U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-389/82-13

Docket No. 50-289

License No. DPR-50 Priority -- Category C

Licensee: GPU Nuclear Corporation

P.O. Box 480

Middletown, Pennsylvania 17057

Facility Name: Three Mile Island Nuclear Station, Unit 1

Inspection at: Middletown, Pennsylvania and Reading, Pennsylvania

Inspection conducted: August 2-5, August 31, and September 1-3, 1982

Inspectors: DR Haverkamp 10/1/82
D. Haverkamp, Reactor Inspector date

P.K. Eapen 10/1/82
P. Eapen, Reactor Inspector date

Approved by: DR Haverkamp 10/1/82
for A. Fasano, Chief, Three Mile Island Section date
Projects Branch No. 2

Inspection Summary:
Inspection conducted on August 2-5, August 31 and September 1-3, 1982
(Inspection Report Number 50-289/82-13)

Areas Inspected: Special safety inspection by region-based inspectors (91 hours) of licensee action on previous inspection findings; reactor building flood level limitations; plant shielding design review; and TMI-1 restart modifications-implementation.

Results: No violations were identified.

DETAILS

1. Persons Contacted

GPU Nuclear Corporation

J. D. Abramovici, Senior Mechanical Systems Engineer, Technical Functions
S. Furman, Auxiliary Operator, TMI-1
*J. W. Garrison, Quality Assurance Engineer, Nuclear Assurance
L. W. Harding, Senior Licensing Engineer, Technical Functions
*C. H. Kimball, Engineer Assistant, Nuclear Assurance
G. A. Kuehn, Deputy Radiological Controls Manager TMI-1, Radiological Controls
*J. R. Pearce, Engineer Senior I, TMI-1
W. P. Potts, Radiological Controls Manager TMI-1, Radiological Controls
*M. J. Ross, Manager Plant Operations, TMI-1
*H. B. Shipman, Engineer Senior II, TMI-1
D. A. Smith, Shift Supervisor, TMI-1
*C. W. Smyth, TMI-1 Licensing Supervisor, Technical Functions
*R. J. Toole, Operations and Maintenance Director, TMI-1

Gilbert/Commonwealth, Inc.

R. Brems, TMI-1 Project Manager
J. Kamphouse, Project Engineer, Plant Engineering and Analysis

*denotes those present at exit interview on September 3, 1982

2. Licensee Action on Previous Inspection Findings

(Closed) Inspector Followup Item 289/82-BC-39: Complete shielding modifications (TASK LM-51A) to protect personnel in the area of a motor control center. Details are discussed in paragraph 5.

(Closed) Inspector Followup Item 289/80-22-HO: Review licensee's plant shielding design review. Details are discussed in paragraph 4.

(Closed) Inspector Followup Item 289/80-22-H1: Review licensee's post-accident sampling studies/modifications required by NUREG-0578, Section 2.1.8.a. The NUREG-0578 requirements for this item have been incorporated in NUREG-0737, Item II.B.3. Licensee actions are being tracked by the NRC resident staff under 82-BC-42 and 82-BC-43.

(Closed) Inspector Followup Item 289/80-22-H2: Review inplant iodine instrumentation used to meet requirements of NUREG-0578, Section 2.1.8.c. The NUREG-0578 requirements for this item have been incorporated in NUREG-0737, Item III.D.3.3. Licensee actions are being tracked by the NRC resident staff under 82-BC-46.

(Closed) Inspector Followup Item 289/80-22-H3: Review high range effluent monitor modifications required by NUREG-0578, Section 2.1.8.b. The NUREG-0578 requirements for this item have been incorporated in NUREG-0737, Item II.F.1. Licensee actions are being tracked by the NRC resident staff under 82-BC-45.

3. Reactor Building Flood Level Operational Limitations

a. Background and Scope

The Atomic Safety and Licensing Board (ASLB) in the matter of Three Mile Island Unit 1 (TMI-1) Restart, directed the NRC staff to conduct a complete review of the operational limitations that must be imposed on the licensee to ensure that the reactor building flood level does not exceed the licensee's calculated maximum flood level. In particular, the staff was directed to review the ability of the licensee to enter the recirculation mode under all postulated accident conditions where the recirculation mode would be necessary to maintain flood levels within the licensee's calculation. The staff was to review all emergency procedures for these accidents to ensure these operational limits are properly incorporated into the procedures. Additional details of this matter and its bases are described in paragraph 1174 of the ASLB's Partial Initial Decision, Volume 1, "Plant Design and Procedures and Separation Issues," dated December 14, 1981. The staff action was required to be completed and certified to the Commission prior to TMI-1 restart.

