TABLE OF CONTENTS

		Page
1.0	INTRODUCTION AND REPORT ORGANIZATION	1-1
1.1	Introduction	1-1
1.2	Report Organization	1-1
1.3	Abbreviations	1-2
1.4	Definitions	1-2
2.0	PLANT MODIFICATIONS	2.1-1
2.1	General	2.1-1
2.1.1	Short-Term Modifications	2.1-1
2.1.1.1	Reactor Trip on Loss of Feedwater/ Turbine Trip	2.1-1
2.1.1.2	Position Indication for PORV and Safety Valves	2.1-3
2.1.1.3	Emergency Power Supply Requirements for Pressurizer Heaters, PORV, Block Valve, and Pressurizer Level Indication	2.1-5
2.1.1.4	Post LOCA Hydrogen Recombiner System	2.1-8
2.1.1.5	Containment Isolation Modifications	2.1-11
2.1.1.6	Instrumentation to Detect Inadequate Core Cooling	2.1-17
2.1.1.7	Auxiliary Feedwater Modifications	2.1-20
2.1.1.8	Leak Reduction Program For Systems Outside Containment	2.1-29a
2.1.2	Long-Term Modifications	2.1-30
2.1.2.1	Post Accident Monitoring	2.1-30
2.1.2.2	RCS Venting	2.1-31
2.1.2.3	Plant Shielding Review	2.1-31
2.1.2.4	Post Accident Sampling Capability	2.1-33
2.1.2.5	Reactor Coolant Pump Trip on HPI	2.1-34

Am. 16
8004150351

generated when any of the valves open. Calculations have been made for saturated, liquid and two phase flow. A summary of these calculations is provided in Appendix 2A. Tests run by B&W on the electromatic relief valve under reduced flow condiitons have confirmed the validity of this approach. Because of the straight-forward and well known relationships that exist between flow conditions and differential pressure across the elbow, the signal from one differential pressure transmitter can be confidently predicted for any flow conditions. For this reason it has been concluded that operating tests, which would be difficult since they involve opening the PORV and relief valves, will not be required.

Acoustic monitoring of the electromatic relief valve makes use of well proven equipment and techniques which have been used in the B&W Loose Parts Monitoring System. Tests run on this valve at the B&W Alliance facility demonstrated that the acoustic monitoring system gave satisfactory results.

2.1.1.2.5 Safety Evaluation

Instrument taps will be installed on elbows in the discharge piping of pressurizer code safety valves RC-RV1A and RC-RV1B and electromatic relief valve RC-RV2. This piping is classified as N2, Seismic I. Analysis has been performed to demonstrate that this modification will not degrade the integrity of the existing pipe. The pipe classification has been maintained up to and including the instrument root valves. The mounting of new equipment which will be located in the vicinity of safety related systems has been analyzed to ensure that no hazardous missiles will be generated in a seismic event. It has been concluded that this modification will not degrade any safety related systems. All of the equipment inside containment for Pressurizer PORV and safety valve detection will be seismically and environmentally qualified. Work is underway to upgrade the portion of the system outside containment. This involves specifying and procuring of additional equipment. This should be installed by January 1981.

2.1.1.2.6 Instrumentation

The output signals from the three differential pressure transmitters will be displayed on indicators in the control room. They will be calibrated in "inches of water". Each signal will also go to an alarm bistable. A control room alarm will be initiated if any of the signals exceed a pre-determined value. This will alert the operator that one of the valves is open. The differential pressure signal will also be monitored by the plant computer for logging, trending, and alarm functions.

The outputs from the accelerometers which will be mounted on RC-RV2 will be processed by monitoring equipment installed in the existing Loose Parts Monitoring Cabinet. An output signal indicative of flow through the valve will be displayed and recorded locally. A control room alarm will be initiated if flow is detected. This signal will also be monitored by the plant computer for logging, trending, and alarm purposes.

2.1.2 Long Term Modification

2.1.2.1 Post Accident Monitoring

2.1.2.1.1 System Description

Certain post accident monitoring capability will be provided in compliance with Reg. Guide 1.97, Rev.2 as discussed below. Pending the availability of appropriately qualified instrumentation and equipment, the following modifications will therefore be completed by January 1, 1981. The conceptual design will be provided for NRC review by January 1, 1980.

Containment Pressure - Continuous containment pressure indication will be provided in the control room using a range from -5 psig to three times the design pressure of the containment. The pressure indication will be safety grade and will meet the design and qualification requirements of Reg. Guide 1.97. Redundant indication of pressure will be provided.

Containment Water Level - Continuous containment water level indication shall be provided in the control room. A safety grade wide range indicator from the bottom of containment to a level of 10 feet will be installed in accordance with the requirements of Reg. Guide 1.97. In addition, a narrow range indicator from the bottom to the top of the sump with continuous indication in the control room shall be installed which meets the requirements of Reg. Guide 1.89 and is capable of being periodically tested.

Containment Hydrogen Indication -Safety grade continuous indication of containment hydrogen will be provided in the control room. The range of indication will be 0-10% concentration assuming commercial availability over this range.

High Range Containment Radiation Monitor - Two safety grade containment radiation monitors that are physically separated shall be provided with recording display and continuous indicator presentation in the control room. The range of this monitor shall be 10^7 R/hr and shall detect photon radiation down to 60 Kev. The design of the radiation monitors shall be provided in accordance with Reg. Guide 1.97 Rev. 2 (Dec. 1979). To our knowledge, manufacture of appropriately qualified equipment to satisfy these requirements will commence by July 1980. They will be installed and operational by January 1, 1981.

High Range Effluent Monitor - One high range effluent monitors intended as the Long-Term modification shall be oeprational for each normal gas release point by January 1, 1981.

The range of these monitors shall be as follows:

- o Undiluted Containment Exhaust . . . . . 10^5uCi/cc
- o Main Steem Lines . . . . . 10^2uCi/cc
- o Auxiliary & Fuel Handling Building Exhaust . . . 10^3uCi/cc
- o Condenser OFF GAS Exhaust . . . . . 10^5uCi/cc

Regulatory Guide 1.97 Rev. 2 (Dec. 1979) will be followed for the design of high range effluent monitors. Vital bus power shall be employed for each system's modular assembly with the normal power supplying the monitor pumps with diesel generations as back ups. Further descriptions of increased range capabilities are provided in Section 2.1.2.

High Range Effluent Radio Iodine & Particulate Sampling Analysis - The existing sampling system will be expanded and will include the addition of silver zeolite cartridges. The system design and operation will both decrease the activity on the cartridges so they can be handled and will decrease the xenon to iodine ratio. Counting of the cartridges will be by use of NaI crystal connected to a single or dual channel analyzer with appropriate window and discrimination settings for th 364 Kev gamma of I-131, or by use of a GELI/MCA system. The expanded portion of the sampling system would be placed in service following an accident and will be located in an applicable area exhibiting low background. The system will be on site and operable by 1 January, 1981.

Prior to incorporation of the expanded sampling system, procedures will be developed for the use of silver zeolite cartridges and normal particulated filters for sampling with a NaI detector and a single or dual channel analyzer for iodine and gross particulate release rate determination. Specific details to insure exposures are maintained as low as reasonably achievable will be incorporated into the procedures.

These procedures will be available for NRC review prior to restart or 1 October 1980 whichever occurs first.

2.1.2.2 RCS Venting

2.1.2.2.1 System Description

Power operated vents will be provided for the reactor coolant system in order to ensure that natural circulation and adequate core cooling can be maintained following an accident. The vents will be from the top of the pressurizer and the top of both hot legs using existing connections on the reactor coolant piping. The discharge from the hot leg vents will be directed to the containment atmosphere. The system is shown schematically in Figure 2.1.-11.

The vents to the containment atmosphere will tie into existing hot leg vent piping outside the secondary shield wall. As part of this modification, remote operation of the vent valves in the existing vent line from the pressurizer to the reactor coolant drain tank will be provided and the system will retain the existing venting capability. Control and position indication for the power operated vent valves will be provided in the control room.

The key parameters to be monitored with on-line instrumentation include containment hydrogen concentration, reactor coolant boron concentration and letdown failed fuel monitors. The on-line hydrogen monitoring capability has been previously described in this restart report. An on-line boronmeter will be installed. The conceptual design and schedule for installation will be forwarded to the NRC by March 1, 1980. The existing reactor coolant system letdown monitors will remain on scale with up to 10% failed fuel based on the FSAR definition of failed fuel. This existing monitor is deemed adequate as an indicator that significant core damage has occurred.

