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Writer's Direct Dial Number:

January 25, 1989
C311-88-2153

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
GPUN Comments on Inspection Report 88-16

The purpose of this letter is to comment on the Inspection Report's discussion and the conclusion reached regarding deficiencies in the non safety-related portion of instrumentation that provides automatic initiation of Emergency Feedwater (EFW) on a loss of both Main Feedwater (MFW) Pumps. We feel that the NRC's concern over these deficiencies as expressed in IR 88-16 appears to be overstated and somewhat out of context.

The current design of the automatic initiation of Emergency Feedwater (EFW) on a loss of both Main Feedwater (MFW) Pumps has been reviewed and approved by the NRC. The MFW differential pressure switches were existing plant equipment that was not upgraded as part of the EFW System upgrades. These instruments are located in the Turbine Building which is a non-seismic structure. Therefore, as part of the EFW System upgrade to safety-grade there was no technical merit in upgrading the system components which provide this input to the EFW initiation logic.

In order to maintain a safety-grade EFW System, it was necessary to install buffer isolation devices for the signal providing automatic initiation of EFW on a loss of both MFW pumps to ensure that any failure of this non safety-grade portion of the system not affect the safety-grade portion. Further upgrade of this equipment design would not result in a significant improvement in safety and could not result in upgrading the equipment to safety-grade seismic design criteria because of its location in the Turbine Building.

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The Restart SER (NUREG-0680 and NUREG-0680, Supplement 1) dated June 16, 1980 in Section 2.1.7 (page C8-35) discusses the acceptability of the loss of Main Feedwater initiation of EFW. The SER states that the MFW differential pressure switches could not be treated as fully safety-grade because they are installed in the Turbine Building and that these circuits would therefore be tied to the respective safety-grade EFW initiating circuits through buffer relays. The buffer relays were found to be acceptable isolation devices and the proposed design was therefore found to be acceptable.

The NRC's SER on EFW, dated February 18, 1987, further acknowledges the determination of acceptability for the design of the loss of Main Feedwater EFW initiation circuitry. Section 3.2.1 of the report attached to the NRC's SER states on page 7 that the EFW actuation on loss of MFW is existing auto start circuitry that was previously evaluated.

GPUN's System Design Description (SDD) for the EFW upgrades may have been taken to mean that the turbine building equipment would be further upgraded to seismic design requirements. The upgrade referred to in fact was intended to mean those portions of the system in seismic areas.

Upgrade of the Turbine Building equipment was never a commitment to the NRC. The NRC in its SER on the acceptability of the EFW upgrade to safety-grade did not find fault with the non-seismic Turbine Building portion of the system.

Section 4 of IR 88-16 discusses the EFW System upgrades. Two items were identified in this section as deficiencies. These two items pertain to the Main Feedwater differential pressure switches in the Turbine Building as follows:

1. Setpoint drift of the Main Feedwater differential pressure switches (DPS-542, 543, 829, and 830).

This issue was previously identified as an unresolved item in IR 87-06, and discussed again in IR 88-07. GPUN has increased the frequency of calibrating these instruments and we are currently performing analysis to support new setpoints which will take into account this drift and allow surveillances to be performed at the original surveillance interval.

During the exit meeting for IR 88-16, the inspectors acknowledged that the drift of the instrument setpoints was a separate issue unrelated to the instrumentation mounting clips that were found to be loose. However, the inspection report does not clearly indicate that these deficiencies are unrelated.

2. Instrumentation mounting attachments were found to be loose.

During a walkdown of the Main Feedwater differential pressure switch instrumentation in the turbine building, the NRC inspectors noted that this installation was not upgraded to safety-grade and that some of the attachments were not properly secured. During the exit meeting for IR 88-16 on September 2, 1988, GPUN committed to make the necessary repairs in order to bring this installation to a condition of proper workmanship in accordance with the intended design by the end of October, 1988.

In followup to this commitment GPUN performed a walkdown of the Main Feedwater differential pressure instruments on September 12, 1988 and prepared a structural evaluation of the discrepancies that were noted. An engineering evaluation was prepared and it was turned over to maintenance to do the work. Those repairs that could be accomplished without removing equipment from service were completed at that time. It was determined that some of the recommendations could only be accomplished by taking equipment out of service during a shutdown of sufficient duration to allow disassembling the components. This revised commitment schedule was discussed with the senior NRC resident inspector on October 26, 1988. The balance of the work was completed during the outage which occurred between December 15, 1988 and December 29, 1988.

The commitments made during the exit meeting were intended to bring this installation to a condition indicative of proper workmanship and were not intended as a commitment to backfit the design of this instrumentation. Upgrade of the Turbine Building equipment was not part of the NRC approved EFW System upgrades.