The above review was performed during this inspection. The inspector reviewed various emergency and operating procedures which were, or potentially were, related to all postulated accident circumstances where the recirculation mode would be necessary to maintain the reactor building flood level within the licensee's calculated maximum flood level. The operational limitations considered during this review included (1) the ability to enter several recirculation modes by performing actions as described in applicable procedures and (2) the maximum available water sources that could be transferred or discharged to cause reactor building flooding.

b. Discussion

The following licensee procedures and drawings were reviewed.

-- Emergency Procedure (EP) 1202-6A, "Loss of Reactor Coolant/Reactor Coolant Pressure Within Capability of Makeup System (RC Pressure Above ESAS Setpoint)," Revision 10, dated July 20, 1982.

-- EP 1202-6B, "Loss of Reactor Coolant/Reactor Coolant Pressure (Small Break LOCA) Causing Automatic High Pressure Injection," Revision 13, dated June 17, 1982.

-- EP 1202-6C, "Loss of Reactor Coolant/Reactor Coolant Pressure Causing Automatic High Pressure Injection, Core Flood and Low Pressure Injection," Revision 9, dated January 15, 1982.

-- Operating Procedure 1104-4, "Decay Heat Removal System," Revision 33, dated June 21, 1982.

-- Drawing C-302-650, Revision 19, dated May 3, 1982.

-- Drawing C-302-640, Revision 28, dated February 17, 1982.

-- Drawing C-302-712, Revision 16, dated October 1, 1980.

The review consisted of a step-by-step comparison of each procedure with the applicable piping and instrument drawings, to verify that proper flow paths can be established, and a plant walkdown of portions of each procedure, to determine the ability to perform the procedure and the accessibility of manual valves that may require local operation to enter and maintain long-term recirculation.

Based on this review, the inspector determined that the licensee is able to enter several recirculation modes by performing the actions described in the above procedures. Various manual valves needed for long-term recirculation are located in potentially inaccessible areas in the event of an accident that causes substantial core damage. This problem is discussed further in paragraph 4.c of this report. However, these valves do not require operation to prevent exceeding the licensee's calculated containment flood level, as no additional water sources are used.

Of the three emergency procedures for a loss of coolant accident (LOCA), EP 1202-6C (large break LOCA) is the most limiting with respect to total water inventory transferred to the containment. The sources of water for containment flooding include the borated water storage tank (BWST), the sodium hydroxide tank, the core flood tanks and the reactor coolant system (RCS) water volume. These are the same sources of water assumed in the licensee's containment flood level calculations, as described in a licensee letter to the NRC dated June 11, 1982. The licensee's calculations assume maximum possible tank emptying and complete discharge of the RCS water volume, however, less water would actually be discharged or transferred. Some RCS water would remain in the intact portion of the reactor vessel up to the level of the reactor vessel nozzles. Additionally, the BWST and sodium hydroxide tanks are isolated by procedure before their levels decrease to the height of the suction piping tank connections. No other water sources would be transferred to the containment in the event of a LOCA. Therefore, no operational limitations are required to prevent exceeding the licensee's calculated flood level.

This information and additional details of the licensee's emergency procedures have been provided to NRR. The inspector had no further questions regarding this matter.

4. Plant Shielding Design Review

a. Background and Scope

As discussed in NUREG-0737, "Clarification of TMI Action Plan Requirements," each power reactor licensee was required to perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review was intended to identify the location of vital areas and equipment in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems. Additionally, each licensee was required to provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review was to determine which types of corrective actions were needed for vital areas throughout the facility.

These requirements were originally issued by NRC letters to all operating nuclear power plants, dated September 13 and October 30, 1979, and were incorporated into NUREG-0660, "TMI-2 Action Plan." Significant changes in requirements or guidance were described in NUREG-0737, Item II.B.2. In the case of TMI-1, the shielding design review and corrective actions were discussed by the licensee in Section 2.1.2.3 of the TMI-1 Restart Report ("Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1"). The licensee's shielding study and planned actions were evaluated by the NRC staff in Section 2.1.6.b of NUREG-0680, "TMI-1 Restart Evaluation Report, to comply with NRC Commission Order of August 9, 1979." The licensee subsequently discussed the status and some design details for the modifications in letters to the NRC dated August 10, 1981 and June 15, 1982.