A design and operational review of the reactor coolant and containment atmosphere sampling systems shall be performed. Modifications shall be completed as necessary to ensure that personnel can obtain samples under accident conditions without incurring a radiation exposure to any individual in excess of 3 rems to the whole body and 18 3/4 rems to the extremities. The source terms to be considered shall be those previously listed under the Design Basis of Plant Shielding. In addition, a design and operational review of the radiological spectrum analysis facilities and the chemical analysis facilities will be conducted in order to identify any additional design features or shielding required to ensure that confirmatory samples can be obtained and analyzed within the 8 hour period previously mentioned. The chemical analyses to be considered shall include both boron and chloride analysis. The conceptual design for required modifications will be forwarded to the NRC by June 1, 1980.

RCS sampling can be accomplished within one hour and analyzsed in an additional hour under normal circumstances. Met-Ed's experience sampling under accident conditions at TMI-2 has taught us that slow deliberate steps are necessary to prevent personnel overexposures. Since an early indication of significant fuel failure is obtained from the failed fuel monitor and an on line baronometer is being provided it is considered unnecessary and imprudent to attempt to draw a confirmatory sample in less than 8 hours. All necessary and appropriate operator and emergency actions can be taken based on the early indications. In addition chloride analysis provides information which is only useful in the long term and therefore is not immediately necessary.

2.1.2.5 Reactor Coolant Pump Trip on HPI

2.1.2.5.1 System Description

The purpose of this proposed modification is to provide automatic trip of the Reactor Coolant Pumps when degraded primary system conditions associated with a LOCA have been detected. This will be accomplished by requiring that RCP trip be initiated when the Engineered Safeguards System has actuated Safety Injection and an increasing RC void fraction has been detected as indicated by low RC pump motor current. The proposed logic will preclude RC pump trip during those events such as severe overcooling or very small breaks where maintenance of forced cooling is very

THREE MILE ISLAND UNIT NO. 1

TABLE 2.1-2

LIST OF CONTAINMENT ISOLATION VALVES REQUIRING MODIFICATIONS

Penetration No.	Service	System	Valve Tag No.	Valve Type	Line Size, In.	Method of Actuation	Normal Valve Position	Post Accident Position		Valve Actual Position Indication	Valve Actuation Signal Source		Notes
								Existing	Modified		Existing	Modified	
108	Containment Air Sample	RM	CM-V1	Ball	1	Air	Open	Closed	Closed	Yes	1,10	1,2,6,10	
			CM-V2	Ball	1	Air	Open	Closed	Closed	Yes			
			CM-V3	Ball	1	Air	Open	Closed	Closed	Yes			
			CM-V4	Ball	1	Air	Open	Closed	Closed	Yes			
213	Steam Generator Sample	CA	CA-V4A	Globe	3/8	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	
			CA-V5A	Globe	3/8	Air	Open	Closed	Closed	Yes			
214	Steam Generator Sample	CA	CA-V4B	Globe	3/8	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	
			CA-V5B	Globe	3/8	Air	Open	Closed	Closed	Yes			
302	Intermediate Cooling Water Outlet Line	IC	IC-V2	Gate	6	EMO	Open	Closed	Open/Closed	Yes	1,3,10	3,7,8,9,10	
			IC-V3	Gate	6	Air	Open	Closed	Open/Closed	Yes			
307	Demin. Water to Reactor Building	CA	CA-V189	Gate	2	Air	Open	Closed	Closed	Yes	1,10	1,5,10	
309	Letdown Line to Purification Demineralizers	MU	MU-V2A	Globe	2-1/2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	
			MU-V2B	Globe	2-1/2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,6,10	
			MU-V3	Gate	2-1/2	Air	Open	Closed	Closed	Yes	1,10	1,6,10	
323	RC Makeup	MU	MU-V18	Gate	2-1/2	Air	Open	Closed	Closed	Yes	1,10	1,2,10	
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1	Globe	3/8	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,6,10	
			CA-V2	Gate	3/8	Air	Closed	Closed	Closed	Yes			
			CA-V3	Globe	3/8	EMO	Closed	Closed	Closed	Yes			
			CA-V13	Globe	3/8	EMO	Closed	Closed	Closed	Yes			
329	Reactor Coolant Pump Seal Return	MU	MU-V25	Globe	4	EMO	Open	Closed	Open/Closed	Yes	1,7,10	3,7,8,10	
			MU-V26	Gate	4	Air	Open	Closed	Open/Closed	Yes			
330	Reactor Coolant Drain Tank Vent	WDG	WDG-V3	Globe	2	EMO	Open	Closed	Closed	Yes	1,10	1,4,5,10	
			WDG-V4	Gate	2	Air	Open	Closed	Closed	Yes			
331	Reactor Coolant Drain Tank Pump Discharge	WDL	WDL-V303	Gate	4	EMO	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			WDL-V304	Gate	4	Air	Closed	Closed	Closed	Yes			
333	Intermediate Cooling Water Supply Line	IC	IC-V4	Gate	6	Air	Open	Closed	Open/Closed	Yes	1,3,10	3,7,8,9,10	
334	Intermediate Cooling to CRDM Cooling Coils	IC	IC-V6	Gate	3	Air	Open	Closed	Closed	Yes	1,3,10	3,7,8,9,10	

Am. 16

THREE MILE ISLAND UNIT NO. 1
TABLE 2.1-2 (CONT'D)
LIST OF CONTAINMENT ISOLATION VALVES REQUIRING MODIFICATIONS

Penetration No.	Service	System	Valve Tag No.	Valve Type	Line Size, In.	Method of Actuation	Normal Valve Position	Post Accident Position Existing	Post Accident Position Modified	Valve Actual Position Indication	Valve Actuation Signal Source Existing	Valve Actuation Signal Source Modified	Notes
336	Reactor Building Outlet Purge Line	AH	AH-V1A	Butterfly	48	Air	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			AH-V1B	Butterfly	48	EMO	Closed	Closed	Closed	Yes			
346	Reactor Coolant Pump Motor Cooling Water Supply	NS	NS-V15	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10,	
347	Reactor Coolant Pump Motor Cooling Water Return	NS	NS-V4	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10	
			NS-V35	Gate	8	EMO	Open	Closed	Open/Closed	Yes	1,10	7,8,9,10,	
353	Reactor Building Sump Drain	WDL	WDL-V534	Gate	6	Air	Closed	Closed	Closed	Yes	1,10	1,4,5,10	
			WDL-V535	Gate	6	Air	Closed	Closed	Closed	Yes			
421	Reactor Building Normal Air Coolers Supply Line	RB	RB-V2A	Gate	8	EMO	Open	Closed	Open	Yes	1,10	1,2,10;	Add auto initiation of Emerg. R.B. cooling on 4 psig R.B. and 1600 psig R.C. pressure isolation signals.
422	Reactor Building Normal Air Coolers Return Line	RB	RB-V7	Gate	8	Air	Open	Closed	Open	Yes	1,10	1,2,10;	
423	Reactor Building Inlet Purge Line	AH	AH-V1C	Butterfly	48	EMO	Closed	Closed	Closed	Yes	1,4,10	1,4,5,10	
			AH-V1D	Butterfly	48	Air	Closed	Closed	Closed	Yes			
348,349	Core Flood TK. Sample and N_2 Fill Lines	CF	CF-V2A&B	Globe	1	EMO	Closed	Closed	Closed	Yes	1,10	1,5,10	
			-V19A&B	Gate	1	Air							
			-V20A&B	Gate	1	Air							

Valve Actuation Signal Source

1) 4 psig reactor building pressure isolation
2) 1600 psig (SFAS) isolation
3) Radiation alarm, operator action required
4) High radiation (non-safety) isolation
5) Reactor trip isolation
6) Override capability on individual valves
7) Classify line to Seismic Category I
8) 30 psig reactor building pressure isolation
9) Line break isolation signal
10) Remote manual control