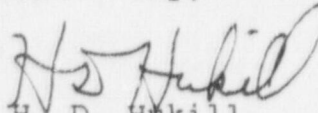
Failure of the equipment located in the Turbine Building will not result in a failure of the EFW System to initiate. Backup sensing systems associated with the Heat Sink Protection System (HSPS) will ensure that EFW initiation is completed. Also, the deficiencies that were found would not likely result in inadvertent initiation of EFW.

GPUN disagrees with the NRC's statement of concern that GPUN has not fully implemented and evaluated the EFW System upgrades as appears on page 1 of IR 88-16. We agree that the condition of mountings noted in IR 88-16 as Unresolved Item 88-16-01 were unacceptable and the repairs have been made to bring the

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installation to an acceptable condition. We do not however believe these deficiencies support the NRC's concern that the EFW upgrades are not fully implemented or evaluated. The basis for our disagreement is given above.

Sincerely,



H. D. Hukill

Vice President and Director, TMI-1

HDH/MRK

cc: J. Stolz
R. Hernan
R. Conte
W. Russell



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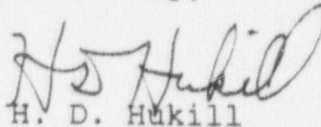
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Vice President and Director, TMI-1

HDH/MRK

cc: J. Stolz
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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-289/88-16

Docket No. 50-289

License No. DPR-50

Licensee: GPU Nuclear Corporation
P.O. Box 480
Middletown, Pennsylvania 17057

Facility Name: Three Mile Island, Unit 1

Inspection At: Parsippany, New Jersey and Middletown, Pennsylvania

Inspection Conducted: August 29 - September 2, 1988

Inspectors: *Thomas Koshy* 10-18-88
Thomas Koshy, Senior Reactor Engineer date

Roy K. Mathew 10-18-88
Roy K. Mathew, Reactor Engineer date

Approved by: *C. J. Anderson* 10/18/88
C. J. Anderson, Chief, Plant Systems Section date

Inspection Summary: Inspection on August 29-31, 1988 (Corporate Office)
September 1-2, 1988 (TMI-Site) - Inspection Report No. 50-289/88-16

Areas Inspected: This was an announced inspection to review the licensee's action on previously identified inspection findings.

Results: No violations or deviations were identified. Five unresolved items were closed. Two deficiencies were noted. The licensee has not fully implemented and evaluated the EFW system upgrade. The adequacy of the diesel generator capacity was not well supported in the licensee's plant loading calculations.

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The analysis concludes that the domestic water line above the back-up instrument air banks in the EDG building can withstand an SSE without falling and damaging the back-up air supply. A review was made of the licensee's seismic evaluation analyses of the EDG air ducts and air intake filter including its supports. This seismic evaluation was made by the licensee's engineering mechanics group under calculation numbers 1101 X dated May 22, 1980 and calculation number 1101 X dated May 22, 1980 and calculation number 1101 X -322C-A27 dated May 26, 1981. As a result this analysis additional supports and bracing were added to the ducting and the air filter. The licensee evaluation concludes that with the additional support in place in accordance with the details provided by the analyses, the EDG ducting and air filter do not constitute a missile hazard to the air banks during SSE. The inspector confirmed the additional support and bracing by a visual inspection.

This item is closed.

4.0 Emergency Feedwater System Upgrades

During this inspection, the NRC inspectors reviewed certain areas of the licensee's modification to upgrade the emergency feedwater system to a safety grade system. The EFW is designed to initiate on any of the following signals.

1. Low level in either OTSG
2. High Containment pressure
3. Main Feedwater Loss
4. Loss of reactor coolant pumps

The inspectors verified the installation of instruments, cable routing, trays, conduits for high containment pressure signal, a new signal and main feedwater loss signal, a previously existing signal to determine the adequacy of the cable routing and installation.

The high containment pressure signal instruments PT1186, 1187, 1188 and 1189 and its respective conduits, trays, cables up to heat sink protection cabinets were verified and found to be color coded and installed per GPUs SOD 772-A electrical cable and raceway routing criteria. However, the existing main feed water loss signal instruments DPS829, 542, 543 and 830 sensing lines, trays, conduits and cables were not upgraded. The licensee considers this to be a non safety related signal. At instruments DPS 829 and 542 the inspectors observed that one of the two mounting U bolts of the instrument had missing nuts, the tubing supports were missing, and some loose tubing was tied with loose wire to a conduit. The

inspectors reviewed the surveillance record in Procedure 1302-06.17 dated June 19, 1988. This record indicated a random drift of a setpoint in the instruments. The present condition of the instrument mounting and the cable routing for the loss of main feed flow signal for the emergency feedwater actuation system could lead to undue challenges to the safety system. The inspectors relayed these concerns to the licensee management. The licensee committed to implement corrective action by October 30, 1988. This is an unresolved item pending NRC review of the licensee action to improve the reliability of the loss of main feed flow signal (50-289/88-16-01).

