U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Report No. 50-289/85-20

Docket No. 50-289

License No. DPR-50 Priority -- Category C

Licensee: GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

Facility At: Three Mile Island Nuclear Station, Unit 1

Inspection At: Middletown, Pennsylvania

Inspection Conducted: June 28, 1985 - August 2, 1985

Inspectors: D. Haverkamp, Technical Assistant for TMI-1 Restart, Region I
R. Urban, Reactor Engineer, Region I
P. Wen, Reactor Engineer, Region I
F. Young, Resident Inspector (TMI-1), Region I

Approved By: Richard J. Urban 8/26/85
For R. Conte, TMI-1 Restart Manager Date
TMI-1 Restart Staff
Division of Reactor Projects

Inspection Summary:

Routine safety inspection (180 hours) of hot shutdown plant activities in preparation for TMI-1 restart; licensee action in response to events occurring at other reactor facilities including emergency feedwater (EFW) system steam binding and mispositioned control rods; operability of TMI task action plan equipment modifications; modification control program improvements including updated status on EFW modifications; tube plugging in steam generators; pressurized thermal shock flux reduction program; licensee action on 10 CFR 21 reports; examination security; control room habitability ventilation system test; licensee action on previous inspection findings; and overall restart readiness.

Inspection Results:

Licensee upper management continued their detailed involvement in site activities. Personnel properly implemented facility procedures for hot shutdown evaluations and testing. Licensee personnel took appropriate actions through testing, procedural changes, and training in response to events at other reactor facilities, i.e., steam binding in the emergency feedwater (EFW) system and mispositioned control rods. The installed TMI Action Plan equipment was operable although some discrepancies existed with Technical Specification covering this equipment. The licensee continued to implement initiatives to improve the modification control program. Reasonable progress continued on modification work to upgrade the EFW system to full safety grade status. The licensee design documents reflected restart hearing-imposed safety design objectives. The licensee took appropriate action on issues addressed in an NRC safety evaluation; various 10 CFR 21 reports, and previous inspection findings including two violation responses. The plant remains physically ready for restart, however, the reliability of one of two source range channels for nuclear instrumentation needs to be evaluated as to its impact on a safe restart of TMI-1.

DETAILS

1.0 Introduction

At the beginning of this inspection period on June 28, 1985, the plant was at normal hot shutdown conditions (about 530°F and 2150 psig) to complete licensed operator familiarization training, pending further action by the U.S. Court of Appeals for the Third Circuit in Philadelphia, Pennsylvania. As discussed in NRC Region Inspection Report Number 50-289/85-19, the Court of Appeals on June 7, 1985, stayed the Commission Restart Order (CLI-85-09). No further court action has been taken and the facility remained at hot shutdown for the remainder of the inspection period. Inspection coverage was provided by the resident inspector and support staff components of the NRC TMI-1 Restart Staff.

2.0 Plant Operations During Hot Shutdown

2.1 Routine Review

The resident inspectors periodically inspected the facility to determine the licensee's compliance with general operating requirements of Section 6 of the Technical Specifications (TS) in the following areas:

-- review of selected plant parameters for abnormal trends;

-- plant status from a maintenance/modification viewpoint including plant housekeeping and fire protection measures;

-- control of ongoing and special evolutions, including control room personnel awareness of these evolutions;

-- control of documents including log-keeping practices;

-- implementation of radiological controls; and,

-- implementation of the security plan including access control, boundary integrity and badging practices.

The inspectors focused on the following areas:

-- control room operations during regular and backshift hours including frequent observation of activities in progress and periodic reviews of selected sections of the shift foreman's log and control room operator's log and selected sections of other control room daily logs;

-- areas outside the control room; and,

-- selected licensee planning meetings.

The inspectors identified no conditions adverse to nuclear safety or inconsistent with regulatory requirements.

2.2 Summary of Findings

Overall, personnel stationed in the control room exhibited control of daily activities, including problem areas that needed resolution. Licensee planning meetings stressed attentiveness to proceed safely with daily activities, including surveillance and maintenance and to resolve any inter-departmental interface problems. Licensee upper management continued their detailed involvement in site activities.

3.0 Licensee Actions in Response to IE Information Notices

The inspector reviewed the licensee's action associated with IE Information Notice (IN) 84-06, "Steam Binding of Auxiliary Feedwater Pumps," and IN 83-75, "Mispositioned Control Rods." The inspector also reviewed the corresponding Institute of Nuclear Power Operations (INPO) Significant Operating Experience Reports (SOER).

3.1 Steam Binding in Emergency Feedwater System

The inspector reviewed the licensee's current methodology on identifying steam back-leakage into the emergency feedwater (EFW) system. The inspector noted, at the beginning of the inspection period, that the licensee was training their auxiliary operators in a general manner to determine if there were any abnormal plant conditions in the intermediate building. It was not apparent how the specific problem of back-leakage in the EFW system was addressed in training sessions.

Licensee representatives initially indicated that if steam back-leakage into the EFW system occurred, this would probably be noticed during the monthly surveillance check of the EFW pumps. Prior to this time, the licensee had performed a one-time test during the last hot functional test (HFT) in April 1985 to determine if such back-leakage occurred. Temperature readings at specific EFW piping locations were taken over an eight hour period. These test results indicated no back-leakage into the system. In addition, the licensee inspected certain EFW valves prior to this HFT and they found no significant wear or wastage of valve internals.

After discussion of the problem with the inspector, the Plant Operations Manager initiated a procedure change request (PCR) that added a requirement to auxiliary operator shift logs to check for back-leakage into the EFW system. In addition, he wrote a shift briefing sheet to re-emphasize the safety concern about steam binding. Operations personnel reviewed the shift briefing sheet.

The inspector concluded that the licensee actions were adequate measures to detect back-leakage into the EFW system if it occurred.

3.2 Mispositioned Control Rods

Based on discussions with the TMI-1 lead nuclear engineer, the inspector reviewed the licensee's actions regarding rod mispositioning and how they would recover from this type of problem. The inspector discussed the rod misalignment event that occurred at ANO-1 to ensure that the lessons learned from that event were incorporated into station procedures. The engineer demonstrated how their procedures addressed this concern.

The licensee conducted specific simulator training that addressed mispositioning of a control rod. A review of the lecture guides and other training material showed that there were specific and defined methods on how to recover a misaligned or dropped rod. The inspector also reviewed the licensee's training plans used during simulator training and, independently, reviewed applicable station procedures.

The inspector concluded that information presented was detailed enough that an operator could develop a working knowledge of the core physics aspects of a misaligned rod. Review of station procedures indicated that necessary guidance was reflected in the applicable procedures.

4.0 Operability of TMI Task Action Plan Equipment Modifications

The purpose of this review was to assess, on a sampling basis, the operability status of completed TMI Action Plan (TAP) required equipment and procedural controls modifications specified in NUREG 0737. The review consisted of evaluating the present status of installed plant equipment, modifications, and any program/procedure requirements to determine if the TAP requirement was met on a continuing basis. The TAP items reviewed included I.A.1.2, I.C.3, I.C.6, I.A.1.3, II.B.3, II.D.3, II.F.2, II.E.1.1, II.E.4.2 and II.G.1.