Am. 16

THREE MILE ISLAND UNIT NO. 1

TABLE 2.1-3

LIST OF CONTAINMENT PENETRATIONS REQUIRING ISOLATION ON HI-RADIATION

Penetration No.	Service	System	Isolation Valve Tag No.	Radiation Detector Location	Type of Monitor
213 and 214	Steam Generator Sample	CA	CA-V4A -V5A -V4B -V5B	Locate the monitors outside the R.B. near the sampling line downstream of the containment isolation valve and upstream of connection for Turb. Plant sampling	Area Gamma Detectors (New)
309	Letdown Line to Purification Demineralizers	MU	MU-V2A -V2B	Utilize existing Rad. Monitor RM/L-1 located outside R.B.	Inline (Existing)
328	Pressurizer and Reactor Coolant Sample Lines	CA	CA-V1 -V2 -V3 -V13	Locate the monitor outside the R.B. between the isolation valve and the sample cooler.	Area Gamma Detector (New)
329	Reactor Coolant Pumps Seal Return	MU	MU-V33A -33B -33C -33D	Locate the online radiation monitor downstream of the containment isolation valves outside of the R. B. for Alarm Operator action is required to close valves.	Area Gamma Detector (New)
330	Reactor Coolant Drain Tank	WDG	WDG-V3	Locate the monitor on the outside of the tank.	Area Gamma Detector (Existing)
and	Vent		-V4		
331	Reactor Coolant Drain Tank	WDL	WDL-V303		
	Pump Discharge		-V304		
336	Reactor Building Outlet and	AH	AH-V1A -V1B	Utilize the existing purge outlet line Rad. Monitor RM/A-9 located outside of R.B.	Inline (Existing)
and 423	Inlet Purge Lines		-V1C -V1D		
353	Reactor Building Sump Drain	WDL	WDL-V534 -V535	Locate an area radiation monitor in the R.B. Sump mounted inside a seismically supported pipe.	Sump Area Monitor (New)
302	Intermediate Cooling	IC	IC-V2,3	Locate the radiation monitor on the 6" IC return line between valve IC-V3 and the 2" pump recirc. line.	Strap on GM (New)
333 and 334	Supply & Return		-V4,6		

Am. 16

THREE MILE ISLAND UNIT NO. 1

TABLE 3

List of Isolation Signal Override Capability

	Penetration No.	Isolation Signal: Reactor Trip	High Radiation	4 psig Building	30 psig Building	1600 psig (SFAS)	Line Break
Containment Air Sample	108	N/A	N/A	C	N/A	C	N/A
R.B. Sump	353	C	IB	C	N/A	N/A	N/A
RCDT	330,331	C	IB	C	N/A	N/A	N/A
RCS Sample	328	C	IB	C	N/A	N/A	N/A
R.B. Purge	336,423	C	NO	NO	N/A	N/A	N/A
RCS Letdown	309	A	IB	C	N/A	N/A	N/A
RCS Makeup	323	N/A	N/A	C	N/A	C	N/A
Demin Water	307	C	N/A	C	N/A	N/A	N/A
OTSG Sample	213, 214	C	IB	C	N/A	N/A	N/A
NSCCW	346, 347	N/A	N/A	N/A	NO	N/A	NO
ICCW	302, 333, 334	N/A	N/A	N/A	NO	N/A	NO
R.B. Air Coolers	421, 422	N/A	N/A	C	N/A	C	N/A
Core Flood TK	348,348	C	N/A	C	N/A	N/A	N/A

Legend
- C = Common Signal Override; initiating isolation condition may still exist.
- I = Individual isolation signal override capability; procedures governing override to be developed.
- IB = Individual isolation signal bypass capability
- A = Automatic isolation signal override.
- NO = No override or bypass capability; initiating condition must clear to allow reopening of valve.
- N/A = Not applicable.

Note: For combinations of initiating signals that are allowable, refer to Table 2.1-2.

Am. 16

and is trucked offsite for disposal at a municipal sewage treatment facility. In the event that any TMI-2 sanitary facilities become contaminated with radioactivity, the ability to remove TMI-01 sanitary waste from the site would not be reduced.

7.2.6 Radiation Protection and Decontamination Areas

TMI-1 has, installed as permanent plant equipment, all facilities necessary to support the radiation protection and decontamination activities of a single unit. Any facilities in TMI-1, that in the past had been shared by both units, will be used exclusively by TMI-1 subsequent to startup. Since the accident, TMI-2 has established a separate radiation protection organization utilizing dedicated personnel and equipment. External facilities have been established under the recovery program to replace facilities previously shared with TMI-1.

7.2.7 Nuclear Sampling and Radiochemistry Laboratory

The Temporary Sample Sink System consists of tubing valves and other equipment necessary to satisfy all Unit II sampling requirements without utilizing the Unit I primary sample lab.

Tubing and valves will be provided to facilitate sampling the following; Reactor Coolant Bleed Tanks, Miscellaneous Waste Holdup Tank, Mini Decay Heat System, Pressurizer Steam Space, Pressurizer Water Space, Reactor Coolant Letdown and the Fuel Pool Waste Storage System.

The system will include all necessary tubing and equipment to allow adequate sample recirculation from the sample source back to that source. In addition, any purge required into the Sample Sink will be contained in a drain cask at the sink. This cask will be pumped to the Reactor Coolant Bleed Tank when the level reaches a predetermined point.

The system will also provide an in line Boronometer for measuring boron concentration in the samples drawn from the Unit II RCS. The Boronometer will be equipped with its own calibration equipment for periodic, on-line calibration.

The entire sink will be shielded and enclosed with its own independent ventilation system which will exhaust into the Unit II Auxiliary Building Ventilation System.

The TMI-2 Nuclear Sampling Facility is scheduled to be operational by June 1, 1980.

7.2.8 Industrial Waste Treatment Facilities

The Industrial Waste Filter System (IWTS) and Industrial Waste Treatment System (IWTS) have been installed to comply with the

7.3.5.2 Sample Drains

The TMI-1 Nuclear Sampling sample and analysis drains are routed to the TMI-1 Auxiliary Building Sump to be processed by the Liquid Radwaste System. In the event of a plant accident accident water including reactor coolant would be discharged to the sump as the result of sampling and analysis. Since the sump is not sealed radioactivity from the accident would escape uncontrollably into the Auxiliary building from the sump.

To prevent this from happening, modifications will be made to pipe the radiochemical laboratory drains to the Miscellaneous Waste Storage Tank either directly or by way of an intermediate collection tank and pump(s). The Miscellaneous Waste Storage Tank would contain the laboratory wastes because the tanks gas space is connected to the vent header. Intermediate tanks and pumps that may be used would also be vented to the waste gas system.

The laboratory waste collection modification will be operational by October 1, 1980.

7.3.5.3 Improved In-Plant Radioiodine Monitoring Instrumentation

See section 2.1.2.1.1.

7.4 Affect of TMI-2 Recovery on TMI-1 Operation

Activities in TMI-1 related to radwaste processing and activities in the common fuel handling building will not be affected by the TMI-2 recovery program. As demonstrated in Section 7.2.3, TMI-1 does not have to rely on any TMI-2 facilities for the processing of radwaste. Section 7.2.1 describes specifics to be taken to isolate the radwaste piping systems of the two units. Through the isolation of the piping system, interface between the two unit's radwaste systems will be eliminated. Waste processing activities related to TMI-2 will be performed in the fuel handling building during the recovery program. These activities will not affect activities in the auxiliary building because the areas will be separated by an environmental barrier (Section 7.2.2). Communication of the air spaces (TMI-1/TMI-2) of the fuel handling building will be minimized with appropriate modifications of the ventilation equipment in the building. Continuous access to the fuel handling building is not required for the safe operation of TMI-1 (with environmental barrier installed).

8.3.7 Loss of Electric Power (FSAR Section 14.1.2.8)

1. Description

Separation of the unit from the transmission network results in the trip of the turbine. A more severe transient occurs if the ICS does not run back the reactor load demand. The result is reactor trip on high pressure. Cooldown is accomplished through the atmospheric dump or steam relief valves. In the presence of failed fuel and primary to secondary leaks, this event can lead to low levels of radioactivity release.

2. Acceptance Criteria

i. DNBR shall not be less than 1.3.

ii. Reactor coolant system pressure will not exceed code allowable limits of 2750 psig.

3. Mitigation

i. Reactor trip on high pressure.

4. Conclusion

This transient has an increased safety margin over the analysis performed in the FSAR as a result of the high pressure trip setpoint reduction to 2300 psig and the anticipatory reactor trip with turbine trip. In addition, a PORV setpoint of 2450 assures that the PORV will not be activiated (Ref. 1).

8.3.8 Station Blackout (Loss of AC) (FSAR Section 14.1.2.8)

1. Description

All AC power to the unit is lost, with only battery power available. The reactor and turbine trip, and reactor coolant and feedwater pumps are lost. Core cooling is accomplished through heat rejection to the secondary side using the turbine driven emergency feedwater pump with steam relief to the atmosphere. The analysis is performed starting at full power 2535 Mw (t), and takes credit for a condensate inventory of 200,000 gallons. NNI and ICS instrumentation is taken credit for in controlling the plant when it is powered from the vital ac inverters.

2. Acceptance Criteria

i. DNBR is not less than 1.3.

3. Mitigation

i. Reactor trip on low pressure or high neutron flux.

ii. Feedwater isolation of the affected OTSG as a result of low steam generator pressure.

iii. Isolation of the unaffected steam generator by the turbine stop valves.

iv. Decay heat removal through the unaffected OTSG by manual control of emergency feedwater (Procedure 2203-2.3) and either atmospheric dump valves or the turbine bypass valves if they are available.

v. Containment temperature and pressures are limited by the containment fan coolers (and reactor building spray systems if reactor building pressure exceeds 28 psig).

