GPU Nuclear

GPU Nuclear Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054-1149
(201) 263-6500
TELEX 136-482
Writer's Direct Dial Number:

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Office of Nuclear Reactor Regulation
Attn: Harold R. Denton, Director
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Schedular Extensions - Environmental Qualification

In accordance with provisions of 10CFR50.49(g) GPU Nuclear hereby requests schedular extensions for certain electrical components required to be environmentally qualified by March 31, 1985. These electrical components are listed below:

Component	Component Manuf.	Reason for scheduler extension	JIO No.
Incore Thermoccuple Extension Cables and Connectors	Continental/ Bendix	Qualification testing of the existing cables and connectors is planned to be complete in April 1985 based on vendor bids.The final report is due out in May 1985. Pending the outcome of the testing, replacement may be necessary.GPUN is proceeding in a parallel path with the testing to procure Environmentally qualified replacement cables should the	JIO-TI-84-6 Rev. 1

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		cables fail but the connectors pass.Projected scope of procurement and fabrication is 30 weeks and engineering and installation is 28 weeks (18 weeks of which overlap procurement and fabrication). This installation schedule will permit installation (if required) in November 1985.Installation (if required) will be accomplished during a scheduled plant shutdown.	
High Range Radiation Monitoring Cable & Assembly	Anaconda/ Victoreen	The modified Time Wire cable will be replaced with qualified triaxial cable and environmental seals will be provided at the terminal box. Delivery of the cable and connector at the detector end is expected in approximately 15 weeks after receipt of purchase order. Installation would require a scheduled shutdown of 30 days.	JIO-TI-84-9 Rev. 1

For each component listed above we have included a Justification for Interim Operation (JIO) in GPUN letter dated December 11, 1984 (5211-84-2292) and revised versions are attached. Further, we are requesting schedular extensions for those components until November 30, 1985.

Sincerely,

RWKeaton for

R. F. Wilson
Director, Technical Functions

lr/0534e

cc: R. Conte
J. Stolz
J. Van Vliet

COMPONENT

Tag No: None
Description: TMI-1 Incore Thermocouple Cable Assembly

OBJECTIVE

The objective of this JIO is:

a. to demonstrate that the safety function can be accomplished within the time frame for which the cable assembly is qualified.

b. to demonstrate that the safety function can be accomplished by designated alternative instrumentation.

c. to demonstrate the applicability of TMI-2 accident test data in support of the cable assembly qualification.

EQUIPMENT DESCRIPTION

Equipment Description	Manufacturer	Model No.
Incore Monitoring System	Babcock & Wilcox	DA7L-1B-1T-1C-128
Incore assembly to cable connector	Bendix	JT07A-14-19P (302) JT06A-14-19S (101)
Cable	Continental Wire & Cable Co.	GAI B/M EK-15L

The incore thermocouple cable, which extends from the incore assembly connector to the reactor building penetration has not been demonstrated to be environmentally qualified in accordance with 10CFR50.49 for the combined steam and pressure/temperature environment associated with a large break LOCA or steam line break accident.

EQUIPMENT LOCATION

The incore thermocouple junctions are located in the outlet plenum of the fuel assemblies, inside the reactor vessel. The thermocouple wires for each assembly pass through a pressure boundary (seal plate) located on the south side of elevation 346' inside containment. This elevation is the operating deck and is outside the reactor cavity and D-ring. The cable assembly is, therefore, protected against the jet impingement effects of LOCAs and HELBs. It is exposed to the bulk containment environment. The Bendix connector, which is located outside of the seal plate pressure boundary, enables the incore detector assembly to be connected to an external cable that extends to the containment penetration.

EQUIPMENT FUNCTION

The incore thermocouples provide a direct measurement of the core outlet temperature from 52 locations in the outlet plenum of the fuel assemblies. Data from all of the 52 incore T/C data are available in the control room via the plant computer. Sixteen of these T/Cs (4 per quadrant) may be read out digitally in the control room. These 16 T/Cs are designated as the Backup Incore Readout System (BIRO).