The licensee's plant shielding design review and corrective actions were reviewed during this inspection. The review included (1) a sampling verification of the shielding design review methodology and representative calculations, (2) a review of selected emergency procedures to determine if the vital areas where personnel must go are safely accessible, and (3) a review of corrective actions taken or planned by the licensee including plant modifications.

b. Shielding Design Review Verification

The licensee's shielding design review methods, including (1) source terms, (2) calculation of dose rates, (3) calculation of doses to personnel during post-accident access to vital areas, and (4) acceptance criteria, were described in Section 2.1.2.3.3 of the TMI-1 Restart Report. The shielding design review report was evaluated by the NRC staff, as discussed in Section 2.1.6.b of NUREG-0680 and NUREG-0680, Supplement 3, and the short-term aspects of this item were found acceptable.

The inspector discussed the details of the Shielding Design Review with the licensee and his contractor. The contractor's project engineer made detailed presentations on the assumptions and methodology used in shielding calculations and the results obtained from such calculations. The inspector noted that the assumptions were consistent with the guidelines of NUREG-0737 and the methodology employed by state-of-the-art technology for mathematical modeling.

The Project Engineer also demonstrated how he arrived at calculated dose levels for Plant Areas X, XII, XIV and XV from the available sources of radiation (pipes) and shielding in each of the areas. The inspector found these dose level estimates to be conservative, reasonable and consistent with the state-of-the-art technology for shielding design. The inspector had no further questions on this matter.

c. Vital Area Accessibility

(1) Procedure review

The inspector reviewed three emergency procedures and one operating procedure that would be implemented by the licensee in the event of various severities of a loss of coolant accident. The review included (1) a step-by-step comparison of each procedure with applicable piping and instrument drawings to verify that proper flow paths can be established, (2) a plant walkdown of portions of each procedure to determine the ability to perform the procedure and the accessibility of manual valves that may require local operation, and (3) an assessment of potential exposures of plant personnel based on the results of the licensee's shielding design review. The procedures reviewed included Emergency Procedure (EP) 1202-6A, EP 1202-6B, EP 1202-6C and portions of Operating Procedure 1104-4, and the drawings reviewed included C-302-640, C-302-650 and C-302-712. (These procedures and drawings were also reviewed regarding the ASLB concern on reactor building flood level operational limitations. Specific revisions of procedures and drawings reviewed are listed in paragraph 3b of this report.)

The three emergency procedures provide immediate and follow-up actions to be taken for a loss of coolant accident (LOCA), based on the severity of the LOCA. EP 1202-6A is applicable for a LOCA that is within the capability of the makeup system; EP 1202-6B is applicable for a small break LOCA causing automatic high pressure injection; and EP 1202-6C is applicable for a large break LOCA causing automatic high pressure injection (HPI), core flood and low pressure injection (LPI). The immediate and initial follow-up actions are performed either automatically or by remote-manual operation of components from the control room or other accessible plant areas. EP 1202-6B and EP 1202-6C follow-up actions include provisions that transfer HPI, LPI and/or reactor building spray flow paths from a borated water storage tank suction mode to an initial reactor building (RB) sump recirculation mode, again by remote-manual operation of components from the control room. However, subsequent to establishing the initial sump recirculation mode, the operator is directed to throttle the LPI motor-operated discharge valves as required to maintain LPI flow greater than 1000 gpm and less than 3500 gpm, and, when time permits, to throttle the LPI manual discharge valves and reopen the motor-operated valves. Within about 24 hours, a long term cooling circulation mode is established, as described in OP 1104-4 and listed below.

-- Mode 1 - Forced circulation using decay heat drop line

-- Mode 2 - Gravity draining reactor coolant hot leg to the RB sump via the decay heat drop line

-- Mode 3 - Hot leg injection using pressurizer auxiliary spray line

-- Mode 4 - Reverse flow through the decay heat drop line into the "B" reactor coolant hot leg

In the event that core damage has occurred with substantial release of the core fission product inventory, such as assumed in the licensee's shielding design review, the above operations may not be able to be conducted. Based on the shielding study results, the LPI manual throttle valves are located in potentially inaccessible areas due to the calculated high radiation levels. In addition, establishing any of the four modes of long term recirculation requires operation of one or more manual valves in potentially inaccessible areas. Specific operational considerations and radiological concerns are discussed below.

