that failure of these items will result in no degradation of any safety function, continued operation is considered justified pending finalization of the qualification documentation.

References

a) General Electric Qualification Report, NEDC 30602-P, dated April, 1984 - "Scram Pilot Solenoid (ASCU HVA-90-405) Environmental Qualification Study for NYPA"

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: REACTOR CORE ISOLATION COOLING

COMPONENT I.D.: 13LS-12 BAROMETRIC CONDENSER TANK LEVEL SWITCH

This switch performs its safety related design function in a mild environment. However, the existing system logic design incorporates common electrical fusing for this item and harsh environment electrical equipment requiring qualification. The main concern for this switch is that it does not lose its insulation resistance to ground when exposed to harsh environment, de-energizing other safety related equipment.

In a postulated HELB, isolation of affected lines is detected and the isolation signal provided within a matter of seconds. The probability that this switch could develop a significant ground within this short time is extremely small.

Failure of this switch will result in no significant degradation of any safety function, and misleading information to the operator will not occur since the RCIC system is not relied upon to mitigate the subject design basis event.

Based on this summary, continued operation is considered justified pending replacement of this item.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: CONTAINMENT SYSTEM (16)

COMPONENT I.D.: 16-1RTD-107, -108 - DRYWELL AMBIENT TEMPERATURE DETECTOR

The materials in the RTD's which may be subject to deterioration due to the harsh environment are the RTD mandrel, the lead wire insulation and the terminal blocks. However, considering that all materials used in the construction of the RTD's have temperature ratings greater than the maximum postulated accident temperature and that the terminal block is enclosed in a gasketed exlosion proof head, the RTD's can be expected to function after exposure to the harsh environment from a postulated accident.

In addition, there are a large number of thermocouples also measuring drywell ambient temperature, which provides a large measure of redundancy and diversity.

Based on the ratings of the materials used in the construction of the RTD's in conjunction with the large measure of redundancy in the system to accomplish the safety function, continued operation is justified pending item replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: RADIATION MONITORING (17)

COMPONENT I.D.:
a. 17RE-50A, B - STACK EXHAUST EFFLUENT MONITOR (LOW RANGE)
17RE-53A, B - STACK EXHAUST EFFLUENT
17RT-53A, B - MONITOR (HIGH RANGE)
b. 17RE-431, -432 - TURBINE BLDG. EXH. EFFLUENT MONITOR (LOW RANGE)
17RE-434A, B - TURBINE BLDG. EXH.
17RT-434A, B EFFLUENT MONITOR (HIGH RANGE)
c. 17RE-458A, B - RADWASTE BLDG. EXH. EFFLUENT MONITOR (LOW RANGE)
17RE-463A, B - RADWASTE BLDG. EXH.
17RE-463A, B EFFLUENT MONITOR (HIGH RANGE)

These monitors provide for measurement of post-accident plant effluents for release assessment. This equipment is located remote from the plant areas experiencing direct postulated accident environments. However, sample stream radiation levels can result in higher than normal radiation levels in the area of this instrumentation. Both the high and low range units are specifically designed to measure the required range of radioactivity. The high range units are also shielded to protect their electronic components from the accident stream radiation.

Based on this data, continued operation is considered justified pending completion of re-analysis of the shielding design for these units.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: RADWASTE (20)

COMPONENT I.D.: 20PNS-83, 20PNS-95 DRYWELL FLOOW DRAIN SUMP OUTBOARD ISOL. VALVE POSITION SWITCH

The radwaste position switches are required to function short term for indication of primary containment isolation following a postulated LOCA. The valves on which these switches are located fail safe on loss of power. Therefore, there can be high confidence in the position of these valves without relying on the functioning of these position switches. The exposure to this accident environment requires the switches to operate in an elevated radiation environment. No significant temperature or pressure increases are postulated for the location of the position of the position switches due to LOCA event. Radiation is the only harsh environment accident parameter due to the switches' location in the East Pipe Tunnel. Component materials of the NAMCO limit switch have been identified and qualification documentation located. The qualification data has been evaluated per DOR guidelines and by applying Arrhenius techniques. Results of this evaluation indicate that the non-metallic components have greater than 9×10^3 years of expected life at the maximum pipe tunnel temperature of 104°F except for Buna-N. The Buna-N components have an expected life of greater than 11 years.

Additionally, a radiation analysis performed on the component materials shows that the radiation threshold for each non-metallic material is greater than the postulated total integrated dose.

