2.1.1.2 Position Indication for PORV and Safety Valves

2.1.1.2.1 System Description

The purpose of this modification is to provide the Control Room Operator with information on the status of the pressurizer electromatic relief valve RC-RV2 and the pressurizer code safety valves RC-RV1A and RC-RV1B. Discharge flow will be measured by differential pressure transmitters connected across elbow taps downstream of each of the valves. In addition, the electromatic relief valve will be monitored by accelerometers mounted on the valve. These will detect flow if the valve opens. Alarms and indications will be provided in the control room to inform the operator if any of these valves are open.

2.1.1.2.2 Design Bases

A reliable and unambiguous control grade indication will be provided to the Control Room Operator if the pressurizer electromatic relief valve or code safety valves open. The monitoring system will remain functional in containment conditions associated with any transient for which valve status is required by the operator. Redundant and diverse means will be provided for monitoring the electromatic relief valve (RC-RV2). The monitoring systems will remain functional during a loss of off-site power. All equipment inside containment will be seismically mounted. The integrity of existing safety related systems will not be impaired by this modification.

2.1.1.2.3 System Design

All of the system components have been selected for reliable operation and, where applicable for operation under adverse conditions inside containment. The differential pressure transmitter model which has been selected has previously been qualified for operation in a post-LOCA environment, for ability to operate after a seismic event and withstand 2.2×10^8 rads. The components comprising the acoustic flow detector have been previously used by B&W in the Loose Parts Monitoring System. They have been seismically tested and have been tested under the B&W "Steam Line Break" and "Small Break LOCA" containment environments. They will withstand 10^8 rads. The monitoring systems will be supplied from on-site electrical power supplies. Diverse and redundant means will be used for monitoring of the electromatic relief valve. Both differential flow measurement and acoustic detectors will be provided.

The pressurizer safety and relief valve position indication features described above are designed to provide information which supplements information from existing instrumentation. Figure 7-17 of the TMI-1 FSAR presents the TMI-1 reactor coolant system instrumentation including diverse features to determine the status of the pressurizer relief and safety valves. These features include the following:

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- Temperature elements are located on the discharge piping of the pressurizer relief and safety valves. This instrumentation is provided with an alarm to indicate a "high" condition.

- The Reactor Coolant Drain Tank, which receives the discharge from the pressurizer safety and relief valves, has the following instrumentation:

  (1) Pressure indication and alarm to indicate a "high" condition.

  (2) Level indication and alarm to indicate a "high" condition.

  (3) Temperature indication and alarm to indicate a "high" condition.

The instrumentation features described above are individually capable of determining relief and safety valve status. Our response to NRC Question 36, Supplement 1, Part 2, describes how these instruments will be used by the reactor operators. The existing instrumentation, together with the proposed modifications, acoustic on the PORC, and differential pressure instrumentation, on all three valves) provides a highly reliable and diverse means of determining safety and relief valve status. In this regard, the accoustic and differential pressure instrumentation are most suitable for indicating gross valve position (open/closed); instrumentation associated with the reactor coolant drain tank and safe/relief valve discharge piping should be used to detect the lower levels of steam flow associated with a leaking valve.

Plant procedures direct the operator to observe the acoustic monitor (PORV), Reactor Coolant Drain Tank instrumentation, and the tailpipe thermocouples to confirm valve closure for transients and accidents which may open the PORV (or safety valves). Once closed after an opening cycle, plant procedures call for determining RCS leakage to verify that Technical Specification leakage limits are met.

The range of the elbow tap instruments was selected to be 400 inches. This range was selected since 20 inches of water was determined to correspond to the lowest significant flow, and 5% of full scale (eg. 20 inches for a full scale of 400 inches) corresponds to the lowest recommended setting for alarm reliability. In addition, due to the square root relationship of differential pressure to flow a smaller range such as 0-100 inches would not significantly improve the sensitivity of the system. The existing sensitivity between 10 and 20 inches of water is considered adequate in light of other backup instruments discussed above. The PORV will be test lifted once during the restart startup program to test the elbow tap instrument.