As part of this review, the inspectors referred to NUREG 0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident;" NUREG 0737, "Classification of TMI Action Plan Requirements," TMI Unit 1 Technical Specifications (TS); and the NRC resident office files related to TAP correspondence.

The inspectors determined which TAP items were applicable and if any plant equipment, modifications, or TS changes were made as a result of each TAP item. All TS associated with the TAP items were reviewed for adequacy and completeness.

The inspectors also walked down various systems throughout the plant such as emergency feedwater equipment, controls and indication, and inadequate core cooling instrumentation.

The inspectors identified no conditions adverse to nuclear safety or inconsistent with regulatory requirements. The licensee completed modifications and related technical specification (TS) changes (approved by NRR) in accordance with NRC staff requirements and guidelines. The inspectors identified certain existing TS that might be improved by additional changes in wording and/or format; but, in all instances, the inspectors considered the TS to be enforceable in assuring operability of related equipment. Further, the TS, as approved, were consistent with applicable NRC staff generic letter guidelines. The inspectors' suggested TS improvements will be reviewed by Region I for generic applicability and forwarded to NRR, if applicable.

5.0 Modification Control Program - Selected Aspects

At the corporate office (Parsippany, New Jersey), the resident inspector reviewed selected aspects of the modification control program. He also obtained a status on the upgrade modifications for the emergency feedwater system.

5.1 Modification Task Force Improvements

The licensee responded to the latest NRC Systematic Assessment of Licensee Performance (SALP) (NRC Inspection Report 50-289/85-99) by letter dated May 7, 1985, from H. Hukill, Director, TMI-1, to T. Murley, Region I Regional Administrator, and specifically addressed improvements in the Design, Engineering, and Modification functional area. They reported that the Office of the President approved the GPUN Modification Task Force (Task Force) recommendations discussed in the SALP. During this inspection period, the inspector discussed the status of the Task Force recommendations with the GPUNC Director of Engineering, Director of Licensing, and Manager, TMI-1 Long Range Planning.

The inspector learned that a documented update to these recommendation was to be prepared and he determined that reasonable progress is being made to resolve the concerns embodied in the Task Force report. Of particular interest to the inspector were actions related to the Task Force observation that too much work (beyond resource capability) was planned for a given outage period.

The inspector determined that licensee management took direct steps for better outage planning. They created two new planning positions that now report to the Vice President of Technical Functions (for Oyster Creek and TMI-1). Modification items were placed on a computerized list which was to identify the regulatory source organization (such as NRR, IE), the particular outage and cycle during which they planned to implement the item, responsible engineer, and estimates of ALARA exposure in addition to staff-hour estimates for

engineering and construction. The computerized list for TMI-1 was to receive inputs from various GPUNC divisions for complete planning information and assurance that regulatory and corporate schedule objectives were met. Overall, the inspector concluded that these positive licensee initiatives, if properly and fully implemented, could serve to enhance reactor safety. These measures would assure a methodical outage implementation with the avoidance of last minute engineering changes for safety-related system modifications.

Further, during the above discussions, the inspector determined that licensee management planned to issue a consistent set of engineering specifications, installation standards, and inspection standards to assure an up-front and consistent engineering implementation, and verification of regulatory requirements and industrial standards for all modifications including those affecting safety systems. This is a long-term improvement item.

5.2 Preliminary Engineering Design Review

The inspector reviewed Technical Functions Procedure 5000-ADM-7311.03 (EMP-014), Revision 1-00, effective April 12, 1985, "Project Reviews" as it related to the conduct of Preliminary Engineering Design Reviews (PEDR) meetings. At the site, on July 18, 1985, the inspector monitored a PEDR (second meeting for operations department representatives) on modifications for 10 CFR 50, Appendix R. Also, at the corporate office on July 24, 1985, the inspector monitored a PEDR (the last in the series of several PEDRs) on modifications to install an engineered safety features ventilation system for the fuel handling building.

On a sampling basis, the inspector verified proper implementation of EMP-014 with respect to PEDR requirements. Licensee attendees provided their expertise and experience to pose thoughtful and challenging questions to the design engineers. Various company disciplines were represented including site plant engineering, site radiological engineering, site maintenance and construction planning, in addition to corporate engineering and quality assurance representatives. The PEDR chairman controlled the meeting and ensured that safety concerns, regulatory requirements, industrial standards, and practical site-specific concerns were adequately addressed in licensee design documents (primarily system design descriptions and safety evaluations).

Overall, the meetings showed continued implementation of this resource-intensive licensee initiative and appeared to be focused toward producing quality products that directly enhance plant safety.

5.3 Emergency Feedwater System Upgrade

5.3.1 Introduction

As a result of the TMI-2 accident, the NRC ordered TMI-1 shut down because it did not have "...the requisite reasonable assurance that the same licensee's Three Mile Island Unit 1 can be operated without endangering the health and safety of the public" (Commission Order, dated July 2, 1979). In a Commission Order, dated August 9, 1979, the Commission specified that the following licensee actions must be taken with respect to the emergency feedwater (EFW) system:

-- upgrade the timeliness and reliability of the EFW in accordance with licensee proposed actions, in letter dated June 28, 1979;

-- develop and implement operating procedures for initiating and controlling EFW independent of integrated control system (ICS);

-- complete analysis of potential small breaks, and develop and implement instructions to define operator actions; and,

-- provide reasonable assurance of the safety of long term operation with outstanding category B items of NUREG 0578 (later became Task Action Plan Items II.E.1 and II.E.1.2 of NUREG 0737).

The short term licensee actions were verified as a part of the staff's restart certification process to the Commission (SECY 85-192, May 29, 1985). Additional licensee actions taken were as a result of Union of Concerned Scientist (UCS) 2.206 Petition in early 1984. The NRC staff reviewed and verified these actions as noted in NRR Director Decisions 84-12, dated April 27, 1984, and 84-22, dated September 25, 1984; and NRC Inspection Reports 50-289/84-21, 84-22, and 84-38. The long-term licensee actions with respect to upgrading the EFW system to full safety grade status are summarized in paragraph 5.3.3.

5.3.2 Scope of Review

The purpose of this review was to update the status of EFW long-term modifications and verify that the licensee incorporated NRC-imposed design objectives into licensee design packages/documents for subsequent plant installation. The inspector reviewed applicable licensing and appeal board decisions and related staff safety evaluations to identify the outstanding long term items. He also reviewed applicable licensee letters and licensee internal design documents.

5.3.3 Detailed Status

Listed below are the applicable design documents for various EFW upgrade modifications/or licensee evaluation along with status of construction and testing.