Based on the time-temperature and radiation analysis performed in conjunction with the validity of partial test data, continued operation is considered justified pending replacement.

References

1) Masoneilan International Report No. 1003

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: RADWASTE (20)

COMPONENT I.D.: 20SOV-83, 20SOV95 DRYWELL FLOOR DRAIN SUMP OUTBOARD ISOL. VALVE PILOT SOLENOID

The Radwaste solenoid operated valves are required to function short term for primary containment isolation during a postulated LOCA. The exposure to this accident environment requires the valves to operate in an elevated radiation environment. A radiation exposure analysis indicated that the radiation threshold for the device exceeds the postulated total integrated dose. No significant temperature or pressure increases are postulated for the location of the solenoid valves due to the LOCA event.

In addition, the solenoid valve is required to de-energize in the performance of its safety related function, i.e., the valve is "fail safe". Based on the short term operational requirement of the valve, and that failure of this device would not cause degradation of any safety function, continued operation is considered justified pending replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: HIGH PRESSURE COOLANT INJECTION

COMPONENT I.D.: 23LS-99 GLAND SEAL COND. HOTWELL LEVEL SW.
23LS-100 GLAND SEAL COND. HOTWELL HIGH LEVEL SW.

This switch performs its safety related design function in a mild environment. However, the existing system logic design incorporates common electrical fusing for this item and other harsh environment electrical equipment requiring qualification. The main concern for this switch is that it does not lose its insulation resistance to ground when exposed to harsh environment, de-energizing other safety related equipment.

In a postulated HELB, isolation of affected lines is detected and the isolation signal provided within seconds. The probability that this switch could develop a significant ground within this short time is extremely small.

Failure of this switch will result in no significant degradation of any safety function and misleading information to the operator will not occur since the HPCI system is not required to mitigate a Reactor Building HELB design basis event.

Based on this summary, continued operation is considered justified pending replacement of this item.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: PRIMARY CONTAINMENT ATMOSPHERE CONTROL/MONITORING (27)

COMPONENT I.D.: 27E/P-103A,B N2 FLOW TO CONTAINMENT ELECTRO-PNEUMATIC CONVERTER FOR 27FCV-103A, B

These instruments are utilized to control the flow of nitrogen to the containment following a postulated LOCA inside containment. This equipment is not exposed to the direct LOCA environment but to secondary environmental effects in the Reactor Building (radiation, elevated temperature).

Alternate methods are available for establishing nitrogen flow to the containment for venting purposes should this electro-pnuematic converter fail. The alternate methods utilize fully qualified equipment in the Reactor Building in conjunction with manual control of the nitrogen flow from a mild environment (CAD Building).

Since the safety function can be accomplished by alternate equipment, continued operation is considered justified pending replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: PRIMARY CONTAINMENT ATMOSPHERE CONTROL/MONITORING

COMPONENT I.D.: 2702-AZ-101A,B AND 27DWA-HTA, HTB - PRIMARY CONTAINMENT OXYGEN ANALYZER AND HEAT TRACING SYSTEM (BECKMAN,T..YLOR)

If the containment 02 analyzers were assumed to fail as a result of postulated environmental conditions in the Reactor Building due to a LOCA inside containment (radiation, elevated temperature 120°F, humidity), the determination to vent the containment of accumulated gases is made based on hydrogen concentration only. This is allowed by the revised emergency procedures based on the BWR Emergency Procedure Guidelines.

This is further justified because JAF operates with an inerted containment (nitrogen), therefore, the only source of oxygen is small and due to radiolytic reaction. The containment is completely isolated from all sources of outside air. In addition, all instrument piping uses nitrogen instead of air for normal operations.

Furthermore, the containment atmosphere can also be sampled using the Post-Accident Sample System (PASS), and the sample analyzed in the laboratory to determine oxygen content.

Based on the availability of other qualified equipment (hydrogen analysis) and PASS system, continued plant operation is justified pending O_2 analyzer replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: PRIMARY CONTAINMENT ATMOSPHERE CONTROL/MONITORING (27)

COMPONENT I.D.: 27RTD-101(A-D)- SUPPRESSION POOL TEMPERATURE SENSORS (RTD)

These instruments are utilized to provide a secondary indication of suppression pool temperature following postulated design basis events. This instrumentation provides redundant indication to the fully-qualified 16 RTD suppression pool temperature monitoring system.