4. Conclusion

Recent, detailed analyses of TMI-2 (Refs. 5 through 8) allow broader conclusions about the acceptability of TMI-1 regarding steam line break. The TMI-2 analysis considered additional single failures, the most limiting were the feedwater regulating and turbine stop valve failures. In addition, the reactor core performance was analyzed assuming that: feedwater is not isolated, offsite power is available if results are worse for that case, and both steam generators blow down outside containment. Reference 3 explains why the TMI-2 core performance analysis bounds Unit 1.

At the Cycle 5 refueling outage, the feedwater latching signal was added to the upstream block valves (FW-V-5A/B). The TMI-2 feedwater regulating valve and turbine stop valve failures cases thus bound the TMI-1 design. Although these failures are not a licensing basis for the plant, they do demonstrate the additional safety margins available in this accident.

The difference in design of the main steam isolation valves between TMI-1 and TMI-2 results in less severe containment transients for TMI-1. The Unit 1 valves are a stop/check design, so that they would prevent the blowdown of both steam generators inside containment. The TMI-1 design isolates emergency feedwater on low steam generator pressure (600 psig). The TMI-2 containment analyses bounds since it was performed assuming 1250 gpm EFW flow beginning at 2 sec and continuing throughout the accident.

TMI-1 has not analyzed the environmental effect inside containment for the worst case single failure (because of the

stop/check MSIV's, the worst failure is the feedwater regulating valve failure). As noted previously, the blowdown will be less severe than for Unit 2. This issue will be resolved in response to I&E Bulletins 80-01 and 79-22. There are several reasons to expect acceptable results.

i. Heat shrink tubing is being added to splices inside containment. This change was made to TMI-2 prior to receipt of the operating license to resolve this concern.

ii. Much of the equipment which was analyzed and shown acceptable for TMI-2 is also used on TMI-1.

The radiological consequences of the unmitigated steam line break accident have also been addressed on the TMI-2 docket (Ref. 6 and 7). These analysis results demonstrate that worst case doses from a steam line break accident are within the limits of 10CFR100.

8.3.10 Steam Generator Tube Failure (FSAR Section 14.1.2.10)

1. Description

The rupture of a steam generator tube concurrent with 1% failed fuel results in the release of radioactive steam to the environment via the condenser air ejector. Leakage is greater than the capacity of the makeup system, so that the RCS depressurizes.

2. Acceptance Criteria

i. Doses are less than 10CFR100 limits.

3. Mitigation

i. Reactor trips on low pressure.

ii. High pressure injection initiates and maintains primary system pressure and inventory.

iii. Turbine trip isolates the steam generator, and the release path of steam to the environment is via the turbine bypass line, through the condenser to the air ejector.

iv. Cooldown is achieved first via the unaffected steam generator and then through the decay heat cooling system.

4. Conclusions

There have been no plant changes which change the results of this analysis. Results are still valid and acceptable.

REFERENCES

1. Three Mile Island Unit 1 Nuclear Station, Final Safety Analysis Report, USNRC Docket No. 50-289.

2. "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," Volumes I & II, Babcock and Wilcox, May 7, 1979.

3. "GPUSC Safety Evaluation Report for Three Mile Island Unit 1 Cycle 5 Reload," dated March 1979.

4. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid) on "High Pressure Trip and Pressurizer Code Safety Valve Settings," GQL-0669, April 17, 1978.

5. "Supplement No. 2 to the Safety Evaluation Report by the office of Nuclear Reactor Regulation, Three Mile Island Nuclear Station Unit No. 2, Docket Number 50-320," USNRC, NUREG 0107, dated February, 1978.

6. Letter, Met-Ed (J. G. Herbein) to USNRC (S. A. Varga), on "Analysis of Fuel Performance During a Steamline Break for TMI-2," License No. CPPR-66, Docket No. 50-320, dated November 18, 1977.

7. Letter, Met-Ed (R. C. Arnold) to USNRC (S. A. Varga), on "Response to Staff Questions on Analysis of Fuel Performance During a Steamline Break," dated December 9, 1977.

8. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid) on "TMI-1 Fuel Handling Accident Inside Containment," GQL-0460, dated April 20, 1977.

9. Letter, USNRC (R. W. Reid) to Met-Ed (J. G. Herbein), dated February 4, 1979.

10. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid) on "TMI-1 Fuel Handling Accident Inside Containment, "GQL-0460, dated May 8, 1979.

11. ECCS Analysis of B&W's 177-FA Lowered Loop NSS, BAW-10103, Rev. 2, Babcock & Wilcox, April 1976.

12. USNRC to Met-Ed "Order for Modification of License," Docket No. 50-289, May 19, 1978.

13. Letter, Met-Ed (J. G. Herbein) to USNRC (R. W. Reid), on "Small Break LOCA," GQL-0809, May 3, 1978.

14. Safety Evaluation and Environmental Impact Appraisal by the Office Nuclear Reactor Regulation, Supporting Amendment No. 65 to Facility Operating License No. DPR-47, Amendment No. 62 to Facility Operating License No. DPR-55 Duke Power Company, Oconee Nuclear Station, Units Nos. 1, 2 and 3, Docket Nos. 50-269, 50-270, and 50-287, October 23, 1978.

15. TMI-1 Fuel Densification Report, BAW-1389, Babcock & Wilcox, June 1973.

(3) Draft Technical Specification 4.9.1.3 would require valve verification (correct position and locked, if appropriate) for valves in the emergency feedwater system, every 31 days. In addition, locked valves could only be maintained in an unlocked condition under administrative control.

(4) Draft Technical Specification 4.9.1.4 would require a test, each 18 months, of automatic pump start logic and automatic valve lineup following an emergency feedwater actuation signal. In addition, the operability of the manual control valve station would be verified.

(5) Draft Technical Specification 4.9.1.5 would require testing of the EFW injection valves on a quarterly basis.

(6) Item 10F of NRC's October 26, 1979 letter to Mr. R. C. Arnold requires a, "...Technical Specifications to assure that prior to plant startup following an extended cold shutdown, a flow test would be performed to verify the normal flow path from the primary EFW system water source to the steam generators. The flow test should be conducted with EFW system valves in their normal alignment." This test is incorporated in draft Technical Specification 4.9.1.6 where the term "extended cold shutdown" is interpreted as "a cold shutdown of longer than 30 days' duration."

(7) Existing Technical Specification 4.1.2 (Table 4.1-2), would require a functional test of the Backup Instrument Air Supply System (backup air supply for the emergency feedwater control valves), every refueling period.

(8) Existing Technical Specification 4.1.1 (Item 50 in Table 4.1-1), as modified, would require a check each shift, monthly testing, and calibration each refueling period, for the emergency feedwater flow instrumentation. The "check" and "test" surveillances would not be required when T_{AVG} is less than 200°F since the reactor would be shutdown and this safety function not needed.

(9) Existing Technical Specification 4.5.1.1, as modified would incorporate the motor driven feedwater pumps into the list of equipment whose operation is verified during the testing of the emergency diesel generators. In this case, an additional test signal is required to start the pumps since the pumps do not actually start on loss of AC power (the actual start signal is on loss of main feedwater or loss of reactor coolant pumps.)

Conclusion

In conclusion, with regard to the modifictions to the emergency feedwater systems and associated Technical Specifications:

The post-accident monitoring instruments to be installed at TMI-1 are responsive to the recommendations of the NRC and the ACRS.

Evaluation

Section 2.1.2.1 of "Report in Response to NRC Staff Recommended Requirements for Restart of Three Mile Island Nuclear Station Unit 1" describes the post-accident monitoring instrumentation to be installed at TMI-1. The post-accident instrumentation, in conformance with Regulatory Guide 1.97, consists of the following:

(1) Containment Pressure - the range will be - 5 psig to three times the containment design pressure;

(2) Containment Water Level - a narrow range monitor will measure containment sump level while the wide range monitor will measure from the bottom of the containment to a 10 ft. level;

(3) Containment Hydrogen Indication - continuous reading of the concentration of hydrogen in the containment, from 0 to 10%, will be available in the control room;

(4) High Range Containment Radiation Monitor - two monitors with a range to 10^7R/hr will be provided;

(5) High Range Effluent Monitors:

(a) Undiluted Containment Exhaust - 10^5 µCi/cc

(b) Main Steam Lines - 10^2 µCi/cc

(c) Auxiliary and Fuel Handling Building Exhaust 10^3 µCi/cc

(d) Condenser Off Gas - 10^5 µCi/cc

(e) High Range Effluent Radio Iodine & Particulate Sampling and Analysis - silver zeolite cartridges.