2.1.1.2.4 Design Evaluation

Elbow taps are widely used for flow measurement in fluid systems and a great deal of empirical data is available for calculating expected differential pressure across elbow taps for given flow

conditions. Calculations have been made, using conservative assumptions, to demonstrate that a satisfactory signal will be 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 tran mitter 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 t..e 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. Shock loadings were considered in the original design of the PORV and safety valve discharge piping. It was also considered for the design of the elbow tap instrument since water loop seals are maintained on the PORV and safety valves to prevent contact of steam during normal operation. When the safety/relief valve opens a slug of water is discharged to the tailpipes. The small size of the elbow tap instrument lines compared to the 4 inch (PORV) tailpipe results in only a small portion of the pressure wave (due to slug flow) from effecting the elbow tap instrument. Any effect is further dampened by condensing pots in the instlrument lines close to the tailpipe elbow and due to the relatively long length of instrument sensing lines. The differential pressure cell is also designed to withstand full system pressure of 2500 psi across the diaphram compared to an operational requirement of 400 inches of water with no loss of accuracy or damage.

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.

2.1.1.3 Emergency Power Supply Requirements for Pressurizer Heaters, PORV, Block Valve, and Pressurizer Level Indication

2.1.1.3.1 Pressurizer Heaters

2.1.1.3.1.1 System Description

The purpose of this modification is to provide redundant emergency power for the 126 KW of pressurizer heaters required to maintain natural circulation conditions in the event of a loss of offsite power. A manual transfer scheme will be installed to transfer the source of power for 126 KW of pressurizer heaters from the balance of plant (BOP) source to a "Red" engineered safeguards (ES) source. A similar manual transfer scheme will be installed to transfer the source of power for 126 KW of pressurizer heaters from the BOP source to a "Green" ES source. Each manual transfer scheme will have double isolation on each end of the transfer and have mechanical key interlocks to govern the order of the transfer procedure. Figure 2.1-4 is a schematic representation of these transfer schemes.

2.1.1.3.1.2 Design Basis

Babcock and Wilcox has recommended that at least 126 KW of pressurizer heaters be restored from an assured power source within two hours after a loss of offsite power. Separation and isolation of Class IE equipment and circuits from non-Class IE equipment and circuits will be in accordance with Regulatory Guide 1.75.

The 480 volt ES circuit breaker is the isolating device between Class IE and non-Class IE portions of the design. The Class IE portion of the design is that portion up to and including the 480 volt ES circuit breaker and protective elements. Unervoltage relays connected to the 480 volt ES bus will detect a fault that is of sufficient magnitude to endanger the ability of safety loads on the bus to start or run. The undervoltage relays will initiate tripping of the 480 volt ES circuit breaker feed to the pressurizer heaters and thereby remove any endangerment caused by that circuit.

While the remaining portion of the design is classified non-Class IE, separation will be maintained up to the pressurizer heater terminal box T-161 which is located on the secondary shield wall. The remaining portion of the design (i.e., terminal box T-161, pressurizer heater elements and interconnecting cable routing) will remain as it presently exists. The constraints imposed by the plant's original physical construction make this necessary. The relative closeness of the pressurizer heater elements and heater bundles does not permit further physical separation.

The pressurizer heaters are passive devices. The only common event to affect both redundant emergency pressurizer heater circuits could only occur in that area between terminal box T-161 and the pressurizer heater elements. For such an event, the protection on the 480 volt ES feed to the pressurizer heaters will be fully coordinated with the protection on the 480 volt ES bus main circuit breaker.

If both emergency circuits to the pressurizer heater are lost, reactor coolant pressure shall be sufficiently maintained through use of the make-up (HPI) pumps.

The double isolation in the form of the circuit breakers and removable elements (see Figure 2.1-4) along with the kirk key interlocks, preclude lining up the system with a 480 volt ES bus connected to the BOP bus or lining up the system with the "Red" 480 volt ES bus connected to the "Green" 480 volt ES bus.

B&W determined the number of pressurizer heaters by taking into account the following:

1. The loss through the pressurizer insulation was calculated. The service areas of the insulation was determined and an average heat flux for the outside service area was assumed to be 80 BTU/hr ft^2. This calculated to a heat loss of approximately 96,000 BTU/hr.

2. The loss through the uninsulated pressurizer areas around the horizontal heater bundles was calculated in the same manner as item 1 and resulted in an approximate heat loss of 50,000 BTU/hr.

3. B&W's experience has shown that the insulated heat losses account for less than half of the total losses. Therefore, a factor of 2.5 was applied to the sum of the accounted losses.

Thus, the total calculated heat loss from the system is 365,000 BTU/hr or 107KW. Due to the grouping at the pressurizer heaters, one bank of pressurizer heaters consisting of 126KW was recommended.