Based on the use of these indicators as a secondary source of torus temperature, continued operation is justified pending replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: REACTOR BUILDING VENTILATION (66)

COMPONENT I.D.: 66UC22A-K FAN MOTORS

The Crescent Area Unit cooler motors are Severe Duty Motors mounted within totally enclosed air-over enclosures (TEAO). The motors are designed to operate in a continuous ambient of 150 F with 100% relative humidity. The maximum temperature in the Crescent Area after a postulated LOCA is 110°F and for a HELB a temperature transient above 150°F for 10 minutes occurs. However, the motors will not experience these temperatures as they are in-duct mounted downstream of the cooling coils. The maximum integrated radiation exposure in Crescent Area is 6.9×10^6 R. Testing of similar motors with same class insulation shows no significant degradation of insulation due to these levels of radiation.

Based on the validity of partial radiation test data in conjunction with the analysis that shows the equipment will not be exposed to conditions more severe than its design ratings, continued operation is justified pending replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: DRYWELL COOLING SYSTEM (68)

COMPONENT I.D.: 68TE-201 thru 212 DRYWELL
68TE-301 thru 310 THERMOCOUPLES

Component materials of the thermocouple have been identified, and a preliminary evaluation of time-temperature and radiation effects performed.

The materials in these thermocouples consist of metal (Cu-Const), ceramic insulators, and a pressed asbestos terminal block with material trade name "Hemit". The ceramic insulators are aging and radiation insensitive and the "Hemit" material which is functional up to 400°C is also listed as aging and radiation insensitive.

In addition, there are a total of twenty-six (26) thermocouples sensing drywell air temperature which provides a large measure of redundancy.

Based on the ratings of the materials used in the construction of the thermocouples, no degradation to equipment operation should occur as a result of the postulated accident condition. Therefore, continued operation is justified pending replacement.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: ELECTRICAL POWER (71)

COMPONENT I.D.: 71ACA5, B5 - 120 VAC DISTRIBUTION PANEL
PT-71ACA4, B5 INCLUDING 600/120 VAC TRANSFORMER

These panels power two (2) electrical loads which may be required to mitigated postulated design basis accidents.

LOAD

a. 27NS-CA, CB - Nitrogen Instrument Supply Cabinet
b. 71INV-3A,3B - LPCI Independent Power Supply Control Power

1. 27NS-CA, CB - These panels provide 120VAC power to various insturments and control components associated with the nitrogen Containment Air Dilution (CAD) System. This equipment would only be required to perform its intended design function following a postulated LOCA inside primary containment. These electrical panels would not be exposed to the direct accident environment but would be exposed to secondary environmental effects on elevation 300' of the Reactor Building. This accident environment would consist of a mild increase in temperature (110°F maximum), a mild humidity transient, and radiation ($3.0x10^5$ rads).

   Distribution breakers of similar design have been shown by type testing to withstand radiation doses of 4.4x10 rads. In addition, since the electrical circuits are loaded to a maximum of 80% of their trip rating by design (40-104F), operation at a maximum temperature of 110°F will not trip the breaker.

2. 71INV-3A,3B - This load is the control power supply for the LPCI independent power supply charger/ inverter logic. This power source is only required for startup of the LPCI charger/ inverter. Once output voltage is established, failure of the external power source to the control logic will will not affect inverter operation.

Based on the above reasons, continued operation is considered justified pending relocation of these loads to a distribution panel in a mild environment.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: ELECTRICAL POWER SYSTEM (71)

COMPONENT I.D.: 71L15, L16, SWITCHGEAR (DISCONNECT, TRANSFORMER, BREAKERS) - GENERAL ELECTRIC AKD-5

The AKD-5 switchgear has been qualified by a combined test and analysis program to accident environmental conditions for pressure, temperature, and humidity which envelope those conditions postulated for the JAF plant locations. Thermal aging and radiation have been addressed by analysis per DOR Guideline requirements.

The G.E. transformer with Class 220°C insulation system has been tested in a 90% humidity environment and has a maximum hot spot temperature rating of 220°C (150°C rise). The JAF transformers are loaded to 50% of design rating and the rated nameplate KVA of these transformers operate with an 80°C average winding rise.

Thus, the insulation winding system is being operated appreciably below its rating. As a result, postulated ambient temperature increases due to a HELB (less than 70°C) will not affect the operability of this component. Internal cabinet humidity is lower than the external cabinet humidity due to the heat generated in the transformer. Thermal aging and radiation have been addressed by analysis per DOR Guideline requirements.