EVALUATION

a. Large Break LOCA Steam/Pressure/Temperature

The incore thermocouples are used to indicate that the core is being cooled by verifying the effects of LPI cooling. The minimum qualification period is the time necessary to achieve and verify stable cooling by the low pressure injection (LPI) system. For the smallest large break LOCA of 0.5 Ft^2, the RCS depressurizes to pressures at which LPI achieves full flow in under five minutes (refer to Figure C-10 of Reference 1). Within two hours, the LPI suction would be established from the building sump and long term core cooling established. Flow of at least 1000 gpm per LPI train is specified in plant procedures. In the case of a core flood line break accident, this assures at least 1000 gpm to the vessel. This is sufficient flow to assure adequate core cooling for any large break LOCA. Long term indication that core cooling is maintained may also be obtained from LPI heat removal path components.

The principal method of assuring stable long term cooling is by monitoring low pressure injection flow. This instrumentation will be qualified for operation in a post-LOCA environment before restart of TMI-1. Additional qualified instrumentation will also be available to determine that the DH heat exchangers are functioning (DH outlet temperature and certain secondary services cooling water temperature will be available to the operator as per R.G. 1.97). Verification of proper heat exchanger performance and LPI flow assures long term core cooling following any large break LOCA.

b. Use of Incore Thermocouples During Accident/Transients (SB LOCA and HELB Inside Containment)

1. Alternate Instrumentation

The symptom oriented emergency procedures at TMI-1 cover both design basis and beyond design basis accidents and transients. ATPs refer to the incore thermocouples for use in the following situations:

INDICATION	ACTION	ALTERNATE
1. Subcooling margin (SCM) less than 25F as calculated by T/Cs and RCS pressure[1]	• Initiate HPI • Raise OTSG level to 90 - 95%	Use RCS pressure and temperature and calculate SCM once flow is verified in the RC loops in accordance with plant procedure ATP 1210-10. (see Item 5)
2. Subcooling margin greater than 25F[1]	• Allowed to throttle HPI if pressurizer level is on scale • Restart RCPs	Use RCS pressure and temperature and calculate SCM once flow is verified in the RC loops in accordance with plant procedure ATP 1210-10. (see Item 5)
3. Subcooling margin approaching 100F subcooling[1]	• Throttle HPI to prevent exceeding PTS limits	Use RCS pressure and temperature and calculate SCM once flow is verified in the RC loops in accordance with plant procedure ATP 1210-10. (see Item 5)
4. Incore T/Cs do not indicate a decreasing temperature trend	• Maintain a minimum EFW flow of at least 225 gpm until OTSG level reaches 90 - 95% (Ref 8)	Plant procedure requires OTSG level to be raised to 90-95% upon loss of SCM (see Item 1) although minimum flow would not be specified. OTSG level would still be raised to 90-95% in time to prevent prevent core uncovery

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1 With reactor coolant pumps running, the subcooling margin is calculated from the hot leg RTDs. If RCPs are off, the subcooling margin is calculated from the incore thermocouples.

INDICATION	ACTION	ALTERNATE
5. Incore T/Cs and hot leg RTDs not trending together	• This is one indication of a loss of natural circulation. Attempt to re-establish natural circulation flow	Use RCS temperature T_{HOT} + T_{COLD} OTSG level and pressure steam & feed flow

None of the above actions are taken during a large break LOCA. Rather, they are intended to maintain or restore the plant to subcooled conditions during HELBs and SB LOCAs and prevent core damage.

2. T/C Required Operating Time and Environment

The RCS would be cooled down and depressurized with LPI removing decay heat in 2 - 10 hours during an SB LOCA. For an SB LOCA which does not quickly depressurize the RCS to LPI operating conditions, the containment cooling system heat removal capacity will exceed the energy release to the reactor building while the OTSG heat removal system removes the core decay heat and cools the RCS. Thus, the containment pressure and temperature will be returned to nearly pre-event conditions within a relatively short period. With LPI established, the core is kept covered by the LPI flow. LPI mode core cooling can be confirmed from LPI flow and DH heat removal system performance monitoring.