5.0 Unresolved Items

Unresolved items are matters for which more information is required in order to ascertain whether they are acceptable, violations, or deviations. One unresolved item is discussed in Section 4.0 of this report.

6.0 Exit Interview

At the conclusion of the inspection on September 2, 1988, the inspectors met with the licensee representatives denoted in Section 1.0. The inspectors summarized the scope and findings of the inspection at that time.

No written material was provided to the licensee by the inspectors.

- a) The oscillation is momentary and has not caused the alarm. The level signal to the alarm is in the conservative direction causing, at worst, an early low-low level alarm
- b) The level error would not normally activate the low-low level alarm set point because the tank level must be maintained above the Technical Specification level of 11.2 feet (150,000 gallons)
- c) If the actual tank level drops below 9 feet during abnormal plant conditions and a premature low-level alarm occurs, the operator would have sufficient time to switch to the secondary source. This early transfer to the secondary condensate source is acceptable since it does not create an undue risk to safe plant operation. Both of these transient and steady state errors are in the conservative direction. They would not cause any substantial operational problems nor any safety concerns.

The inspectors verified the installation and concurs with the licensee's justification for utilizing the safety grade level transmitters.

This item is closed.

3.3 (Closed) Unresolved Item (289/86-12-17) Remote Shutdown Panel EFW Instrumentation Electrical Isolation from Control Room Panels and Seismic Qualification of EFW Digital Indications

During NRC Inspection 86-12 the licensee committed to provide electrical isolation between the power supplies to the EFW digital indicators on the remote shutdown panel and the control room panels.

This isolation was considered essential to prevent the loss of both indications in the event of a power supply problem in either of the locations for any reason including a seismic event. The inspector confirmed by a review of Gilbert Drawing 5130-B-600-509, revision 1D-0 dated October 27, 1986 that the power supply isolation design modification provides the required isolation. A review of the licensee installation confirmed that this modification has been installed and is operational.

During the NRC Inspection 86-12, the inspector noted that the electrical isolation would be of significant concern if the control room indications were not seismically qualified. A failure of the indicator due to a seismic event could affect the entire safety grade instrument loop. During the 86-12 inspection the licensee reported that the Weston Series 2470 indicators are seismically qualified. Their qualification was left as an unresolved item pending Region I review of the licensee qualification data package for these instruments.

The inspectors reviewed the Wyle Laboratories seismic qualifications test report 47430-1 Revision A dated October 3, 1984 for the Weston

Series 2470 Digital Panel Meter. This report concludes that "It was demonstrated that the specimen possessed sufficient integrity to withstand, without compromise of structure of electrical functions, the prescribed simulated seismic environment." No discrepancies were observed.

This item is closed.

3.4 (Closed) Unresolved Item (50-289/86-13-06) Seismic Qualification of Breaker Modification

The licensee modified the electro mechanical tripping device of Westinghouse DB-25 and DB-50 breakers with a Westinghouse Ampetector 1A solid state trip system. During a previous NRC Inspection, the inspector witnessed the breaker modification. However, the seismic qualification of the modification was not available for review.

During this inspection, the inspectors reviewed the seismic qualification report WCAP 10449 dated January 1984. This is a generic qualification report applicable to the solid state modification of the DB series of Westinghouse breakers. Westinghouse letter dated September 5, 1985 states that the particular mounting configuration utilized at TMI-1 is a modified version of the original mounting and that Westinghouse has analyzed this configuration as presented in drawing 437B596. They concluded that it is seismically qualified for the specified application. This modification provides better breaker coordination and repeatability of trip characteristics. The licensee modified 44 breakers in safety related applications and 77 breakers in the balance of plant applications.

This item is closed.

3.5 (Closed) Unresolved item (50-289/87-23-01) Evaluation of the Voltage Dip at the 4160 Volt Bus

On November 9, 1987, the output voltage of the "1B" auxiliary transformer (AXT) momentarily dipped. The "1B" AXT normally supplies one-of-two vital 4160 kv buses in addition to other non-safety buses/loads. The voltage dropped down to 2400 volts. However, the duration of the voltage drop was not long enough for the time delay relay to cause the associated emergency diesel generator to start. Various plant equipment responded to the voltage transient, such as alternate d.c. powered equipment starting. The main turbine experienced a runback of about 6MW (megawatts). As a result, reactor power dropped from about 99 percent to about 98 percent. The plant was restored to full power shortly thereafter.

The licensee review determined that the voltage dip was a result of one of the six circulating water pumps (CW-P-1F) for the secondary plant condenser experiencing an overcurrent situation. The circulating water pump motor is protected by instantaneous and time