Although the above instrumentation does not actuate safety equipment, nor is it required by safety analyses, it is appropriate to provide Surveillance Requirements to assure reliable post-accident performance of the instrumentation. Surveillance Requirements for post-accident monitoring instrumentation is incorporated into TMI-1 Draft Technical Specification 4.1.1 (Table 4.1-1) as follows:

(1) Item 13 of Table 4.1-1, "High Reactor Building Pressure," is provided with a footnote to include the post-accident instrumentation in the existing containment pressure instrumentation surveillance program;

(2) Item 28 of Table 4.1-1, "Radiation Monitoring Systems," is provided with a footnote to include the post-accident instrumentation, described in item (5)(a) thru (5)(d) above, in the existing radiation monitor system surveillance program;

By increasing the Low Reactor Coolant System Pressure setpoint from 1800 psig to 1900 psig, the reactor would trip earlier in the LOCA scenario and thus the decay heat would be slightly less when the ECCS functions. Increasing the Low Reactor Coolant System trip setpoint also has the effect of increasing the margin to DNB following a trip on low pressure; the reactor would trip earlier on low pressure and thus the final minimum DNB would be higher (more conservative) than if the reactor tripped at 1800 psig.

Conclusion

With regard to increasing the Low Reactor Coolant System trip setpoint from 1800 psig to 1900 psig:

(1) The probability or consequences of accidents previously considered have not increased. For any accident that involves a pressure decrease, the reactor will trip earlier in the transient and thus the result of the accident will be more conservative.

(2) No accident of a type not previously evaluated, will occur. The increasing of the Low Reactor Coolant System trip setpoint will not have any effect other than tripping the reactor at an earlier time in pressure reduction transients.

(3) No safety margins have been decreased. It is expected that for pressure reduction transients, DNB following the reactor trip will be higher (more conservative) and for the LOCA, the peak clad temperature and other system parameters will be more favorable.

Based upon the above, we conclude that increasing the Low Reactor Coolant System trip setpoint from 1800 psig to 1900 psig does not involve unreviewed safety questions with regard to the criteria of 10CFR Part 50, Section 50.59(a)(2).

11.2.13 Post-Accident Pressure Temperature Limits

Introduction

Item 2 of the IE Bulletin 79-05B addresses actions to be taken by reactor operators following automatic actuation of the High Pressure Injection (HPI) system due to low reactor coolant system pressure. Based upon IE Bulletin 79-05B, continued operation of HPI is based upon maintenance of a 50°F subcooling margin in the reactor coolant system; however, "The degree of subcooling beyond 50 degrees F and the length of time HPI is in operation shall be limited by the pressure temperature considerations for vessel integrity." The purpose of this section is to describe draft Technical Specifications which incorporate pressure temperature limits to be utilized following HPI initiation.

Discussion

During certain Loss of Coolant Accident (LOCA) conditions generated by small breaks in the Reactor Coolant System (RCS), the HPI system is relied upon to effect cooling of the core. Cold HPI water is injected into a relatively high pressure, stagnant, RCS loop which can potentially cause a reduction in the margin to brittle failure of the reactor pressure vessel.

The high differential temperature between the RCS and HPI water has two significant effects:

(1) As HPI water flows into the reactor vessel, it will cool the metal. This consequently reduces the fracture resistance of the metal.

(2) Thermal stresses will be developed because of rapid cooling of the inside metal surface (thermal shock). This stress will be superimposed on the existing residual stresses and those stresses that are generated by internal pressure. This combined stress field is significant, especially at a time when the fracture toughness of the reactor pressure vessel has decreased.

Analyses were made (Reference 1,2) of the consequences of HPI actuation on the TMI-1 reactor pressure vessel considering effects of:

(1) Neutron fluence on fracture toughness of the material.

(2) Resulting stress field.

(3) Linear elastic fracture mechanics techniques that are outlined in Appendix A to ASME Code Section XI and Appendix G to ASME Code, Section III.

The results of the analyses are incorporated in a pressure-temperature curve presented in Figure 3.1-1a of draft Technical Specification 3.1.2.1, "Pressurization Heatup and Cooldown Limitation." Draft Technical Specification 3.1.2.1 requires Figure 3.1-1a to be used following HPI initiation and until HPI is secured and the reactor is in cold shutdown. Existing pressure-temperature limits for heatup/cooldown and in-service leak and hydrostatic testing, which apply to normal operation and testing conditions, would not be effected by the additional limitations. The additional pressure-temperature limitations only apply to transient and accident conditions which result in HPI initiation.

Conclusion

The use of the post accident pressure-temperature limits presented in Figure 3.1-1a of draft Technical Specification 3.1.2.1 will assure that an appropriate margin to reactor vessel brittle failure will be maintained following accidents and transients resulting in HPI initiation.

References

(1) B&W Evaluation of RV Brittle Failure Due to Injection of Cold HPI Water During Small LOCA Events. June 13, 1979.

(2) New B&W study to be finalized 1st week of March, 1980.

Draft Technical

Specification Corresponding

to Section 11.2.7

3.4 DECAY HEAT REMOVAL - TURBINE CYCLE

Applicability

Applies to the operating status of equipment that functions to remove decay heat, utilizing the secondary side of the steam generators.

Objective

To define the conditions necessary to assure immediate availability of the auxiliary feedwater system and main steam safety valves.

Specification

3.4.1 With the reactor coolant system temperature greater than 250°F, three independent steam generator emergency feedwater pumps and associated flow paths* shall be OPERABLE with:

a. Two emergency feedwater pumps, each capable of being powered from an OPERABLE emergency bus, and

b. One emergency feedwater pump capable of being powered from an OPERABLE steam supply system. With one emergency feedwater pump or flow path* inoperable, restore the inoperable pump or flow path to OPERABLE status within 48 hours or be in COLD SHUTDOWN within the next 12 hours. With more than one emergency feedwater pump or flow path* inoperable, restore the inoperable emergency feedwater pumps or flow paths* to operable status or be subcritical within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in COLD SHUTDOWN within the following 6 hours.

c. Four of six turbine bypass valves are OPERABLE.

3.4.2 The condensate storage tanks (CSTS) shall be OPERABLE with a minimum of 150,000 gallons of condensate available in each CST. With a CST inoperable, restore the CST to operability within 4 hours or be in at least HOT STANDBY within the next 6 hours, at least HOT SHUTDOWN within the next 6 hours, and COLD SHUTDOWN within the next 24 hours.

*For the purpose of this requirement, an OPERABLE flow path shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator.

3.4.3 With the reactor coolant system temperature greater than 250°F, all eighteen (18) main steam safety valves shall be operable or, if any are not operable, the maximum overpower trip setpoint (see Table 2.3-1) shall be reset as follows:

Maximum Number of Safety Valves Disabled on Any Steam Generator	Maximum Overpower Trip Setpoint (% of Rated Power)
1	92.4
2	79.4
3	66.3

With more than 3 main steam safety valves inoperable, restore at least fifteen (15) main steam safety valves to operable status within 4 hours or be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

Bases

A reactor shutdown following power operation requires removal of core decay heat. Normal decay heat removal is by the steam generators with the steam dump to the condenser when system temperature is above 250°F and by the decay heat removal system below 250°F. Core decay heat can be continuously dissipated up to 15 percent of full power via the steam bypass to thee condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by the main feedwater system.

The main steam safety valves will be able to relieve to atmosphere the total steam flow if necessary. If main steam safety valves are inoperable, the power level must be reduced, as stated in Technical Specification 3.4.3, such that the remaining safety valves can accommodate the decay heat.

In the unlikely event of complete loss of off-site electrical power to the station, decay heat removal is by either the steam-driven emergency feedwater pump, or two half-sized motor-driven pumps. Steam discharge is to the atmosphere via the main steam safety valves and controlled atmospheric relief valves, and in the case of the turbine driven pump, from the turbine exhaust.(1)

Both motor-driven pumps are required initially to remove decay heat with one eventually sufficing. The minimum amount of water in the condensate storage tanks, contained in Technical Specification 3.4.2, will allow cooldown to 250°F with steam being discharged to the atmosphere. After cooling to 250°F, the decay heat removal system is used to achieve further cooling.

An unlimited emergency feedwater supply is available from the river via either of the two motor-driven reactor building emergency cooling water pumps for an indefinite period of time.

The requirements of Technical Specification 3.4.1 assure that before the reactor is heated to above 250°F, adequate auxiliary feedwater capacity is available. One turbine driven pump full capacity (920 gpm) and the two half-capacity motor-driven pumps (460 gpm, each) are specified. However, only one half-capacity motor-driven pump is necessary to supply auxiliary feedwater flow to the steam generators in the onset of a small break loss-of-coolant accident (Reference 2).