Based on the validity of this partial test data in support of qualification, continued operation is justified.

References

1) Wyle Test Report 17655-SW6-1 dated August 30, 1984 - Nuclear Environmental Assessment Report on G.E. Switchgear Type AKD-5 for Use in Nine Mile Point Unit 1 Nuclear Power Plant

2) Patel Report PEI-TR-92-4-130- Final Assessment Report on General Electric AKD-5 Switchgear Used in JAFFNPP

3) General Electric NEDE-30303, November, 1983 - JAFNPP Switchgear Environmental Evaluation of AKD-5 Switchgear Equipment (L15 and L16)

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: ELECTRICAL SYSTEM (71)

COMPONENT I.D.: 71INV-3A, 3B and 71BAT-3A, 3B
LPCI INDEPENDENT POWER SUPPLY
CHARGER/INVERTER

I. Justification for continued operation is provided based on the following:

a. Primary Containment LOCA (large) - This equipment is located remote from the direct harsh environment of this accident and would perform its intended design function of providing power to the LPCI valve bus prior to the local temperature, radiation, or humidity significantly exceeding normal conditions. The required operating time is less than 3 minutes for operation of reactor recirculation loop isolation valves and opening of the LPCI injection valves.

b. Primary Containment LOCA (small and intermediate) - This equipment is located remote from the direct accident environment. Although the required operating time is significantly longer for this accident (6 hours maximum), no significant accident radiation exposure is expected due to its elevation in the Reactor Building and minimal fuel damage that is postulated for this accident. The long term temperature does not significantly increase above normal (110°F).

c. Reactor Building HELB - A method of plant depressurization and cooldown following a postulated HPCI or RCIC steam line break is described in NEDO-24297, Revision 1 ("High Energy Line Break Evaluation for the James A. FitzPatrick Nuclear Power Plant" dated October, 1980), Section 6.2.2. This method of plant cooldown requires manual depressurization of the reactor using the Automatic Depressurization System (ADS) while restoring and maintaing water level using one of the two Core Spray Pumps. Based on this analysis, there would be some core heat-up. however, there would be considerable margin to the 10CFR50, Appendix K limit of 2200°F peak clad temperature (PCT).

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: ELECTRICAL SYSTEM (71)

COMPONENT I.D.: 71INV-3A, 3B and 71BAT-3A, 3B
LPCI INDEPENDENT POWER SUPPLY
CHARGER/INVERTER

Following a RWCU line break, HPCI and RCIC Systems located in the Crescent Area experience an insignificant change in environmental conditions (5°F rise for less than 30 seconds, and a 0.5 psig pressure rise for less than 30 seconds). Therefore, these systems will remain functional to provide high pressure cooling. Following depressurization using RCIC or HPCI, reactor inventory can be maintained using one of two Core Spray Pumps. Refer to NEDO-24297, Revision 1 (Section 6.2.4).

II. In addition, post-HELB temperature/pressure analyses are extremely conservative.

III. Based on the above analysis, continued operation is considered justified pending completion of installation of environmental enclosures aroung this equipment.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: ELECTRICAL POWER SYSTEM (71)

COMPONENT I.D.: AC AND DC MOTOR CONTROL CENTERS (MCC) - (MCC-151, -152 -153, -155, -161, -162, -163, -165 AND BMCC-1,-2, -3, -4 AND -6 GENERAL ELECTRIC 7700 SERIES MOTOR CONTROL CENTERS

Qualification test data applicable to these General Electric Motor Control Centers (MCC's) has been identified and a plant specific qualification report developed which meets DOR Guideline requirements. This report is being finalized. Reviews of drafts of this report confirms that the specific JAF MCC equipment is qualified to the postulated design basis event and environments for the specific equipment locations. This qualification program consisted of the following phases:

1. Preliminary assessment of applicable G.E. MCC test data resulted in the identification of environmental qualification testing applicable to the JAVNPP MCC's and the postulated harsh environments.

2. Subsequently, a Phase I engineering report (G.E. NEDC-30322-P) was prepared for fifteen specific MCC components installed in the JAF G.E. 7700 Series Motor Control Centers. This report provided further substantiation of the qualification of the subject MCC's to the JAF normal and accident environmental conditions.