5.3.3.1 Mechanical System Modifications

5.3.3.1.1 Add cavitating venturis (and vibration supports) in EFW discharge piping (289/83-BC-16)

References

(1) Atomic Safety and Licensing Board (ASLB) Partial Initial Decision (PID),on the Restart Hearing, dated 12/14/81, paragraph 1037, item No. 1
(2) NUREG 0680, NRC Staff TMI-1 Restart Safety Evaluation Report and Supplement 3, Order Item 8-2.1.7.(a), Item No. 1
(3) System Design Description (SDD)-1-424B, Division (Div) 1, Revision (Rev.) 4, Item 1.1.1
(4) Licensee Letter (LL) (5211-85-2057), dated April 19, 1985, from H. Hukill, TMI-1, to J. Stolz, NRC, Enclosure (Encl.) 1, Item 1.3.1

This was completed for restart and verified in NRC Inspection Reports 50-289/82-26, 83-01, 83-12, 83-14, and 84-01. This was considered by the staff to be a long-term modification but it was relied upon by the staff to limit EFW flow to an affected once through steam generator (OTSG) on main steam/feedwater line rupture to resolve an ASLB concern. The concern was that inadvertent actuation of the then non-safety grade portion of the steam line rupture detection system would isolate EFW. The isolation function was removed by the licensee and it was verified in NRC Inspection Report 50-289/83-01, thereby resolving this concern.

5.3.3.1.2 Provide redundant safety grade EFW control and block valves (289/83-BC-01 and 03)

References

(1) ASLB PID, dated 12/14/81, paragraph 1036
(2) NUREG 0680, and Supplement 3, Order Item 8-2.1.7.a(2)
(3) SDD-424B, Revision 4, Item 1.1.2
(4) LL of April 19, 1985, Enclosure 1, Item 1.3.2

The mechanical portion of this modification is complete. Electrical work is controlled by the installation of the heat sink protection panel, the status of which is addressed in paragraph 5.3.3.5 of this report.

5.3.3.2 Structural Modifications

References
(1) SDD-424B, Div. 1, Rev. 4, Items 1.2.1, .2, .3
(2) LL of April 29, 1985, Encl. 1, Items 1.3.3, .4, and .5
(3) LL of February 13, 1985, from H. Hukill, GPUNC, to J. Stolz, NRC, Correction to NRC Inspection Report 50-289/84-37

These modifications were: upgrade of EFW pumps recirculation lines to Seismic Category I (as described in reference (3)); upgrade vent stacks for safety valve and atmospheric dumps to Seismic Category I; and provide increased flood protection in the intermediate building for a main feedwater line break. These modifications were completed and were verified by NRC Region I in response to the USC 2.206 Petition of 1984 and subsequent NRR Director Decisions.

5.3.3.3 Electrical Modifications

5.3.3.3.1 Provide a safety grade power supply to valves CO-V111 A/B and upgrade cable routing for power supply to valve CO-V V14 A/B

Reference
(1) LL 83-232, dated August 23, 1983, in response to TAP II.E.1.1 (Section IV.B.1, page 5)
(2) SDD-424B, Div. 1, Rev. 4, Item 1.3.1
(3) LL of April 29, 1985, Encl. 1, Item 1.3.6

These valves have the safety function of isolating a damaged condensate storage tank (CST) (CO-V111A/B) or isolating non-safety systems from the EFW system (CO-V14A/B). This work is related to the extensive cable and conduit work required for meeting safety-grade criteria and meeting 10 CFR 50, Appendix R. The status of the cable and conduit effort is addressed in paragraph 5.3.3.5 of this report.

5.3.3.3.2 Delete cross connect between electrical busses that allows an operator to load both EFW electric driven pumps on a single diesel generator

References
(1) LL 83-232, dated August 23, 1983, in response to TAP II.E.1.1 (Section IV.B.2, page 6)
(2) SDD-424B, Div. 1, Rev. 4, Item 1.3.8
(3) LL of April 29, 1985, Encl. 1, Item 1.3.18

Reference 1 reported this item complete from a human factors viewpoint, that is, operator error causing a diesel generator overload. In a later design document (reference (2)) the licensee relied on this modification to fulfill electrical separation criteria for redundant electrical systems. The licensee completed this modification prior to the restart hearing in 1980 by Engineering Change Memorandum S-225. The inspector reviewed licensee records on this modification and he examined control room panels for the control of EFW electric-driven pumps. The inspector concluded that the licensee properly deleted the subject power cross-connect function from EFW pump control.

5.3.3.3.3 Review of diesel generator bus loadings to assure no overload situation exists as a result of system modifications

References
(1) LL 83-232, dated August 23, 1983, in response to TAP II.E.1.1 (Section IV.B.3, page 6)
(2) LL 84-2304, dated January 11, 1984, on computer program for diesel generator bus loadings

As committed to in reference (1), reference (2) documented the satisfactory results of the licensee's computer analysis of diesel generator bus loadings as a result of modifications made to the facility. The licensee concluded that no overload situation would exist.

These documents will be reviewed by NRR as a part of TAP II.E.1.1 and .2 reviews.

5.3.3.4 Instrument and Control Modifications

5.3.3.4.1 Deletion of the main steam line rupture detection system (MSLRDS) signal to emergency feedwater control valves (289/83-BC-10)

References
(1) ASLB PID, dated 12/14/81, paragraph 1064
(2) ALAB 729, dated 5/26/83, pages 35 and 176
(3) LL 82-153, dated August 2, 1982
(4) NRC letters (NRR), dated November 10, 1982 and August 30, 1983

(5) Commission Memorandum and Order CLI 84-11, dated July 26, 1984
(6) NRC Inspection Report 50-289/83-01
(7) SDD-423B, Div. 1, Rev. 4, Item 1.3.2
(8) LL of April 29, 1985, Encl. 1, Item 1.3.17

In reference (1), the ASLB raised the concern of the non-safety grade MSLRDS inadvertently isolating EFW. In reference (2) the ALAB noted that the licensee's proposed resolution (deletion of the MSLRDS signal to EF-V 30A/B) should be reviewed by the Commission after NRC staff review. Reference (3) documented the licensee's proposed resolution. In reference (4) NRC staff accepted the licensee's proposed resolution and forwarded their review to the Commission. In reference (5) the Commission also accepted the licensee's resolution of the ASLB concern. The NRC staff verified completion of licensee action as documented in reference (6). References (7) and (8) incorporated the design objectives into licensee design documents.

5.3.3.4.2 Provide safety-grade automatic initiation and control of EFW (289/83-BC-01 and 06)

References
(1) ASLB PID, dated 12/14/81, paragraph 1036
(2) SDD-423B, Div. 1, Rev. 4, Items 1.3.3, 1.3.4 and 1.3.5
(3) LL of April 29, 1985, Encl. 1, Items 1.3.9, 1.3.11, 1.3.14, and 1.3.16

The EFW auto initiation restart modifications were to be retained and these modifications included auto initiation of EFW on loss of both main feedwater pumps or on loss of all four reactor coolant pumps. Planned modifications were automatic initiation of EWF on high containment pressure and low steam generator water level.

The controlling work in these planned modifications is the installation of the heat sink protection system panels which contain the logic actuation sub-systems for EFW initiation. Licensee design requirements include safety-grade criteria for the initiation system. The status of the HSPS installation is addressed in paragraph 5.3.3.5 of this report.