Therefore, the incore thermocouples would have to be available for about 10 hours in order to perform the above actions specified in the emergency procedure functions. At the end of that time, the RCS would be cooled down and depressurized with LPI removing decay heat. Small break LOCAs can be described into categories based on their effect on RCS pressure. The following discussion describes these three categories and discusses the time required for incore T/Cs during each type break and the effect of such a break on the containment environment.

The largest small break LOCAs result in continuous depressurization to the LPI setpoint. Examples of such RCS pressure responses are illustrated in B&W ECCS analyses (Reference 3) and TMI-1 ATOG, Part II, Volume 2 (Reference 4). Analyzed breaks of .055 ft^2 and larger depressurize to below the core flood tank actuation setpoint in less than 45 minutes. Since the break size is large enough to remove decay heat, continued cooldown and depressurization to pressures at which LPI flow is established is on the order of several hours. Once LPI flow is established, then core cooling is verified by LPI flow and DH heat exchanger performance. Since this size of break is large enough to remove decay heat, the containment temperature and pressure peaks at about 30 psig and 240-250F.

The next size category of breaks stabilizes at pressures near the OTSG pressure. This occurs because the break is too small to remove decay heat. The OTSGs remove the remaining decay heat and cool the cold leg temperature to the OTSG saturation temperature. As RCS pressure decreases, the break flow decreases, and HPI flow increases. With two HPI pumps available, the sensible heat of the HPI in combination with the break flow exceeds decay heat almost immediately and quickly depressurizes the RCS to the LPI setpoint in several hours. With one HPI pump available, the break is large enough to initially exceed the flow from one HPI pump. The RCS either does not refill until several hours into the transient or until the LPI system is initiated. Based on mass and energy balance calculations, the RCS depressurizes to the LPI initiation setpoint in less than 10 hours (Reference 4). HPI flow to the vessel is assumed to be a maximum of 330 gpm (i.e. 64% to the core and 36% out the break).

The containment response to such a break (.085 ft^2 LOCA) with one HPI pump available has been analyzed in Reference 6. Since the break is too small to remove all of the reactor decay heat, the building active and passive heat sinks remove the energy relieved to the building. Pressure never exceeds 15 psig and temperature, which peaks at 200F is reduced below 175F within 10 minutes.

The third class of small breaks belongs to those which are too small to remove decay heat or pass full HPI flow. The initial result is a voiding of the RCS and loss of heat transfer to the secondary side. Eventually, the RCS voids sufficiently to initiate boiler condenser heat transfer to the OTSGs and RCS pressure decreases. Reduction of RCS pressure to the CFT actuation setpoint will cause the RCS to continue cooldown even without OTSG heat transfer. The system begins to fill as a result of this depressurization (break flow decreases and HPI flow increases). In 2-3 hours, the decay heat drops off sufficiently to cause the RCS to begin cooling down even without OTSG heat transfer. HPI flow exceeds the break flow at this time.

Considering only mass and energy balances, the RCS could be refilled in several hours for this break. However, steam voids in the hot leg U-bend could delay refill and re-establishment of single phase natural circulation. Procedural guidance for bumping RC pumps would cause the system to refill. If the RCS repressurized to 2300 psig, the PORV would be opened as per procedures. This action will reduce RCS pressure until the CFTs activate. CFT flow in conjunction with HPI flow is sufficient to cause the RCS to cooldown and depressurize. Alternatively, opening the hot leg vents would allow refill of the system if the hot legs contained steam in the U-bend but subcooled water in the remainder of the hot leg piping.

The containment conditions for this class break are comparable (but less severe) than results shown for the .085 ft^2 LOCA.

3. TMI-2 Experience

Experience from the TMI-2 accident gives indication that the cable assembly can survive high humidity temperatures between 140 and 180F (Reference 9, "Equipment and Actuation Matrix") and chemical spray. Inspections performed on the TMI-2 cables indicate that fifty (50) thermocouples are known to have electrical continuity through the seal table to the internal area of the reactor vessel (References 2 and 7). The remaining two (2) thermocouple cables are in an unknown condition at this time with an open connection either at the seal table or internal to the vessel. This condition is being investigated. The cable and connection at the seal table are being removed for examination and testing (see "Qualification Plan" below). EG&G has concluded that the only evidence that the problem could be at the seal table (i.e. external to the vessel) is that the anomalies seen on these two cables are different from the other cables. All other thermocouples are known to be damaged inside the reactor vessel as a result of core damage. However, they also believe that the condition of these two cables could be an actual physical condition inside the reactor vessel. The incore thermocouple anomaly is similar to two SPND cables that are known to be in a damaged area having indications of being open and dry (while they are expected to be open and wet) (References 2 and 7). This further supports the suggestion that the two anomalous thermocouples may be damaged internal to the reactor vessel.