3. A final qualification test report (G.E. NEDC-30694-P) for the JAF Motor Control Centers was completed by General Electric in draft form in August, 1984, and has been reviewed by the Authority, and its qualification consultant, and an independent reviewer. Comments from this review are presently being resolved or incorporated. Final issuance of this report is scheduled for November, 1984.

Based on the valid test data obtained and evaluations performed, continued operation is justified pending qualification report completion.

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: MISCELLANEOUS

COMPONENT I.D.: CURTIS (TB) TERMINAL BLOCK IN A GASKETED STEEL JUNCTION BOX

Component materials of the Curtis terminal blocks have been identified and qualification documentation on similar equipment located. The physical design of these blocks and materials are similar to the qualified General Electric terminal blocks. The Curtis blocks are one piece, molded, 12 point (30A) manufactured of general purpose black, phenolic. They are designed for a continuous rating of 250°F and have a threshold for radiation damage greater than 1.0×10^6 R.

The qualification documentation of the General Electric terminal blocks shows that the equipment performed successfully under test conditions (temperature, pressure, and radiation) more severe than the postulated accident conditions at JAFNPP. These terminal blocks are located outside of containment at JAFNPP.

Based on the validity of test data on similar equipment (materials and design), continued operation is justified.

References

1) "Environmental Qualification of Terminal Blocks/Boxes -EB-25", Report 50-213

2) Limitorque Report B0119

3) Westinghouse Report PEN-TR-77-83, "Test Report on the Effect of a LOCA on the Electrical Performance of Four Terminal Blocks"

4) "Radiation Hardness of Termimal Blocks", Westinghouse Memo No. 76-1CC-QUAEQ-M24

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: MISCELLANEOUS

COMPONENT I.D.: CINCH JONES (TB) TERMINAL BLOCK IN A GASKETED STEEL JUNCTION BOX

Component materials of the Cinch Jones terminal blocks have been identified and qualification documentation on similar equipment located. The physical design of these blocks and materials are similar to the qualified General Electric terminal blocks. The Cinch Jones blocks are one piece, molded, 12 point (30A) manufactured of general purpose black phenolic. They are designed for a continuous rating of 250°F and have a threshold for radiation damage greater than 1.0×10^6 R.

The qualification documentation of the General Electric terminal blocks shows that the equipment performed successfully under test conditions (temperature, pressure, and radiation) more severe than the postulated accident conditions at JAFNPP. These terminal blocks are located outside of containment at JAFNPP.

Based on the validity of test date on similar equipment (materials and design), continued operation is justified

References

1) "Environmental Qualification of Terminal Blocks/Boxes -EB-25, Report 50-213

2) Limitorque Report B0119

3) Westinghouse Report PEN-TR-77-83, "Test Report on the Effect of a LOCA on the Electrical Performance of Four Terminal Blocks"

4) "Radiation Hardness of Terminal Blocks", Westinghouse Memo No. 76-1CC-QUAEQ-M24

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED EQUIPMENT

JUSTIFICATION FOR CONTINUED OPERATION

SYSTEM: MISCELLANEOUS

COMPONENT I.D.: STATES (TB) TERMINAL BLOCK IN A GASKETED STEEL JUNCTION BOX

Component materials of the States terminal blocks have been identified and partial qualification documentation located. The materials have been evaluated per the DOR guidelines and by applying Arrhenius techniques. Results of the evaluation indicate that the lowest expected life of the terminal blocks is 266 years at the maximum Reactor Building temperature, 104°F. The qualification documentation shows that the terminal blocks have been successfully irradiated to a level of 2.2 x 10^8 rads gamma.

Though qualification documentation was not located for States terminal blocks showing testing to envelope the postulated peak temperature and pressure requirements, there is documentation to demonstrate the qualification of similar terminal blocks. The physical design of these blocks and materials are similar to qualified General Electric terminal blocks.

The qualification documentation on the General Electric terminal blocks shows that the equipment performed successfully under test conditions more severe than the maximum postulated accident conditions at JAFNPP.

Based on the validity of partial test data on the States terminal block and the successful performance of similar terminal blocks under test conditions more severe than the postulated accident conditions at JAFNPP, continued operation is justified.

References

1) "Test Report on States Terminal Blocks and Test Switches", Report No. 15809-82, Rev. 2

2) "Environmental Qualification of Terminal Blocks/Boxes -EB-25", Report 50-213

3) Limitorque Report B0119