The control functions will remain similar to that committed to in the restart hearing, that is, EF-V 30A/B actuation to maintain level in the OTSG startup range with reactor coolant pumps on or in the operating range (higher level) on loss of reactor coolant pumps to assure natural circulation. The safety-grade redundant block and control valves for each OTSG will be controlled by the HSPS logic system.

5.3.3.4.3 Provide safety grade OTSG level instrumentation with signal to initiate EFW and isolate MFW on high water level in the OTSG (289/83-BC-08 and 18)

References
(1) ASLB PID, dated 12/14/81, paragraph 1037, Item No. 4
(2) NUREG 0680 and Supplement 3, Order Item 8-2.1.7.a(3)
(3) SDD-424B, Div. 1, Rev. 4, Item 13.4
(4) LL of April 29, 1985, Encl. 1, Item 1.3.10

Instrument transmitters are installed and the remaining work is being accomplished with HSPS installation.

Licensee design documents (references (3) and (4)) reflect design objectives as noted in references (1) and (2).

5.3.3.4.4 Upgrade MSLRDS to safety grade to assure isolation of MFW and prevent a potential overpressurization of containment on steam line break in containment (289/83-BC-09)

References
(1) ASLB PID, dated 12/14/81, paragraph 1037, Item No. 5
(2) ALAB, 7-29, dated 5/26/83, page 36
(3) SDD-424B, Div. 1, Rev. 4, Item 1.3.6
(4) LL of April 29, 1985, Encl. 1, Item 1.3.12

Safety grade logic actuation is provided by HSPS panels. The HSPS and related cable and conduit installation is addressed in paragraph 5.3.3.5. The design objectives, as noted in references (1) and (2), are incorporated into licensee design documents, references (3) and (4).

5.3.3.4.5 Provide safety-grade condensate storage tank level installation and low water level alarm (289/83-BC-07 and 19)

References
(1) ASLB PID, dated 8/14/81, paragraph 1037, Item No. 2
(2) NUREG 0680 and Supplement 3, Order Item 8-2.1.7.a, Item No. 5
(3) SDD-424B, Div. 1, Rev. 4, Item No. 1.3.7
(4) LL of April 29, 1985, Encl. 1, Item No. 1.3.13

This work is part of the cable and conduit effort. The design objectives of references (1) and (2) are incorporated into the licensee's design documents (references (3) and (4)).

5.3.3.4.6 Provide safety grade OTSG high level alarm (289/83-BC-13)

References
(1) ASLB PID, dated 8/14/81, paragraph 1037, Item No. 3
(2) NUREG 0680 and Supplement 3, Order Item No. 8-2.7.1.a, Item No. 7
(3) SDD-424B, Div. 1, Rev. 4, Item No. 1.3.4
(4) LL of April 29, 1985, Encl. 1, Item No. 1.3.10

The design objectives were incorporated into the design of new OTSG water level instrumentation as noted in paragraph 5.3.3.4.3 above.

5.3.3.4.7 Other licensee proposed modifications/actions

References
(1) SDD-424B, Div. 1, Rev. 4, Items 1.3.9.10, 1.5.3
(2) LL of April 29, 1985, Encl. 1, Items, 1.3.7, .8, .20

The following additional modifications were proposed and are being implemented by the licensee:

-- overspeed trip alarm for the turbine-driven EFW pumps;

-- safety grade pit level (flood detection) alarm for intermediate building and control-grade condenser hotwell low level alarm; and,

-- evaluate performance of electric and instrument control cables in the event of flooding in the intermediate building.

These proposals are under review by the NRC staff in conjunction with SER development for TAP II.E.1.2, Auto Initiation of EFW.

5.3.3.5 Summary and Conclusion

The controlling work effort is cable and conduit installation along with HSPS cabinet installation and wire termination. This work involves extensive resources and is delayed, in part, for procurement and receipt of qualified material. Extensive preoperational and startup testing is planned. If TMI-1 restarts, then system tie-ins and testing will be delayed.

However, the cycle 6 startup (first refueling after restart) commitment to the NRC should be met.

Based on the above review, the inspector concluded that NRC-imposed safety design objectives, as a result of the restart hearing, were properly incorporated into licensee design documents.

The proper implementation of the safety-grade design requirements is unresolved pending completion of licensee action and subsequent NRC Region I review (289/85-20-01).

6.0 Operation of TMI-1 with 2,000 Plugged Steam Generator Tubes

In NUREG 1019, the NRC staff evaluated the licensee's analyses on the effects of operating TMI-1 once through steam generators (OTSG) with 1,500 tubes plugged. The staff found that transient and accident consequences resulting from operation with 1,500 tubes plugged were bounded by the FSAR analyses, and therefore subsequent operation was acceptable.

To date, a total of 1,542 OTSG tubes have been plugged. To support this additional plugging, the licensee provided to the NRC resident inspector, TDR No. 674, Revision 1, "Comparison of Steam Generator Tube Plugging with the TMI-1 Design Basis." This document stated that plugging up to 2,000 tubes will not adversely affect plant operation and is still bounded by the FSAR. The licensee plans to officially submit TDR No. 674 (current revision) to the NRC.

At the request of NRC Region I, the NRC Office of Nuclear Reactor Regulation (NRR) reviewed and evaluated the licensee's analysis. The NRR staff's safety evaluation (SE), attached to this report, confirmed that operation with up to 2,000 plugged tubes does not involve an unreviewed safety question and the conclusions in NUREG 1019 remain valid for up to 2,000 tubes. In addition, the SE stated that the upper number of 2,000 tubes was acceptable as long as the plugging ratio between OTSGs does not exceed a 3 to 1 ratio.

The inspector discussed the SE with appropriate licensee representatives. The inspector stated that if the licensee, at a later date, was required to plug in excess of 2,000 tubes or exceed the 3 to 1 plugging ratio, an additional evaluation would be required to be performed per 10 CFR 50.59 in order to return the unit to operation. No further licensee action regarding this matter is required at this time.

7.0 Pressurized Thermal Shock Flux Reduction Program

A concern on the capability of pressurized water reactor pressure vessels to withstand a severe pressurized thermal shock (PTS) without compromising reactor vessel integrity was under intensive examination by the NRC. A neutron flux reduction program was proposed by the licensee to reduce neutron-induced radiation embrittlement of the reactor vessel. The NRC reviewed the licensee's flux reduction program and concluded in its safety

evaluation that the licensee adequately addressed this issue (NRC letter dated March 14, 1985, from J. F. Stolz, NRC to H. D. Hukill, GPUNC). The NRC's conclusion was based on the licensee's plan to implement a low-leakage fuel loading scheme in future cycles of operation.

Based on discussions with the site lead nuclear engineer and later confirmed by a GPU Headquarters nuclear fuel engineer (through telephone conversation), the inspector noted that the low-leakage fuel loading scheme (in-out-in strategy) is planned for Cycle 6 reload. However, due to the long delay in Cycle 5 restart, the actual Cycle 6 fuel design was not initiated. The proper design and implementation of the flux reduction scheme for Cycle 6 reload is unresolved pending completion of licensee action and subsequent NRC Region I review (289/85-20-02).