(2) Inaccessible vital areas

OP 1104-4 provides several specific procedures related to operation of the Decay Heat Removal (DHR) system including normal system startup, operation and shutdown, DHR emergency standby, chemical addition, long term core circulation modes to prevent boron concentration effects, shifting DHR strings, and oil level determination and addition for DH pump and motor external oil reservoirs. This discussion is related to only that portion of OP 1104-4 concerning long term core circulation modes to prevent boron concentration effects.

OP 1104-4 states that to prevent concentration of boron in the reactor vessel post LOCA, one of the long term circulation modes described below should be placed into operation with 24 hours of the LOCA. Action within this time frame is considered more than adequate to avoid significant boron concentration effects which may occur during natural circulation flow patterns within the reactor vessel, even for the limiting condition of a large reactor vessel inlet pipe break.

The first method of long term circulation (mode 1) establishes decay heat drop line flow through the "A" DH pump. This method is attempted only if both LPI strings A and B are operable. The second method of long term circulation (mode 2) is by gravity draining the reactor coolant hot leg to the reactor building sump via the decay heat drop line. This method is used if satisfactory flow cannot be established by the procedure of mode 1. The third method of long term circulation (mode 3) is by hot leg injection using the auxiliary spray line. This method can accommodate a single failure in that the decay heat drop line is not required. The flow path is from the operating DH pump through the pressurizer auxiliary spray line into the pressurizer. This method is used only if one of the LPI strings is inoperable or if satisfactory flow cannot be established by the procedure of modes 1 or 2.

The fourth method of long term circulation (mode 4) is established by providing reverse flow through the decay heat drop line in the "B" reactor coolant loop hot leg. This flow path provides flow from the operating DH pump through the idle DH pump recirculation and suction lines and back up the DH drop line to result in reverse flow hot leg injection into the core. This method should provide about 70 gpm into the hot leg and should be used if the methods described above are not functional.

[NOTE: The capability to control long-term boron concentration buildup has been reviewed previously by the NRC staff. The specific safety analysis considerations and staff evaluation are discussed in the following documents.

-- BAW-10091 Supp. 1 Topical Report December 1974, "Supplementary and Supporting Documentation for B&W's ECCS Evaluation Model Report with Specific Application to 177-FA Class Plants with Lowered Loop Arrangements."

-- BAW-10103A Rev. 3 Topical Report July 1977, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS - Revision 3."

-- J. F. Stolz (NRC) letter to K. E. Suhrke (B&W) dated February 4, 1976.

It was the staff's position that Mode 1 should not be attempted as a method to control boron concentration, since this action could result in the decrease of required safety equipment. This matter was discussed with licensee management during the inspection. The licensee's staff is reviewing the need for deleting the Mode 1 long term recirculation method from EP 1202-6B and OP 1104-4. NRC disposition of this matter is discussed in paragraph 4.e.(2)(a).]

Each of the above methods of long term circulation requires the operation of certain manual valves in the decay heat removal system. These valves or their handwheels are located in the basement level of the auxiliary building (elevation 281'0") or in the decay heat pump vaults below the basement level. Based on the licensee's shielding design review, the specific areas in which these valves are manually operated would be inaccessible due to the excessive radiation levels which might exist. Therefore, the licensee has planned procedural controls or design modifications for the eight manual valves listed below.

DH-V19A & B   DH Cooler Outlet Throttle Valves

DH-V38A & B	DH Discharge Cross-Connect Valves
DH-V15 A & B	DH Pump Suction Isolation Valves
DH-V12 A & B	Hot Leg Drop Line Cross-Connect Valves

Specific actions planned by the licensee are discussed in paragraph 4.d of this report.

The table below illustrates the need for operation of these valves, in order to establish each of the long term circulation modes described in OP 1104-4. For each mode, the table identifies the required valve operation (open/closed), the location of the valve or its reach rod (shielding study plant area), the licensee's calculated dose in order to manually operate the valve at time zero (exposure at T=0), and an estimated dose in order to manually operate the valve 24 hours after the LOCA (exposure at T=24 hours).

Note: The licensee stated in the shielding design report that some operations may be performed at times significantly after T=0, and in the licensee's calculations, dose rates were generally reduced by a factor of 5 for T=8 hours and by a factor of 30 for T=5 days. The estimated exposures are based on the T=0 dose rate, arbitrarily reduced by a factor of 10 for T=24 hours. The actual reduction factor may be somewhat more or less than this value.