QUALIFICATION PLAN

GPUN is pursuing parallel paths concerning the incore thermocouple extension cable qualification. The first path includes testing of the extension cable between the seal plate connector and the containment penetration to verify qualification. Results of this test will be available by May 1985. This testing will include cable which was exposed to the TMI-2 accident environment. The second path involves the replacement of the extension wires with qualified cable should the existing cables fail and the existing connectors pass the test. The resolution of this item will be based on the results of the testing. A test specification is under preparation. The schedule for completion of testing of the extension cable is 30 weeks for procurement and fabrication, and 28 weeks for engineering and installation. Eighteen of those 28 weeks overlap with the procurement and fabrication schedule.

RESULTS

For large break LOCAs:

- The safety function is accomplished by procedural guidance and qualified instrumentation without reliance on the incore thermocouple assemblies.

For SB LOCAs and HELB inside containment:

- The safety function can be accomplished by alternate qualified instrumentation with existing procedural guidance.

- The safety function can be accomplished within the relatively short time (2-10 hours) and in a relatively benign containment environment using the incore thermocouple assemblies.

- The safety function can be accomplished based on inspections of the TMI-2 incore thermocouple assemblies which are similar to those at TMI-1.

CONCLUSIONS

The information provided in this discussion demonstrates safe plant operation with important to safety electrical equipment that has not yet been qualified in accordance with 10CFR50.49. Safe plant operation has been demonstrated by meeting the acceptance criteria of 10CFR50.49 which are addressed as objectives of this justification.

REFERENCES

1. R. C. Jones, J. Biller, B. M. Dunn, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS-Rev 3". July 1977. BAW 10103A. Babcock & Wilcox, Co. Lynchburg, VA.

2. R. Meininger to Distribution, "Examination of TMI-1 Incore Extension Cable and Seal Table Connector - RDM-58-84". November 7, 1984. EG&G. Middletown, PA.

3. James H. Taylor, Letter to S. A. Varga, US NRC. July 18, 1978. Babcock & Wilcox Co., Lynchburg, VA.

4. Babcock & Wilcox Co., "Abnormal Transient Operating Guidelines". Doc ID 74-1124158-00. April 6, 1983. Lynchburg, VA.

5. Calculation Number C-1101-220-5450-023, "CSMP Analysis of TMI-1 SB LOCA Response".

6. L. C. Lanese, "Containment Response to Small Break LOCAs". TDR 644. Draft.

7. TMI-2 Incore Instrumentation Damage - An Update GEND-INF-031, Volume II. April 1984. EG&G, Idaho Falls, Idaho.

8. Babcock & Wilcox Co. Evaluation of SBLOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W - Designed NSSS. February 1983. B&W Doc ID 77-1141270-00. Lynchburg, VA.

9. "Analysis of Three Mile Island - Unit 2 Accident", Nuclear Safety Analysis Center, NSAC-80-1. March 1980.

Justification for Interim Operation

1. Component

Victoreen Monitor Cable and Connector Assembly
Tag No. : None

2. Objective

The objective of this JIO is to demonstrate that:

2.1 There is partial test data that does not demonstrate full qualification, but provides a basis for concluding that equipment will perform its function.