8.0 Part 21 Report Followup

The inspector reviewed the below noted 10 CFR 21 and 10 CFR 50.55(e) Reports to ascertain the nature of the problems (deficiencies) as related to TMI-1. Subsequently, he reviewed licensee corrective actions to ascertain if the licensee received complete and appropriate information from the applicable vendor and if licensee corrective actions were adequate to resolve the deficiency consistent with vendor recommendations.

8.1 Small Break Operating Guidelines (SBOG)

Reference
B&W letter from J. H. Taylor to R. C. DeYoung, dated July 29, 1983 (Part 21 Report)

The original SBOG did not deal to any great extent with the pressurized thermal shock issue. As a result, it is possible to have misused or misinterpreted a statement contained in the SBOG. The reference letter clarified the ambiguities, specifically the repressurization restriction following the RCS cooldown below 500°F not to exceed a rate greater than 100°F/hr. The inspector reviewed the Abnormal Transient Procedure 1210-10, Figure 1, and noted that the PTS concerns and proper RCS cooldown rates were clearly included in the procedure. The inspector had no further questions.

8.2 HPI Throttle Valves

References
(1) Letter from G. R. Westafer (Florida Power Corporation) to J. P. O'Reilly (NRC Region II), dated June 27, 1983 (Part 21 Report)
(2) IE Information Notice No. 80-48 and Supplement 1

The failure of throttling HPI valves and similar failures involving Rockwell International globe valves were reported to the NRC. The licensee uses 2 1/2" Rockwell International globe valves in the HPI

lines at TMI-1 for throttling purposes (MU-V 16 A through D). The inspector discussed this subject with the licensee mechanical engineering representative and he learned that the licensee was aware of the problem encountered at the other sites. This is evidenced in the completion of the licensee's Licensing Action Item No. 84-9519.

As a result of the licensee's evaluation, a procedure note cautions against backseating valves with torque switches set in accordance with Corrective Maintenance Procedure 1420-LTQ-1, "Limitorque Operator, Limit Switch Adjustment," Revision 8. The inspector further reviewed the machinery history report for MU-V 16A through D, and found no similar deficiencies were ever recorded in the plant history. The inspector also reviewed the high pressure injection flow test results (SP 1303-11.8) performed on April 11-12, 1985, and noted that no unacceptable conditions were identified. The inspector concluded that the licensee either had or had taken reasonable measures to assure operability of the HPI throttling valves.

8.3 D.C. Batteries

Reference
Letter from W. P. Murphy (Vermont Yankee Nuclear Power Corporation) to T. E. Murley (NRC, Region 1), dated June 29, 1984 (Part 21 Report)

A potential deficiency involving corrosion in the lead posts of the batteries supplied by Exide Corporation was reported for Vermont Yankee. Through discussions with the licensee's cognizant representative, the inspector learned that the station batteries were supplied by C&D Corporation. Appropriate preventive maintenance has been implemented to ensure the operability of the station batteries. The inspector physically walked down the 'A' and 'B' battery rooms and noted that battery posts were clean with no crud buildup, and no cracking or negative plate discoloration. The licensee had previously noted cell cracking at TMI-1 but reasonable measures are in place to assure the timely detection of cracks before they would affect the operability of the battery bank. Battery bank replacements are planned for a future refueling outage.

8.4 Hydrogen Recombiner

8.4.1 References

(1) Letter from D. C. Empey (Rockwell International) to U. Potapovs (NRC, Vendor Inspection Branch), dated December 15, 1981
(2) Letter from D. C. Empey to J. Collins (NRC, Region IV), dated May 5, 1983 (Part 21 Report)

(3) Letter from D. C. Empey to J. Collins, dated May 27, 1983 (Part 21 Report)
(4) Letter from D. C. Empey to J. Collins, dated December 19, 1983 (Part 21 Report)
(5) IE Information Notice No. 85-08, Item 3
(6) IE Information Notice No. 83-72, Items 17 & 18

8.4.2 Review/Findings

As a result of environmental qualification testing experience, several concerns related to the hydrogen recombiner components were reported by Rockwell International (Energy Systems Group). The following describes each concern and the licensee's response/action:

8.4.2.1 Viton Seals

Viton seals were used at the recombiner inlet and outlet pipe-to-blower housing flanges, and at the blower flowmeter. The material's sealing capability may be degraded due to exposure to radiation, elevated temperature, and steam environments. The licensee has evaluated this problem (Memorandum from S. U. Zaman to D. Shovlin/R. Knight, dated May 24, 1985). As a result, new qualified seals have been ordered and the replacement has been scheduled on an annual basis.

8.4.2.2 Leadwire Insulation

The original heater leadwire routing areas may reach temperatures higher than the leadwire insulation design rating (194°F). During the accident conditions, this may result in a reduced service for the leadwire insulation. The licensee corrected this problem by installing a new qualified leadwire per job ticket No. CA075 on January 24, 1983.

8.4.2.3 Time Delay Relay and Circuit Breaker

The subject components failed the vendor's environmental qualification test. However, these components at TMI-1 are located within the intermediate building and are not exposed to a postulated LOCA condition. The test results were not applicable to the qualification of these components.

8.5 Conclusion

The licensee has taken proper corrective action in response to the various 10 CFR 21 Reports, IE Information Notices, and Industrial Experience Reports noted above.

9.0 Security of TMI-1 Operator Examinations

Additional review of the subject matter was conducted as a result of Three Mile Island Alert's (TMIA) appeal to the Atomic Safety and Licensing Appeal Board subsequent to the issuance of NRC Inspection 50-289/85-12.

The incident involved the discovery of a microfiche copy of a TMI-1 auxiliary operator examination in the TMI-2 parking lot. Subsequent investigation determined it to be a record of a completed examination that had been administered about a year prior to its discovery. In accordance with the licensee's procedures, examination security requirements for Category 1 examination materials applies to the period of time when an examination is prepared, administered and graded. Once this process is completed, the materials become a record and are not considered to be Category I materials.

Based on discussions with licensee personnel and a review of applicable procedures, the inspector determined that completed examinations are controlled until the administration of examinations to all operators is completed. Copies of the examinations are subsequently decontrolled and made available to the operators if requested.

Previously administered examinations are produced from an examination question bank and the requirements for the selection of questions from the bank are such that the contents of a new examination are not identical to any previous examination. This assures that the potential for any particular examination containing a substantial number of the same questions as a previous examination is extremely remote. Nothing would preclude an operator, once in receipt of a graded examination, from generally distributing it. A parallel of this process is the program for NRC-administered licensed operator examinations in that, although exams to be administered are secured, once they have been administered and graded they, along with the answer key, become a public record.

In summary, the licensee met its procedural requirements for the control of examinations and thereby continued to implement the commitments made to the licensing board.

10.0 Control Room Habitability Test

During this inspection period, the resident inspector accompanied an NRR systems engineer in witnessing portions of Startup and Test Procedure (STP) 141/3, "Control Building (dP) Test with Single Mode Failure." The inspector found the STP to be detailed and specific enough to ensure that the test met the scope or stated objective of the test. The test results

indicated areas within the control building that were at a lower pressure than adjacent areas. The NRR systems engineer discussed the test results with the licensee's representatives and stated that their submittal to the NRC would have to demonstrate that these low pressure areas would not have an adverse affect on the habitability of the control room. The inspector had no further questions regarding this test.