The requirements of Technical Specification 3.4.1 assure that at least 920 gpm is available at all times to both steam generators giving redundant capacity except for a limited time of 72 hours to allow for component maintenance. Further degradation of the emergency feedwater system requires the reactor to be subcritical within 1 hour.

The feedwater line break accident performed for TMI-2 (Reference 3) shows satisfaction of core thermal power limits and reactor coolant system pressure limits assuming full auxiliary feedwater flow within 40 seconds. The Technical Specification 3.4.1 provides assurance that this flow will be available with automatic initiation following loss of both main feedwater pumps.

REFERENCES

(1) FSAR, Section 10.2.1.3

(2) "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant," Volume I and II, Babcock and Wilcox, May 7, 1979.

(3) Three Mile Island Nuclear Station - Unit 2, Final Safety Analysis Report, USNRC Docket No. 50-320.

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
38.	Steam Generator Water Level	W	NA	R		
39.	Turbine Overspeed Trip	NA	R	NA		
40.	Sodium Thiosulfate Tank Level Indicator	NA	NA	R		
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	R		
42.	Diesel Generator Protective Relaying	NA	N	R		
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)	NA	M(1)	R	(1)	Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1)	When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F.
45.	Loss of Feedwater Trip	S(1)	M(1)	R	(1)	When reactor > 10% power.
46.	Turbine Trip/Reactor Trip	S(1)	M(1)	R	(1)	When reactor > 20% power.
47.	Pressurizer Code Safety Valve and Electromatic Relief Valve delta P/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.
48.	Pressurizer Electromatic Relief Valve - acoustic/flow	S(1)	R	R	(1)	When T_{AVG} is greater than 200°F.
49.	Saturation Margin Meter	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.
50.	Emergency Feedwater Flow Instrumentation	NA	M(1)	R	(1)	Emergency Feedwater is not normally in operation.

S - Each Shift
D - Daily
W - Weekly
M - Monthly
T/W - Twice per week
B/M - Every 2 months
Q - Quarterly
P - Prior to each startup
R - Each Refueling Period
NA - Not Applicable
B/W - Every two weeks

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each refueling shutdown
2. Control Rod Movement	Movement of each rod	Every two weeks, when reactor is critical
3. Pressurizer Safety Valves	Setpoint*	50% each refueling period
4. Main Steam Safety Valves	Setpoint	25% each refueling period
5. Refueling System Interlocks	Functional	Start of each refueling period
6. Main Steam Isolation Valves	(See Section 4.8)	
7. Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8. Charcoal and high efficiency filters for Control Room, and RB Purge Filters	DOP test on HEPA filters, freon test on charcoal filter units	Each refueling period and at any time work on filters could alter their integrity
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 Ft. 6 in.)	(a) Silt Accumulation-Visual inspection of Intake Pump House Floor (b) Silt Accumulation Measurement of Pump House Flow	Each refueling period Quarterly
11. Pressurizer electromatic relief valve	Setpoint	Each refueling period
12. Back-up instrument air supply system	Functional	Each refueling period

*The setupoint of the pressurizer code safety valves shall be in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968.

4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 EMERGENCY LOADING SEQUENCE

Applicability

Applies to periodic testing requirements for safety actuation systems.

Objective

To verify that the Emergency loading sequence and automatic power transfer is operable.

Specifications

4.5.1.1 Sequence and Power Transfer Test

a. During each refueling interval, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable.

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

- M. U. Pump
- D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
- R. B. Cooling Pump
- R. B. Ventilators
- D. H. Closed Cycle Cooling Pump
- N. S. Closed Cycle Cooling Pump
- D. H. River Cooling Pump
- N. S. River Cooling Pump
- D. H. and N. S. Pump Area Cooling Fan
- Screen House Area Cooling Fan
- Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psi Pressure Test Signal.)
- Motor Driven Emergency Feedwater Pump.

4.5.1.2 Sequence Test

a. At intervals not to exceed 3 months, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power.

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

Bases

The Emergency loading sequence and automatic power transfer controls the operation of the pumps associated with the emergency core cooling system and Reactor Building cooling system. A successful test of the emergency loading sequence and automatic power transfer is a prerequisite to any system test of the emergency core cooling system or reactor building cooling system.

References

(1) FSAR Section 7

(2) FSAR Section 1.4

4.9 EMERGENCY FEEDWATER SYSTEM PERIODIC TESTING

Applicability

Applies to the perodic testing of the turbine driven and two motor-driven emergency feedwater pumps, assocaited actuation signals, and valves.

Objective

To verify that the auxiliary feedwater system is capable of performing its design function.

Specification

4.9.1 TEST

4.9.1.1 Whenever the Reactor Coolant System temperature is greater than 250°F, the emergency feedwater pumps shall be tested in the recirculation mode in accordance with the requirements and acceptance criteria of ASME Section XI Article IWP-3000. The test frequency shall be at least every 31 days ±7 days of plant operation at Reactor Coolant Temperature above 250°F.

4.9.1.2 During testing of the emergency feedwater system when the reactor is in STARTUP or POWER OPERATION, if one steam generator flow path is made inoperable, a dedicated qualified individual who is in communication with the control room shall be continuously stationed at the EFW manual valves. On instruction from the control room, the individual shall realign the valves from the test mode to their operational alignment.

4.9.1.3 At least once per 31 days each valve listed in Table 4.9-1 shall be verified to be in the status specified in Table 4.9-1.

4.9.1.4 At least once per 18 months, during shutdown, verify that:

(a) each emergency feedwater pump starting logic actuates upon receipt of an auxiliary feedwater test signal, and

(b) valves in the emergency feedwater flow paths* actuate to their correct position on an emergency feedwater test signal and that the manual control valve station functions properly.

4.9.1.5 On a quarterly basis, the valves which are a part of the emergency feed system discharge (EFV-30A and 30B) will be checked for proper operation by cycling the valve over its full stroke.

4.9.1.6 Prior to start-up, following a cold shutdown of longer than 30 days' duration, conduct a test to demonstrate that the motor driven emergency feed pumps can pump water from the CST to the steam generators.

*For the purpose of this requirement, an OPERABLE flow shall mean an unobstructed path from the water source to the pump and from the pump to a steam generator.

4.9.2 ACCEPTANCE CRITERIA

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly.

Bases

The 31 day testing frequency will be sufficient to verify that the turbine driven and two motor-driven emergency feedwater pumps are operable and that the associated valves are in the correct alignment. ASME Section XI Article IWP-3000 specifies requirements and acceptance standards for the testing of nuclear safety related pumps. Compliance with the normal acceptance criteria of IWP-3000 assures that the emergency feedwater pumps are operating as expected. The test frequency of 31 days (nominal) has been demonstrated by the B&W Emergency Feedwater Reliability Study to assure an appropriate level of reliability. If testing under Article IWP-3000 indicates that the flow and/or pump head for a particular pump is not within the normal acceptance standard, Article IWP-3000 requires that an evaluation of the pump performance shall be completed within 96 hours or the pump declared inoperable. For the case of the emergency feedwater system, the system shall be considered operable if under the worst case single pump failure, a minimum of 500 gpm of emergency feedwater can be delivered when steam generator pressure is 1050 psig and one steam generation is isolated. A flow of 500 gpm, at 1050 psig head, ensures that sufficient emergency feedwater, demonstrated to be acceptable for plant cooling requirements under transient and accident conditions, can be delivered to either steam generator flow path. The 18 month surveillance requirements ensure that the overall emergency feedwater system functional capability is maintained comparable to the original design standards.

Table 4.9-1

Status of EFW Valves

Valve No.	Status
CO-V-10A	Open
CO-V-10B	Open
EF-V-1A	Open
EF-V-1B	Open
EF-V-2A	Open
EF-V-2B	Open
MSV-2A	Open
MSV-2B	Open
EF-V4	Locked Closed*
EF-V5	Locked Closed*
EF-V6	Locked Open*
EF-V10A	Locked Open*
EF-V10B	Locked Open*
EF-V-16A	Locked Open*
EF-V-16B	Locked Open*
EF-V-20A	Locked Open*
EF-V-20B	Locked Open*
EF-V-22	Locked Open*

*Locked Valves, if Maintained in a status other than indicated, shall be under the administrative control of a dedicated qualified, individual who is in communication with the control room. The individual shall be continuously stationed at the EFW manual valves and, on instruction from the control room, shall realign the valves from the test mode to their operational alignment.