LONG TERM CIRCULATION MANUAL VALVE OPERATIONS

Valve Operation	Plant Area	Dose at T=0	Dose at T=24 Hours
Mode 1: Forced Circulation Using Decay Heat Drop Line			
Open DH-V12A	XIII	1800 Rem	180 Rem
Throttle DH-V19A	XII	1700 Rem	170 Rem
Mode 2: Gravity Draining Reactor Coolant Hot Leg to the Reactor Building Sump Via the Decay Heat Drop Line			
Open DH-V12A	XIII	1800 Rem	180 Rem
or DH-V12B	XIV	1400 Rem	140 Rem
Close DH-V15A	"A" DH Vault*	Excessive	Excessive
or DH-V15B	"B" DH Vault*	Excessive	Excessive
The following valves are operated only if necessary to verify satisfactory gravity drain flow.			
Open DH-V38A	XII	1700 Rem	170 Rem
Open DH-V38B	XI	580 Rem	58 Rem
Open DH-V15A	"A" DH Vault*	Excessive	Excessive
or DH-V15B	"B" DH Vault*	Excessive	Excessive
Close DH-V15A	"A" DH Vault*	Excessive	Excessive
or DH-V15B	"B" DH Vault*	Excessive	Excessive

* The "A" and "B" Decay Heat Vaults were not included in the licensee's shielding design review of vital areas. However, the post accident radiation levels and associated personnel exposures for access to the vaults to operate DH-V15A or DH-V15B are expected to be excessive, such that access and valve operation would be prohibited.

LONG TERM CIRCULATION MANUAL VALVE OPERATIONS (Continued)

Valve Operation	Plant Area	Dose at T=0	Dose at T=24 Hours
Mode 3: Hot Leg Injection Using Pressurizer Auxiliary Spray Line			
Open DH-V38A	XII	1700 Rem	170 Rem
Open DH-V38B	VI	580 Rem	58 Rem
Throttle DH-V19A	XII	1700 Rem	170 Rem
Throttle DH-V19B	XI	580 Rem	58 Rem
Close DH-V15A or DH-V15B	"A" DH Vault* "B" DH Vault*	Excessive Excessive	Excessive Excessive
Open DH-V64	IV	666 Rem	66 Rem

Note: DH-V64 is operated via a reach rod at the 305' elevation of the auxiliary building. The above exposures are based on original shielding study calculations. The licensee has since provided shielding in Area IV, such that the design exposure would be less than 5 Rem at 11 hours after the accident.

Valve Operation	Plant Area	Dose at T=0	Dose at T=24 Hours
Mode 4: Reverse Flow Through the Decay Heat Drop Line into "B" Reactor Coolant Loop Hot Leg			
Open DH-V38A	XII	1700 Rem	170 Rem
Open DH-V38B	XI	580 Rem	58 Rem
Throttle DH-V19A	XII	1700 Rem	170 Rem
Throttle DH-V19B	XI	580 Rem	58 Rem
Open DH-V12B or DH-V12A	XIV XIII	1400 Rem 1800 Rem	140 Rem 180 Rem

* See bottom of previous page.

The valve operations listed in the preceding table were identified by the inspector during the review of EP 1202-6B, EP 1202-6C and OP 1104-4 to determine whether the vital areas where personnel must go are safely accessible. "Safely accessible" means that the dose to personnel should not be in excess of 5 Rem whole body, or its equivalent to any part of the body for the duration of the accident. This definition is consistent with the design dose rate criteria specified in NUREG-0737. The potential personnel doses associated with the above valve operations are substantially greater than 5 Rem. However, the licensee had identified this problem during the shielding design review and, therefore, the licensee has planned various procedural controls or design modifications for these and other valves. These corrective actions are discussed in the next paragraph. When the actions are completed, post-accident access should no longer be required to the inaccessible areas identified by the licensee. NRC disposition of this matter is discussed in paragraph 4.e(2)(b).

d. Corrective Actions

Based on the results of the plant shielding design review, the licensee determined that the calculated doses would preclude post-accident access needed to perform certain operational actions without appropriate corrective actions. The paragraphs below summarize the required post-accident operational actions and respective commitments for corrective actions as described in the licensee's TMI-1 Restart Report, updated commitments for corrective actions as discussed in licensee letters to the NRC dated August 10, 1981 and June 15, 1982, and the status of corrective actions as identified during this inspection.

(1) Required Action: Manually open valve MU-V198 to bypass seal injection filters.