11.0 Licensee Action on Previous Inspection Findings

The following items were reviewed to assure that the licensee took adequate corrective action in a timely manner and/or met their commitments as stated in applicable inspection reports.

11.1 (CLOSED) Inspector Follow Item (289/83-BC-02): Installation of Engineering Safety Features (ESF) Ventilation System for the Fuel Handling Building (FHB)

The NRC staff TMI-1 Restart Safety Evaluation Report (NUREG 0680, Supplement 3, page 19) accepted licensee plans to install the ESF Ventilation System for the FHB in accordance with Regulatory Guide 1.52, Revision 2. By Partial Initial Decision, dated December 14, 1981, paragraph 1265 and Order, dated April 5, 1982, the TMI-1 Restart ASLB accepted these plans for a commitment of prior to TMI-1 fuel movement from the reactor core for Cycle 6 refueling.

The inspector monitored a licensee preliminary engineering design review at the corporate office (paragraph 5.2). The inspector verified that the safety design objectives as a result of the TMI-1 restart hearings were incorporated into licensee design documents specifically or by reference to Regulatory Guide 1.52, Revision 2. Inspector Follow Item 289/83-BC-02 is considered closed. However, proper implementation of the design requirements is unresolved pending completion of licensee action and subsequent NRC Region I review (289/85-20-03).

11.2 (CLOSED) Inspector Follow Item (289/83-BC-01, 03, 06, 07, 08, 09, 13, 18, and 19): Various modification commitments to upgrade the emergency feedwater system to safety grade

Additional details for each of the items is addressed in paragraph 5.3.3. The inspector verified that the safety design objectives are incorporated into licensee design documents. The proper implementation of these design requirements to meet safety-grade criteria is unresolved (paragraph 5.3.3.5) pending completion of licensee actions and subsequent NRC Region I review.

11.3 (CLOSED) Inspector Follow Item (289/84-11-02): Update RM-L6 Alarm Response Procedure

In NRC Inspection Report 50-289/84-11, the inspector noted that the alarm response procedure (C-2-1 alarm procedure) for the plant liquid release radiation monitor, RM-L6, lacked specific guidance on how many times the monitor could be backflushed if the monitor alarmed. The procedure did not address when sampling was required in conjunction with backflushing. The inspector noted that the alarm response procedure was inconsistent with the licensee's approach to other radiation monitoring alarms. Because the RM-L6 alarm signal is used to terminate a plant release when the monitor exceeds a certain radiation level, the inspector questioned the adequacy of management guidance to shift personnel.

Subsequently, the licensee revised the C-2-1 alarm procedure. The revised procedure stated that the monitor may be backflushed once before a sample must be taken and analyzed. If the monitor did trip after the backflush, the operator was to investigate the cause before re-establishing the release. Based on the inspector's review, the revised procedure is now consistent with the intent of the applicable corresponding emergency and radiological control procedures.

11.4 (CLOSED) Violation (289/84-16-04): Failure to Properly Follow Radiation Work Permit (RWP)

As described in NRC Inspection Report 50-289/84-16, the inspector, while witnessing the demonstration of post-accident chemistry analysis, noted on two occassions the failure on the part of a chemistry technician to wear an alarming dosimeter when entering the Nuclear Sample Room. The applicable RWP required that a "Xetex" alarming dosimeter be worn by each individual. The licensee held a critique to determine the cause of the violation. The licensee's review noted that the chemistry technicians indicated that they were unaware that the Xetex had to be worn by at least one person in the laboratory or sample room when occupied. The technicians indicated that leaving the dosimeter on a laboratory bench gave no less representative exposure reading for technicians not assigned a Xetex than if the dosimeter was being carried by a single technician in the group. The licensee concluded the cause of the incident was the failure of health physics personnel to properly communicate Xetex use requirements to the chemistry technicians.

GPU responded to this notice of violation in a letter (5211-84-2223), dated August 30, 1984, to NRC Region I. The licensee stated that corrective actions taken were:

-- a critique was held on the day of the incident. Radiological Investigative Report No. 84-009 details the actions and conclusions of this critique;

-- a memorandum detailing the requirements for use of a Xetex instrument has been reissued to all TMI-1 departments; and,

-- all chemistry technicians have been instructed or otherwise informed as to the requirements for use of the Xetex dosimeter.

The inspector reviewed the applicable licensee records that documented the above corrective action. The inspector also discussed the corrective action with a plant chemistry foreman to ensure that the use of alarming dosimeters was understood and discussed the cause and corrective measures with station health physics personnel.

During this review the inspector noted that part of the root cause also stemmed from a lack of familiarity by certain personnel on their individual responsibilities associated with personnel radiation protection and as low as reasonably achievable (ALARA) concepts. The inspector stated that these facts should be emphasized in the general employee radiation training. However, the inspector noted that this training was strongly emphasizing that if you had a "Rad Con" problem, station health physics personnel were there to solve the problem. Apparently, some general employees had translated this idea into the belief that the Radiological Controls (Rad Con) Department was responsible for assuring their protection in the area of radiation exposure.

The Rad Con Manager restated that it was both the individuals' and station health physics personnel responsibility. The licensee's training representative stated they would review the training to ensure proper emphasis on individual responsibilities. The Rad Con Manager also stated that he was meeting with operation and maintenance personnel to reemphasize items such as this. The inspector determined that the licensee's corrective and preventive measures were appropriate for this violation, and that individual misunderstandings did not result in a radiological controls programmatic breakdown.

11.5 (CLOSED) Unresolved Item (289/84-24-01): Licensee to review job ticket for short form release to maintenance for its completion

NRC Inspection Report 50-289/84-24 described an inconsistency on how shift foremen were signing off the release of equipment to maintenance. This occurred when work was being performed on important to safety equipment. Maintenance procedure 1407-1 did not provide proper guidance on when shift foremen signatures are required to commence work.

The inspector reviewed Maintenance Procedure 1407-1, Revision 23, dated January 31, 1985. The inspector determined that adequate guidance was now incorporated in this procedure. The inspector reviewed package C-964, "Minor Maintenance on Various Components in the Reactor Building," dated February 8, 1985, and determined that various job tickets were now being completed consistently.

11.6 (CLOSED) Violation (289/84-24-02): Failure to determine the adequacy of minor maintenance work form to meet ANSI 18.7-1976

NRC Inspection Report 50-289/84-24 indicated that Maintenance Procedure 1407-1, "Unit 1 General Corrective Maintenance," Revision 16, dated August 23, 1984, was not reviewed, in part, for adequacy. Specifically, for minor maintenance, the work form was not adequate in that it did not provide for: 1) documented release of important to safety system equipment to maintenance by the operations department, 2) traceability of materials/parts, 3) documented use of maintenance procedures, and 4) specified post-maintenance test procedures including test acceptance criteria.