2.2 The safety function can be accomplished by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

3.0 Existing Equipment Description

The cable assembly is consisted of the following components:

3.1

Device Type	Manufacturer	Model No.
Cable	Anoconda	FR-13AA
Cable Connector	Victoreen	Pt No. 877-1-58
High Temperature Software	Victoreen	Pt No. MS#A-0015
Potting Resin	Victoreen	Pt No. MS# J4375
Shrink Sleeving	Victoreen	Pt No. MS#H-6229
Nickel Seal	Victoreen	Pt No. 877-1-60
Adhesive, Duhesive No. 300	Victoreen	Pt No. MS# J4365
Pull Box	GPU field	None
Stainless Steel Hose	Swagelok	SS-12HU-6-S12
Heat Shrinkable tube/splice	Raychem	WCSF-N

4.0 Equipment Location

The cabie assembly is located inside the containment area 30 at 362-0" elevation. This cable assembly runs from the containment penetration to the Victoreen radiation detectors RMG-22 & 23 mounted on top the "D" rings. The normal radiological dose rate in this compartment is 10 to 100* MR/$_{HR}$ which results in a maximum integrated dose of 3.4×10^4 Rads over 40 years at aging temperature of 125°F. During the worst case accident, the postulated total integrated dose is 2.0×10^7, at a temperature of 275°F.

5.0 Equipment Function

This cable assembly delivers high voltage to the high range monitor ionization chamber (877-1) inside the containment building and provides return signals to the ratemeter mounted in panel PRF in the Main Control Room (MCR). The equipment performs no safety function but is used exclusively for post-accident monitoring.

However, this monitor does perform the following functions:

a. Provides indication which may restrictpersonnel access to the Reactor Building under high radiation conditions.

b. Provides limited gross radiation data following an accident involving high radiation release to the Reactor Building.

As an alternate over the short term, RM G8 is an 8 decade containment dome monitor equipped with a high range gamma sensitive ionization chamber detector located in the Reactor Building. Monitoring airborne activity thru 2inches of lead shields the detector and desensitizes it to dectect radiation levels from .01R/hr. up to $1x10^6$ R/hr. This monitor is powered from the ID vital bus. This ionization chamber detector is housed in a watertight container hermetically sealed to withstand the Reactor Building post accident pressure. This monitor reads out and alarms in the control room. Although this monitor has not been environmentally qualified, based on the accident at TMI-2, this type monitor provided the necessary functions in the post accident environment. TMI-1 and TMI-2 both have Victoreen Model 847-1 detector and 846-1 ratemeter for a dome monitor.

As an alternate over the long term, the Containment Atmospheric Post Accdient Sampling System (CAPASS) is capable of taking a Reactor Building atmospheric sample from which radioactivity level may be determined. (CAPASS) is located outside the reactor building.

6.0 Evaluation

GPUN calculation number C-1101-661-5350-010, illustrated that the subject cable (FR-13AA) is qualified by analysis for it's intended use. It is constructed of materials which have previously proved to be successful at surviving LOCA environment.

The connector at the detector end is installed in accordance to victoreen's recommended, tested connection. The connection of the cable at the penetration is installed using a qualified heat shrink tubing (Raychem WCSF-N), which has been used on many safety related circuits in the plant. Hence moisture intrusion at the connectors is unlikely. The cable is installed in pressure tight flexible hose between the detectors and the stainless steel pull box. The pull box was manufactured in accordance to the instructions provided by Victoreen. Steel conduit has been used as the raceway between the pull box and the penetration. Heat shrink tubing is applied over the connection and at the penetration.

The cable has passed submergence test, and possible cable failure caused by moisture intrusion accelerated by radiation is unlikely since it is demonstrated to withstand LOCA, by analysis. Although, this assembly has not been tested in this configuration, based on this evaluation, there is a high level of confidence that the installation will withstand the LOCA condition.

7.0 Equipment Qualification Program

The modified Time wire will be replaced with qualified triaxial cable. Delivery of the cable and connector at the detector end is expected in approximately 15 weeks after receipt of the purchase order. Installation would require a scheduled shutdown of 30 days.

8.0 Results

a) Based on the evaluation in Section 6.0 the equipment is qualified by analysis for its designated function.

b) The objective of the JIO has been satisfied by the evaluation

9.0 Conclusion

Safe plant operation with electrical equipment important to safety, as defined in 10CFR50.49, which has not yet been qualified has been demonstrated by the information provided in this JIO. This has been accomplished by meeting the acceptance criteria of 10CFR50.49.