Corrective Action: Change operating procedures to require manual opening of MU-V198 before going to recirculation from reactor building sump (i.e. before BWST is depleted)

Updated Corrective Action: EP 1202-6B and EP 1202-6C will be revised when decay heat remote valve operation is achieved.

Status: Procedures not revised. The inspector noted that MU-V198 operation is neither dependent on nor related to decay heat remote valve operation. Therefore, there is no basis for deferral of the revisions to EP 1202-6B and EP 1202-6C, and these procedures should be revised prior to restart, as discussed in paragraph 4.e.(2)(c).

(2) Required Action: Reset any thrown circuit breakers in Motor Control Centers (MCC) 1A and 1B.

Corrective Action: Install a shield wall in Area IV to isolate the MCC's from the piping.

Updated Corrective Actions: A reinforced concrete shield wall has been designed and will be in place prior to Unit 1 restart. The shield wall reduces the radiation dose rate in Area III to permit operator access post-accident to reset circuit breakers in MCC 1A and 1B.

In addition, the licensee noted that the reactor coolant pump seal water supply line and high pressure injection line B bridge the access aisle between the shield wall and the containment outer wall. These lines were shielded by a 3½" thick steel plate assembly, which is removable for maintenance access to valve operators in other overhead piping.

Status: Modifications completed. (See paragraph 5)

(3) Required Action: Manually operate valves DH-V15 A & B, DH-V19 A & B, DH-V38 A & B, DC-V2 A & B and DC-V65 A & B for boron precipitation control and for continued decay heat removal.

Corrective Actions: Change valves DH-V19 A & B and DH-V38 A & B to remote air-operated with air provided from bottled gas supply good for two hours operation. Provide DC power for valve actuation and manual loaders for positioning DH-V19 A & B. Revise procedure 1104-4 concerning post LOCA boron control so that valves DH-V15 A & B remain open and valves DH-V5 A & B and DH-V6 A & B are closed.

Updated Corrective Actions:

Remote Operators for DH-V19 A & B, DH-V38 A & B

Pneumatic operators are to be retrofitted to decay heat exchanger outlet isolation valves DH-V19 A & B and decay heat crossover valves DH-V38 A & B. The primary pneumatic supply is plant instrument quality air with dry air cylinder backup source. The operators will open and close the valves, modulate DH-V19 A & B for decay heat flow control and fail "as is" upon loss of all pneumatic pressure. The backup dry air source is a minimum of three size 1A storage cylinders accessible post-accident for cylinder replacement. Modifications cannot be completed until Cycle 6 refueling, when the licensee expects to have material delivery and the plant conditions necessary to complete the installation.

Revisions to Operating Procedure 1104-4

Changes to the operating procedure will be accomplished when the remote operation capability of DH-V19 A & B is achieved.

Status:

Remote Operators for DH-V19 A & B, DH-V38 A & B

Modifications not completed. NUREG-0737, Item II.B.2 stated that modifications for vital area access should be completed by January 1, 1982. In the case of TMI-1, NRC staff and Commission approval is needed for those NUREG-0737 requirements not completed prior to TMI-1 restart. The licensee is currently seeking approval for deferring modifications for DH-V19 A & B and DH-V38 A & B until Cycle 6 refueling, which is estimated to start in March 1984. The post-accident operational need for these valves is discussed in paragraph 4.c of this report. This information is being provided to the NRR staff for consideration in their review of the licensee's request. The NRC Region I Staff will review the final installation subsequent to licensee completion of the valve modifications, as discussed in paragraph 4.e.(4).

Revisions to Operating Procedure 1104-4

Procedure not revised. The inspector noted that the licensee's planned revisions to OP 1104-4 are not appropriate. The proposed procedure revision would allow the DH pump suction isolation Valves DH-V15 A & B to remain open, but would close both BWST suction isolation valves DH-V5 A & B and both RB sump suction isolation valves DH-V 6 A & B, thus isolating the water sources needed for recirculation. (The licensee may have intended to close the DH pump recirculation isolation valves DH-V56 A & B, vice DH-V5 A & B and DH-V6 A & B.) Additionally, the planned procedure revision to allow DH-V15 A & B to remain open did not appear to be dependent on completing the remote-operation modifications for DH-V19 A & B. OP 1104-4 should be revised prior to restart, as discussed in paragraph 4.e.(2)(d).