In a letter, dated December 5, 1984 (H. Hukill, GPUN to T. Murley, NRC) the licensee responded to the above Appendix A, Notice of Violation. Region I Inspection Report 50-289/84-38, described the licensee's response to this violation. The licensee's corrective actions were acceptable as stated in an NRC Region I letter dated March 13, 1985. However, the licensee was requested to provide a supplemental response to more fully address the root cause and corrective actions taken or planned to avoid further violations of this type. A letter, dated April 12, 1985, (H. Hukill, GPUN to T. Murley, NRC) provided the licensee's supplemental response.

The inspector reviewed Maintenance Procedure 1407-1, Revision 23, dated January 31, 1985. The four items identified in the notice of violation were adequately addressed in Revision 23; therefore, the corrective steps taken by the licensee were acceptable. The corrective actions to prevent further violations of this type was to provide guidance to all safety reviewers in the TMI-1 division. This was accomplished by an internal licensee memorandum dated April 9, 1985, (Nelson, GPUN, to PORC Members, 3200-85-9016). The inspector determined that this memorandum was a reasonable measure to prevent similar violations in the future. The effectiveness of these measures will be routinely reviewed by the resident inspectors.

11.7 (CLOSED) Inspector Follow Item (289/84-37-01): Inservice Testing (IST) Program Stroke Timing Requirements

NRC Region I Inspection Report 50-289/84-37 indicated that the river water supply to emergency feedwater system suction check valve EF-V3 was only partially stroke tested in the in-service testing (IST) program. Also, stroke timing for turbine driven EFW pump steam supply line valves MSV-10A and MSV-10B was not included in the surveillance procedure.

By various letters to the NRC the licensee sought IST relief for EF-V3 to conduct partial stroke testing in lieu of full stroke testing as required by the ASME Code, Section XI. However, this request was denied (NRC letter from J. F. Stolz to H. D. Hukill, dated October 23, 1984). As an alternate proposal, the licensee plans to remove the internals of the check valve. The licensee is now in the process of evaluating the safety impact of this action. The inspector will review the results of the evaluation during a subsequent inspection (289/85-20-04).

The inspector reviewed the surveillance procedure 1300-3K, "IST of Valves During Shutdown and Remote Indication Check," Revision 13, and noted that the stroke timing for MSV-10A and MSV-10B was incorporated in the procedure.

12.0 Restart Readiness

The NRC Inspection Report 50-289/85-19 documented the TMI-1 Restart Staff conclusion that there were no adverse conditions that would affect the safe restart of TMI-1. During this inspection period, the resident inspector continued to monitor plant status from a viewpoint oriented toward major equipment operability problems. Based on this review, the inspector concluded that there still was no adverse condition that would affect the safe restart of TMI-1 except for sporadic inoperability periods for one of two channels of source range nuclear instrumentation. This problem needs further evaluation prior to restart. Prior to any restart authorization, the TMI-1 Restart Staff will conduct another review of all open licensee and NRC issues similar to the restart readiness reviews previously documented.

13.0 Exit Interview

The inspectors discussed the inspection scope and findings with licensee management at the exit interview conducted on August 2, 1985. The following personnel attended the final exit meeting:

-- J. Colitz, Plant Engineering Director, TMI-1
-- W. County, Quality Assurance Lead Auditor, Nuclear Assurance Division
-- E. Eisen, Project Engineer, Technical Functions Division (TFD)
-- D. Hassler, Licensing Engineer, TFD
-- S. Otto, Licensing Engineer, TFD

As discussed at the meeting, the inspection results are summarized in the cover page of the inspection report. The licensee representatives indicated that none of the subject matter discussed contained proprietary information. The inspector noted that there were no obstacles (physical or administrative) to the safe restart of the unit, however, the potentially unreliable channel of source range instrumentation requires further evaluation prior to restart.

Unresolved Items are matters about which information is required in order to ascertain whether they are acceptable items, violations or deviations. Unresolved item(s), discussed during the exit meeting, are documented in paragraphs 5.3.3.5, 7.0, 11.1, and 11.5.

Inspector Follow Items are matters which were established to administratively follow open issues based on licensee or staff commitments from the TMI-1 restart hearing. Inspector follow item(s), discussed during the exit meeting, are documented in paragraphs 5.3.3, 11.1, 11.2, 11.3, and 11.7.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICE OF NUCLEAR REACTOR REGULATION

EVALUATION OF OPERATION OF TMI-1 WITH UP TO

2000 PLUGGED STEAM GENERATOR TUBES

1. Introduction

In the staff's Safety Evaluation Report related to steam generator tube repair and return to operation of TMI-1, NUREG-1019, we examined the effects of plugging up to 1500 steam generator tubes on TMI-1 reactor thermal and hydraulic considerations and on various transients and accidents. We concluded that the thermal-hydraulic consequences of such operation were acceptable, and that accident consequences were bounded by the FSAR analysis or meet appropriate criteria and were therefore acceptable.

Since that time, additional tubes have been plugged and the total now is 1542. The licensee submitted its safety evaluation TDR No. 674, Comparison of Steam Generator Tube Plugging with the TMI-1 Design Basis, in which it also concluded that plugging 3000 tubes would have no adverse affect on performance of the steam generators or on licensing basis analyses for transients and accidents. However, in order to assure conformance with present Technical Specifications, the licensee in Revision 1 to TDR No. 674, limited the applicability of that document to 2000 tubes. The licensee also concluded that plugging up to 2000 tubes does not involve an unreviewed safety question as defined in 10 CFR 50.59.

We have reviewed TDR No. 674 to verify the licensee's conclusions. Our evaluation is summarized below.

2.0 Evaluation

2.1 Core Thermal-Hydraulic Design

The existing TMI-1 safety analysis for Cycle 5 operation is based on a power level of 2568 Mwt and a reactor coolant system (RCS) flow of 106.5% of the design flow of 131.3×10^6 lbm/hr. The licensed TMI-1 power level is 2535 Mwt and the measured four pump flow is reported to be 109.5% of the design flow, with 1.5% flow calibration uncertainty. Plugging of steam generator tubes increases the RCS flow resistance and results in flow degradation. The licensee has calculated that RCS flow reduction of 2.0% would result from the plugging of 3000 tubes. Thus, considering flow uncertainty, this case could result in an actual flow of 106.0% which is slightly below the existing safety analysis and Technical Specification limit of 106.5% for four pump operation. We have considered the impact of this reduced flow on reactor protection system trip limits and capability for full power (2535 Mwt) operation, as discussed below.

Even though tube plugging results in reduced RCS flow, the flux/flow trip function in the TMI-1 plant protection system provides necessary protection with respect to overpower at reduced flow. This trip function is specified in the TMI-1 Technical Specifications where the power level trip setpoint is dependent on the RCS flow rate and power imbalance. Flow reduction reduces the power level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% for a one percent flow reduction. This is based on the sensitivity of DNBR margin to power and flow changes and prevents a DNBR of less than 1.3 (limit value) if a low flow condition should exist due to any malfunction.