Draft Technical

Specifications Corresponding

to Section 11.2.8

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
10.	Flux-Reactor Coolant Flow Comparator	S	M	R	
11.	Reactor Coolant Pressure Temperature Comparator	S	M	R	
12.	Pump Flux Comparator	S	M	R	
13.	High Reactor Building Pressure Channel (1)	S	M	R	(1) Includes post-accident monitoring instrumentation (a) -5 psig to three-times design pressure
14.	High Pressure Injection Logic Channel	NA	Q	NA	
15.	High Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or $T_{av.}$ is greater than 200°F
16.	Low Pressure Injection Logic Channel	NA	Q	NA	
17.	Low Pressure Injection Analog Channels				
	a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Taves is greater than 200°F
18.	Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
28.	Radiation Monitoring Systems (1)	W(2)	M	Q(3)	(1) Includes post accident monitoring instrumentation (a) High Range-Containment (b) Main Steam Lines (c) Auxiliary and Fuel Handling Building Exhaust (d) Condenser Off Gas Exhaust (2) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that the monitor is functioning. (3) Except area gamma radiation monitors RM-G6, RM-G7, and RM-G8 which are located in high radiation areas of the Reactor Building. These monitors will be calibrated quarterly or at the next scheduled reactor shutdown following the quarter in which calibration would normally be due, if a shutdown during the quarter does not occur.
29.	High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
30.	Borated Water Storage Tank Level Indicator	W	NA	R		
31.	Boric Acid Mix Tank					
	a. Level Channel	NA	NA	R		
	b. Temperature Channel	M	NA	R		
32.	Reclaimed Boric Acid Storage Tank					
	a. Level Channel	NA	NA	R		
	b. Temperature Channel	M	NA	R		
33.	Containment Temperature	NA	NA	R		
34.	Incore Neutron Detectors	M(1)	NA	NA	(1)	Check functioning; including functioning of computer read-out or recorder readout when reactor power is greater than 15%
35.	Emergency Plant Radiation Instruments	M(1)	NA	R	(1)	Battery check
36.	Strong Motion Accelerometer	Q(1)	NA	Q	(1)	Battery check
37.	Reactor Building Sump and Containment Level (1)	NA	NA	R	(1)	Includes post-accident monitoring Instrumentation (a) Narrow Range (sump) (b) Wide Range (Containment)

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE		REMARKS
51.	Reactor Coolant Pump Trip Trip	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.
52.	Reactor Building, Hydrogen Concentration	S(1)	M(1)	R	(1)	When T_{AVG} is greater than 200°F.

Draft Technical

Specifications Corresponding

to Section 11.2.13

3.1.2 PRESSURIZATION HEATUP AND COOLDOWN LIMITATIONS

Applicabilty

Applies to pressurization, heatup and cooldown of the reactor coolant system.

Objective

To assure that temperature and pressure changes in the reactor coolant system do not cause cyclic loads in excess of design for reactor coolant system components.

Specification

3.1.2.1 For operations until five effective full power years, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1, Figure 3.1-1a, and Figure 3.1-2 and are as follows:

Heatup/Cooldown

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-1.

Heatup/Cooldown Following HPI Initiation

Following HPI initiation, and until HPI is no longer needed, and the reactor is in COLD SHUTDOWN, allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-1a.

Inservice Leak and Hydrostatic Testing

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1-2. Heatup and cooldown rates shall not exceed those shown on Figure 3.1-2.

3.1.2.2 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.3 The pressurizer heatup and cooldown rates shall not exceed 100°F in any one hour. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.4 Prior to exceeding five effective full power years of operation, Figures 3.1-1 and 3.1-2 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. The highest

predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.5.

Based on the predicted RT NDT after five effective full power years of operation, the pressure-temperature limits of Figure 3.1-1 and 3.1-2 have been established in accordance with the requirements of 10 CFR 50, Appendix G. The methods and criteria employed to establish the operating pressure and temperature limits are as described in BAW-10046. The protection against nonductile failure is assumed by maintaining the coolant pressure below the upper limits of these pressure temperature limit curves.

The pressure limit lines on Figures 3.1-1 and 3.1-2 have been established considering the following:

a. A 25 psi error in measured pressure.

b. a 12°F error in measured temperature.

c. System presure is measured in either loop.

d. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

Following initiation of HPI, additional stresses are introduced as a result of the high differential temperature between the reactor coolant system and HPI water. Figure 3.1-1a (5,6) is for use following HPI initiation, and until HPI is secured and the reactor is in COLD SHUTDOWN.

The spray temperature difference restriction, based on a stress analysis of spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

References

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

(4) BAW-1439, Analysis of Capsule TMI-1E From Metropolitan Edison Company, Three Mile Island Nuclear Station - Unit #1, Reactor Vessel Materials Surveillance Program.

(5) B&W Evaluation of RV Brittle Failure Due to Injection of Cold HPI Water During Small LOCA Events. June 13, 1979.

(6) New B&W Study to be finalized 1st week of March, 1980.

TO BE SUPPLIED LATER

FIGURE 3.1-1a

REACTOR COOLANT SYSTEM HEAT-UP/COOLDOWN LIMITATIONS FOLLOWING HPI INITIATION (APPLICABLE TO 5 EFPY)

SUBJECT CONTENTS OF NRC QUESTIONS
SUPPLEMENT 1, PART 1

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
1	EFW	Auto Initiation
2	EFW	Manual Initiation
3	EFW	Diesel Generator Loading
4	EFW	Flow Indicators
5	EFW	Startup and Testing of Initiating Circuits
6	EFW	Flow Indicators
7	EFW	Testing of Manual Control
8	EFW	Flow Requirements
9	EFW	Failure Mode of Control Valves
10	EFW	System Evaluation
10a	EFW	Level Indication for CST
10b	EFW	Pump Endurance Test
10c	EFW	Flow Requirements
10d	EFW	Testing Requirements
10e	EFW	Water Supply
10f	EFW	Testing Requirements
10g	EFW	Flow to Depressurized Steam Generator
10h	EFW	Pump Protection
10i	EFW	Independence from AC Power
10j	EFW	Turbine Steam Line Break
10k	EFW	Auto Initiation
11	Pressurizer Heaters	Separation of Class I E Circuits
12	Power Assignments	Electrical Diagrams for New or Realigned Equipment

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
13	PORV and SV	Flow Indicators
14	Pressurizer Heaters	Adequacy of 126 kw
15	PORV and SV	Flow Indicators in Procedures
16	PORV and SV	Test Program
17	Inadequate Core Cooling	Use of Existing Instrumentation
18	Procedures	Use of Instrumentation Discussed in Question 17
19	In-Core Thermocouples	Minimum Operable
20	Subcooling Meter	Description
21	Containment Isolation	Reactor Coolant Pump Supporting Auxiliaries
22	Containment Isolation	Identification of Valves
23	Containment Isolation	Electrical Drawings
24	Containment Isolation	Reactor Coolant Pump Supporting Auxiliaries
25	Containment Isolation	Spurious Initiation
26	Containment Isolation	Automatic Reset
27	Containment Isolation	Diverse Signals
28	Containment Isolation	Pressure Setpoint
29	TMI Chronology	Internal Review
30	Changes to PORV and High Pressure Trip Setpoint	Internal Review

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
31	Reactor Trips	Provision for Safety Grade Circuits
32	Emergency Planning	Submittal of Procedures for NRC Review
33	Reactor Coolant Pump Trip	Provision for Safety Grade Circuits
34	Reactor Trip/ Turbine Trip	Supplying Data
35	Small Break LOCA	Analytic Assumptions
36	Small Break LOCA	HPI "Split"
36a	HPI	Core Makeup Capacity Following HPI Line Break
36b	HPI	Venturi Locations and Crossconnect
36c	HPI	Testing of Design Changes
37	HPI	Installation of Venturis
38	Technical Specifications	Submitted to NRC
39	Inadequate Core Cooling	Instrumentation, Procedures, and Training
40	Shift Supervisor	Requirements
41	Shift Foreman	Requirements
42	Shift Technical Engineer	Requirements, Duties, and Authority
43	Shift and Relief Turnover	Schedule for Submittal
44	Control Room Access, Onsite Technical Support Center, and Onsite Operational Support Center	Schedule for Submittal
45	Small Break LOCA	Procedures

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
46	TMI-2 Accident	Exam for Operators
47	OARP Audit Exam	Criteria for Passing Grade
48	Training	Information on Design/Procedure Changes
49	OARP	Integration of Objectives into Training Program
50	OARP	Use of Outside Organizations
51	Natural Circulation	Guidance to Operators
52	OARP	Use of Plant Parameters
53	Safety Related Valve Positions	Operational Review
54	EFW	Valve Check after Maintenance or Surveillance
55	Natural Circulation	Filling of OTSG
56	Engineered Safety Features	Terminating Operation
57	Notification of NRC	Guidance to Operators
58	EFW	Proper Valve Positioning
59	Containment Isolation	Guidance on Resetting
60	Shift and Relief Turnover	Status of Safety Systems
61	TMI-2 Accident	Lecture for Operators and Maintenance Personnel
62	Shift Staffing	Commitment for Two Licensed Operators
63	Automatic Switch Over to Recirculation	Description of ECCS Design

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
64	Reactor Building Spray	Description of Modification
65	Radiation Protection and Chemistry Supervisor	Membership in PORC
66	Radiation Protection and Chemistry Supervisor	Minimum Qualification
67	Radiation Protection Program	Detailed Description

QUESTION:

10. The NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island Unit 2 recently identified additional requirements for auxiliary feedwater systems. In addition to the requirements identified in this letter, other requirements which may be applicable to the subject facility are expected to be generated by the Bulletin and Orders Task Force. Such requirements are those resulting from our review of the loss-of-feedwater event and the small break loss-of-coolant accident. Our specific concerns include system reliability (other than auxiliary feedwater system), analyses, guidelines and procedures for operators, and operator training. The design and procedures of your facility should be evaluated against the following requirements to determine the degree of conformance. Provide the results of this evaluation and an associated schedule and commitment for implementation of required changes or actions described in Appendix A.