Corrective Actions for DC-V2 A & B, DC-V65 A & B

One of the required post-accident actions identified by the licensee, as stated above, was to manually operate DC-V2 A & B and DC-V65 A & B for continued decay heat removal. The TMI-1 Restart Report stated that access may be required post-accident to open and/or throttle the air-operated decay heat closed cooling system valves DC-V2 A & B and DC-V65 A & B to achieve reactor coolant temperature control. The pneumatic controller for operating these valves is located in Area XI on the 281' elevation of the Auxiliary Building. The licensee's estimated dose at T=0 for operation of these valves is 580 Rem. (The estimated dose at T=24 hours would be about 58 Rem.) This exceeds the 5 Rem design dose rate criterion specified in NUREG-0737. However, no corrective actions were described by the licensee in the TMI-1 Restart Report or subsequent correspondence to the NRC. Licensee representatives were not able to resolve this disparity during this inspection. The licensee should determine the corrective actions needed to allow operation of these valves, or the reasons for not requiring post-accident valve operations. This information is being provided to the NRR staff for resolution. The NRC Region I staff will review the implementation of licensee corrective actions during a subsequent inspection, as discussed in paragraph 4.e.(3).

(4) Required Action: Unlock and open valves DH-V12 A & B and DH-V64 for boron precipitation control.

Corrective Actions: Change valves DH-V12 A & B to electric motor-operated. Operate DH-V64 via reach rod extension on the 305' elevation of the Auxiliary Building. Extension stem is located so that the operator is protected by the above noted shield wall.

Updated Corrective Actions:

Remote Operators for DH-V12 A & B

Electric motor operators are to be retrofitted to decay heat suction (hot leg drop line) isolation valves DH-V12 A & B. The 460 V 3 phase power shall be from a Class 1E source. The operators shall fail "as is" upon loss of all electrical power. Modifications cannot be completed until Cycle 6 refueling, when the licensee expects to have material delivery and plant conditions necessary to complete the installation.

Remote Operation of DH-V64

A remote manual floor stand operator for DH-V64 is located in Area IV on elevation 305' of the auxiliary building. The calculated radiation dose rate in Area IV permits operator travel and a 5 minute stay time for valve operation no earlier than 11 hours after a LOCA. The total accumulated dose for this operation is less than 5 Rem.

Status:

Remote Operators for DH-V12 A & B

Modifications not completed. The licensee is currently seeking approval for deferring modifications for DH-V12 A & B until Cycle 6 refueling. The post-accident operational need for these valves is discussed in paragraph 4.c of this report. This information is being provided to the NRR staff for consideration in their review of the licensee's request. The NRC Region I staff will review the final installation subsequent to licensee completion of the valve modifications, as discussed in paragraph 4.e.(4).

Remote Operation of DH-V64

The licensee installed a 3-1/2" thick steel plate assembly (see paragraphs 4.d(2) and 5), which permits post-accident access to the remote floor stand operator for DH-V64 no earlier than 11 hours after a LOCA. At that time the post-accident dose rate in the area was calculated to have decayed to 53 Rem/hour. This would result in a dose of less than 5 Rem for a 5 minute stay time to operate DH-V64. DH-V64 is a manual isolation valve in the pressurizer auxiliary spray line, which might be used for one of the long term recirculation modes about 24 hours after a LOCA. Therefore, the existing configuration of DH-V64 is acceptable with respect to the design dose rate criteria of NUREG-0737. However, the inspector noted it appeared that DH-V64 could be operated with minimal dose to the operator prior to establishing the RB sump recirculation mode. Licensee representatives stated that post-accident emergency procedures would be reviewed to determine the ability to operate DH-V64 prior to establishing RB sump recirculation. If procedure revisions are appropriate, the revisions should be implemented prior to restart, as discussed in paragraph 4.e(2)(e).

e. Findings

(1) The licensee's shielding design review methodology and calculations were acceptable, and the inspector had no further questions in that area.

(2) The licensee's emergency and operating procedures should be reviewed and revised, as appropriate, with respect to the following matters.

(a) As noted in paragraph 4.c, EP 1202-6B and OP 1104-4 allow a long term recirculation mode that NRC staff stated should not be attempted, based on eva ion of B&W topical safety analysis reports. If licensee anagement considers that procedure revisions are not appropriate, additional NRC staff review and evaluation is required. This matter is being brought to the attention of the NRR staff for information at this time.

(b) As described in paragraph 4.c., the licensee's procedures for small break and large break LOCA's (EP 1202-6B, EP

1202-6C, and OP 1104-4) include requirements for post-accident manual valve operations that could result in personnel doses substantially greater than 5 Rem.