For the licensed power level of 2535 Mwt. the safety analysis DNBR margin and protection system setpoint bases would be preserved for reduced flow to 105.3% of design flow based on the power/flow versus DNBR sensitivity relationship. Thus, the existing setpoint for overpower protection (105.5%) could be justified for actual flow as low as 105.3%. However, the flow is limited by current Technical Specifications to a minimum value of 106.5%. We have also reviewed the licensee's statements that up to 2000 tubes could be plugged without reducing flow below the TS figure, and we concur with that conclusion.

The licensee has also evaluated the plugging of 3000 tubes with a plugging ratio of 3:1, i.e., 2250 tubes in OTSG "A" and 750 tubes in OTSG "B". The licensee's calculation has determined that this plugging configuration would result in loop "A" flow rate approximately 2.5% smaller than loop B. However, the licensee also states that field data during the last cycle had shown that A loop had typically about 3% more flow than B loop. As a result, the net flow difference due to 3:1 plugging configuration would be approximately 0.5%, and therefore, the 3:1 plugging configuration is acceptable.

In summary, existing reactor protection system setpoints provide DNBR protection for power operation at 2535 Mwt with flow reduced to 105.3% even though the flow is limited to a low value of 106.5% by current Technical Specifications. We conclude that TMI-1 can be operated within the TS limits on RCS flow with up to 2000 tubes plugged. As part of the power escalation test program, the licensee will verify by flow calibration that the RCS flow remains above existing Technical Specification limits.

2.2 Transient and Accident Analysis

The effect of plugging up to 3000 steam generator tubes on the consequences of design basis transients and accidents will be minimal. The steam generator tubes account for less than 25% of the total RCS pressure drop. Plugging 3000 out of a total of more than 30,000 tubes would cause the pressure drop through the steam generator tubes to increase. As discussed above, this increase in pressure drop would cause the total coolant flow to decrease by approximately 2% as

calculated by the licensee. For transients and accidents involving loss of forced coolant flow the effect of the tube plugging on flow coast down and natural circulation flow would also be of little significance, because the stalled coolant pumps introduce a large flow resistance in the coolant loops that is substantially greater than the flow resistance imposed by tube plugging. The effect of the plugged tubes on the ability of the steam generators to remove heat would also be minimal. During power operation, secondary system water level would be adjusted upward as needed to provide for reactor system heat removal. Following reactor trip the heat transfer surface would be more than adequate to remove core decay heat.

The licensee evaluated the consequences of design basis transients and accidents and concluded that the evaluations previously submitted in support of the original license application would still be bounding.

The design basis loss of coolant accidents for TMI-1 were evaluated for a generic B&W Plant with lowered loops having a power level 9% higher than that of TMI-1 using approved 10 CFR Appendix K models. The most severe small break LOCA was determined to be a $0.07ft^2$ cold leg break. This break size would be sufficient to remove decay heat so that steam generator heat removal would not be required. Uncovery of a region at the top of the core was calculated to occur between 1350 seconds and 1750 seconds. At this time the steam generators would be acting as a heat source and not be aiding in core cooling. Loss of steam generator heat transfer surface from tube plugging would not affect the consequences of this accident.

One class of small break LOCA depends on steam generator heat removal for event recovery. Break sizes of $0.01ft^2$ and smaller would be unable to remove reactor decay heat solely through the break and would require steam generator heat removal in the boiler-condenser mode. Previous analyses of small breaks in this size range without tube plugging have demonstrated that the consequences would not be bounding and that neither core heatup nor core uncovery would occur. The boiler-condenser mode of decay heat removal involves condensation of steam generated by the core on condensing surface in the steam generators. The condensing surface would be provided by emergency feedwater EFW spray on the outermost tubes and the action of the operator to raise the steam generator water level to 95% on the operating range, which is well above the top of the core. The establishment of an adequate condensing surface above the top of the core is important to provide for reactor system depressurization which increases high pressure injection flow preventing core uncovery. The staff has concluded that at the 95% level an adequate condensing surface would be available to remove all decay heat, with a considerable margin. The plugging of 3000 tubes would remove 10% of this condensing surface. However, the remaining surface would still be more than adequate to remove all decay heat. The staff concludes that core uncovery would not occur for breaks in the size range of $0.01ft^2$ and smaller if up to 3000 tubes are plugged.

For large break LOCA analysis, a critical feature for some plants is the resistance of steam flow through the reactor coolant loops and steam generators during reflood. For TMI-1 the peak cladding temperature was calculated to occur during the reflooding period. The analysis did not take credit for flow in the reactor coolant loops including the steam generators during this period and assumed they were completely blocked by water in the cold legs. Relief of steam from the core was assumed only through the core barrel vent valves. This assumption would be unaffected by steam generator tube plugging. The staff concludes that the consequences of a large break LOCA would not be affected by plugging up to 3000 steam generator tubes.

The licensee also evaluated the consequences from non-LOCA transients and accidents. The reactor system flow coast down curve in the FSAR for loss of forced flow events was determined to be still bounding for the case of 3000 plugged tubes. This determination was made using the B&W PUMP computer code which has been approved by the NRC staff. After the coolant pumps were stopped, natural circulation flow would continue through the core. The natural circulation flow was calculated to be negligibly affected by tube plugging.

The licensee has previously committed to confirm the adequacy of natural circulation flow in tests at low power during power escalation. This action is included in the restart license conditions.

Since steam generator secondary side water inventory will increase in order to compensate for the reduced heat transfer surface following tube plugging, the licensee evaluated the revised inventory in comparison to that assumed in the FSAR for steam line break analysis. The FSAR inventory assumption of 55,000 lbs was determined to bound the revised steam generator water mass by a considerable margin.

In the event of loss of feedwater or a main feedwater line break, the increase in inventory would provide an additional heat sink until EFW could be actuated. More time would be available before steam generator dryout could occur. The FSAR analyses would therefore be bounding for events of this type.

Although the analysis of a locked reactor coolant pump rotor is included in the FSAR, the licensee did not evaluate the consequences of a locked rotor accident accounting for plugging 3000 tubes. The licensee has stated that the design basis locked rotor analysis, which assumed no plugged tubes, also assumed very conservative values for loop resistance, core power peaking, and core flow bypass. Because of these factors and the small change that tube plugging produces on reactor coolant flow, the staff does not expect that the consequences from accidents of this type would be significantly affected. The licensee should, however, analyze this event to confirm this conclusion.

The licensee evaluated other design basis events including boron dilution, coolant pump startup, loss of electric power and steam generator tube failure. The licensee concluded that these events would not be significantly affected by plugging up to 3000 tubes. The staff agrees with these conclusions. The effect of the plugged steam generator tubes on reactivity initiated transients and accidents has been reviewed. The reductions in flow and heat transfer are not large enough to affect uncompensated operating reactivity changes, CRA withdrawal events from startup or power conditions, misaligned or dropped CRA events, fuel handling events or the rod ejection accident. The FSAR analyses for these events, therefore, remain bounding.

3.0 Conclusions

Based on our review as summarized above, we find that our earlier conclusions in NUREG-1019 regarding the effects of plugged steam generator tubes remain valid for up to 2000 tubes. We agree with the licensee's conclusion that operation with up to 2000 plugged tubes does not involve an unreviewed safety question as defined in 10 CFR 50.59.