RESPONSE

The results of our evaluation of the requirements specified in questions 10a through 10k are following. The results of our evaluation of Appendix A to your October 26, 1979 letter is attached.

SUBJECT CONTENTS OF NRC QUESTIONS
SUPPLEMENT 1, PART 2

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
1	EFW	Additional Information on Questions 1 and 10k (Supplement 1, Part 1)
2	EFW	Additional Information on Question 9 (Supplement 1, Part 1)
3	EFW	Additional Information on Question 8 (Supplement 1, Part 1)
4	EFW	Additional Information on Question 4 (Supplement 1, Part 1)
5	EFW	Qualification for EFW Flow Measuring Devices
6	EFW	Control Room Annunciation of Automatic Starting
7	EFW	Additional Information on Question 10b (Supplement 1, Part 1)
8	EFW	Additional Information on Question 10e (Supplement 1, Part 1)
9	EFW	Additional Information on Question 10g (Supplement 1, Part 1)
10	EFW	Loading of Motor Driven EFW Pumps
11	EFW	Tech. Specs. on Flow Capacity
12	EFW	Passive Failure of Discharge Line
13	EFW	Additional Information on Question 10a (Supplement 1, Part 1)
14	EFW	Additional Information on Question 10 (Supplement 1, Part 1)
15	Reactor Trips	Evaluation of Parameters
16	Reactor Coolant Pump Trip	Additional Information on Question 33 (Supplement 1, Part 1)
17	Reactor Trip	Additional Information on Question 34 (Supplement 1, Part 1)

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
18	Pressurizer Heaters	Additional Information on Question 14 (Supplement 1, Part 1)
19	PORV and SV	Test Program
20	PORV and SV	Additional Information on Question 13 (Supplement 1, Part 1)
21	Safety Related Valve Positions	Additional Information on Question 53 (Supplement 1, Part 1)
22	Safety Related Valve Positions	Independent Verification
23	EFW	Additional Information on Question 58 (Supplement 1, Part 1)
24	Containment Isolation	Additional Information on Question 59 (Supplement 1, Part 1)
25	Shift Supervisor	Implementation of Shift Supervisor Program
26	Shift Technical Engineer	Accident Assessment Function
27	Shift Technical Engineer	Training
28	Shift and Relief Turnover	Schedule for Development
29	Shift and Relief Turnover	Evaluation of Effectiveness
30	Pressurizer Heater Changeover from Control Room, PORV and Block Valve Power	Design Details
31	EFW	Isolation of Manual Control from ICS
32	Containment Isolation	Electrical Drawings
33	EFW	Circuit Interaction

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
34	Power Assign-ments	Additional Information on Question 12 (Supplement 1, Part 1)
35	Reactor Trip	Basis for Bypass of Turbine/Feedwater Reactor Trips
36	PORV and SV	Backup Position Indication
37	PORV and SV	Electrical Drawings for Position Indication
38	ICS	Vulnerability to Loss of Power
39	Accident Analyses	Use of RETRAN
40	Airborne Activity	Definition of "Clear A-ea"
41	Ingestion LPZ	Definition
42	EPZ	Location of Schools and Hospitals
43	Hospitals	Distance from Site to Hospital
44	Emergency Planning	Relationship to Participating Authorities
45	Small Break LOCA	Procedures
46	Emergency Planning	General Emergencies, Precautionary Measures
47	Emergency Planning	Classification System Used by PEMA
48	Emergency Planning	Early Warning Methods and Protective Action within EPZ
49	Emergency Planning	State and Local Emergency Plans Cross Referencing with TMI Emergency Plan
50	Emergency Planning	Offsite ESC, Manning by State and Local Authorities
51	Emergency Planning	Information on Regulatory Guide 1.101

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
52	Fuel Handling Building Environmental Barrier	Design Details
53	Solid Radwaste System	Design Details
54	TMI-2 Decontamination	Independence from TMI-1 Systems
55	Natural Circulation	System Response to Filling of the OTSG
56	Station Discharge	Location of Monitors
57	Plant Ventilation System	Leak-tight Testing
58	Met-Ed Organization	Duties of the Senior Vice President
59	Met-Ed Organization	Duty Station for the Vice President Nuclear Operations
60	TMI-1 Site Organization	Interaction with Unit 2
61	TMI-1 Site Organization	Rad. Chem and H.P. Functions
62	TMI Station Support Organization	Clarification
63	TMI Site Organization	Provide Additional Details
64	TMI Site Organization	Staffing Details
65	TMI-1 Site Organization	Clarification
66	TMI-1 Site Organization	Reporting Chains
67	TMI-1 Site Organization	Qualifications for Radiological Controls Manager

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
68	Radiation Protection Monitor	Role of the Auxiliary Operator
69	Fire Protection Organization	Role of the Tech. Analyst
70	TMI-1 Site Organization	Delegation of Authority During Contingencies
71	TMI-1 Site Organization	Personnel Resumes
72	TMI Station Support Organization	Duties of the Director, Technical Function
73	TMI Station Support Organization	Duties of the Director, TMI-2 Recovery
74	TMI Station Support Organization	Staffing Level
75	TMI Station Support Organization	Staffing Experience
76	TMI Station Support Organization	Resumes
77	TMI Station Support Organization	Responsibilities Other Than TMI-1
78	TMI Station Support Organization	Site Assignment
79	Emergency Planning	Regular Assignment of Emergency Organization Members
80	Emergency Planning	Extent of Mobilization of Emergency Organization
81	Manager of Support Services and Logistics	Conflict between Sections 4 and 5
82	Emergency Planning	Staffing Level for Onsite Emergency Organizations

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
83	Emergency Planning	Qualifications for Onsite Emergency Organizations
84	Emergency Planning	Interfacing Between Onsite Emergency Organizations
85	Emergency Planning	Mobilization Times for Onsite Emergency Organizations
86	Emergency Planning	Description of Duty Sections
87	Emergency Planning	Training for Members of Onsite Emergency Organizations
88	Emergency Planning	Details of Long-Term Recovery Organizations
89	Emergency Planning	Provisions for Keeping the Emergency Plan Up-to-Date
90	Single Failure Criteria	Definition for "Safety Grade"
91	Hydrogen Recombiner	Request for Additional Information
92	Natural Circulation	Additional Information on Question 17 (Supplement 1, Part 1)
93	Emergency Planning	Additional Information on Question 18 (Supplement 1, Part 1)
94	Subcooling Meter	Additional Information on Question 20 (Supplement 1, Part 1)
95	Inadequate Core Cooling	Additional Measures

SUBJECT CONTENTS OF NRC QUESTIONS
SUPPLEMENT 1, PART 3

QUESTION NO.	SYSTEM/TOPIC	SUBJECT
1	HPI	Additional Information on Questions 26 and 37 (Supplement 1, Part 1)
2	HPI	Additional Information on Question 36a (Supplement 1, Part 1)
3	HPI	Additional Information on Question 36c (Supplement 1, Part 1)
4	EFW	Single Failure Criteria for EFW Flow Control Valves
5	Post-Accident Sampling	Unacceptability of Letdown Monitor
6	Range of Radiation Monitors	Provide Additional Information
7	In-Plant Iodine Monitoring	Provide Additional Information
8	Solid Radwaste System	Storage Capacity
9	Solid Radwaste System	Process Control Program
10	Solid Radwaste System	System Capacity
11	Reactor Coolant Pump Trip	Design Requirements
12	ICS	Address Recommendations of BAW-1564
13	TMI-1/TMI-2 Separation	Systems Other Than Radwaste and Effluent Monitors
14	Radiation Protection Program	Recent Audit
1 (Enclosure 2)	PORV	Automatic PORV Block Valve Closure