The licensee is aware of this problem and has planned appropriate corrective actions by installation of remote-operated valves. However, pending completion of these modifications, the procedures should be revised to reduce dose rates as low as reasonable achievable. This can be accomplished by incorporating route maps into these procedures that illustrate methods of obtaining access to vital areas, by rearranging specific steps of the procedures, by providing permanent or temporary shielding to manual valves prior to their operation, or by implementing other post-accident procedural controls.

(c) As stated in paragraph 4.d.(1), the licensee's planned revisions to EP 1202-6B and EP 1202-6C, that require manual opening of MU-V198 before going to the RB sump recirculation mode, should be implemented prior to restart.

(d) As stated in paragraph 4.d(3), the licensee's planned revision to OP 1104-4, that allow DH-V15 A & B to remain open but would close DH-V5 A & B and DH-V6 A & B, requires review by the licensee to ensure such changes are correct, and an appropriate revision should be implemented prior to restart.

(e) As stated in paragraph 4.d (4), post-accident emergency procedures should be reviewed to determine the ability to operate DH-V64 prior to establishing RB sump recirculation. Procedure revisions, if appropriate, should be implemented prior to restart.

(f) Licensee action with respect to the emergency and operating procedure reviews and revisions noted above is considered an unresolved item (289/82-13-01).

(3) As stated in paragraph 4.d.(3), the licensee's corrective actions to allow operation of DC-V2 A & B and DC-V65 A & B have not been determined. This matter is being identified to the NRR staff for resolution. Licensee implementation of corrective actions, if needed, will be reviewed during a subsequent NRC Region I inspection. (289/82-13-02)

(4) As stated in paragraph 4.d.(3) and 4.d.(4), the licensee's modifications for valves DH-V19 A & B, DH-V38 A & B, and DH-V12 A & B are not completed. The final installation will be reviewed during a subsequent NRC Region I inspection. (289/82-13-03)

5. TMI-1 Restart Modifications - Implementation

a. General

The inspector reviewed selected facility modifications (listed below) which are required to be completed prior to TMI-1 restart to verify that the new designs are provided consistent with the following items.

-- licensee commitments stated in the TMI-1 Restart Report, "Report in Response to NRC Staff - Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1"

-- requirements delineated in NUREG-0680 (and supplements), "TMI-1 Restart Evaluation Report, to comply with NRC Commission Order of August 9, 1979"

-- requirements delineated in ASLB Partial Initial Decision (PID), "Plant Design and Procedures and Separation Issues," dated December 14, 1981

-- TMI-1 Operational Quality Assurance Plan, Revision 9

-- Administrative Procedure (AP) 1043, "Control of Plant Modifications," Revision 3

The inspector verified that the modification task was installed in accordance with the approved design based upon observation of completed work, review of related portions of the licensee's QA program, examination of installation records, rev'ew of nondestructive examination (NDE) and/or other inspection records, and other related documentation. Specific modification task observations and records reviewed by the inspector are identified below.

(b) Modification Task LM-51A, Post-Accident Shielding

(1) Description

Task LM-51A added an L-shaped reinforced concrete shield wall and local steel plate shielding in the Auxiliary Building at elevation 305'. The shielding will protect Motor Control Centers 1A and 1B and allow personnel access to them under post-accident conditions. The shielding is designed to account for the following sources.

-- Seal injection line (penetration 337)

-- Two high pressure injection lines (penetrations 338 and 339)

-- Seal injection filters and valve station

-- Makeup suction line below the 305' slab at elevation 292'

In addition LM-51A provided certain electrical modifications to allow construction of the shield wall. These modifications included (1) relocation of an Auxiliary Building Ionization Detector Control Unit, (2) relocation of Lighting Panel AB-2, and (3) installation of 3/4" conduit imbedded in the shield wall.

(2) Review/Observation

The inspector reviewed selected portions of GPU Nuclear Corporation Engineering Change Modification (ECM) S-242, accepted August 2, 1982, and ECM S-230, accepted May 18, 1982. The inspector observed the installed shield wall and steel plate shielding and verified the component location and installation was as described in applicable modification documents.

(3) Findings

Based on the modification documents reviewed and observation of the completed installed, the inspector determined that Task LM-51A was satisfactorily completed in conformance with the referenced commitments and requirements.

No violations were identified.

6. Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, violations or deviations. An unresolved item is discussed in paragraph 4.e.(2).

7. Exit Interview

The inspector met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection on September 3, 1982, to discuss the inspection scope and findings.