The time for establishing the heaters is determined by the amount of heat losses from the pressurizer and the initial water level in the pressurizer. Figure 2.1-2 shows the expected response after establishing natural circulation with no heat input from the heaters. From the Figure, two hours is sufficient time, for the heaters to operate, to insure natural circulation at hot standby after a loss of offsite power.

2.1.1.3.1.3 System Design

Existing spare Class IE 480 volt circuit breakers on the "Red" and "Green" ES systems will be utilized for the two transfer schemes. The removable element assemblies for each transfer scheme will consist of two cabinets and one tab-keyed, removable element. One cabinet will be located near and connected in series with the 480 V ES circuit breaker. The other cabinet will be located near and connected in series with the Pressurizer Heater Control Center circuit breaker. Class IE qualified power cable will connect the load sides of the disconnect switches as shown in Figure 2.1-4. Class IE qualified under-voltage relays will be installed on each ES bus. They will initiate tripping of the ES circuit breaker to the pressurizer heaters when the bus voltage drops below its set point. The set point will be chosen so that starters on the ES bus can pickup if energized and the voltage at the ES motors is not lower than their ratings allow. An Engineered Safeguards actuation signal shall trip but not lockout each ES circuit breaker to the pressurizer heaters. The remainder of the electrical power distribution system to the pressurizer heaters will remain as it presently exists.

2.1.1.3.1.4 System Operation

All pressurizer heaters will be powered from the BOP electrical power distribution system when offsite power is available. Upon a loss of offsite power, manual transfers will enable each of the onsite emergency diesel generators ("Red" and "Green") to provide power to 126 KW of pressurizer heaters when the diesel generators can accommodate that load. Procedures will call for tripping non-essential loads to accomplish this within the two-hour requirement. Mechanical key interlocks will dictate that the order of events in the transfer from BOP to ES power source will be as follows:

A. Opening the circuit breaker in the PHCC which will allow removal of key #1.

B. Key #1 will open the cabinet door of the disconnect switch located near the PHCC. The removable element will then be removed along with Key #2 and carried to the 480 V ES switchroom.

C. The removable element will be inserted into the appropriate cabinet. Key #2 will lock that cabinet door and allow removal of Key #3.

D. Key #3 will remove the inhibit feature from the 480V circuit breaker.

E. The circuit breaker control switch will then be operated to close the ES circuit breaker feed to the transferred pressurizer heaters when it has been established that bus loading and emergency D/G loading permit doing so.

When offsite power is restored, the reverse procedure will be used to transfer back to the BOP source.

2.1.1.3.1.5 Safety Evaluation

The manual transfer scheme design provides double Class IE separation of the ES system from the BOP system - the ES circuit breaker and the removable element. Taking into account the single failure criteria, faults on the BOP system will, at most, cause the loss of one 480 volt ES system. The transfer scheme design also precludes the connection of the "Green" ES system to the "Red" ES system.

2.1.1.3.1.6 Inservice Testing Requirements

The emergency diesel generator loading procedure will be rewritten to incorporate this modification. Therefore, these transfer schemes will be tested when the emergency diesel generators are tested.

2.1.1.3.2 Power Operated Relief Valve (PORV)

The present plant design is such that emergency diesel generator power will be supplied to the PORV (RC-V2) upon loss of offsite power. The PORV is powered from the 250 VDC Distribution Panel IC which in turn is powered from the "Red" and "Yellow" ES batteries and ES Battery Chargers 1A, 1C, and 1E.

2.1.1.3.3 Block Valve

The present plant design is such that emergency diesel generator power will be supplied to the block valve (RC-V3) upon loss of offsite power. The block valve is powered from the 480 V Engineered Safeguard Valve Conrol Center 1C.

2.1.1.3.4 Pressurizer Level Instrumentation

The present plant design is such that emergency diesel generator power will be supplied to the pressurizer level instrumentation power supplies (RC-1-LT1, RC-1-LT2, RC-1-LT3) upon loss of offsite power. The pressurizer level instrumentation power supplies are part of the ICS, NNI System, and are powered from the 120 volt ICS, NNI Power Distribution Panel ATA. That panel is, in turn, powered from the 120 volt Vital Distribution Panel VBA.

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.

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

Valve position shall be derived from stem operated limit switches or comparable means.

Annunciation (visual and audio), for the not closed position, will be provided in the main control room.

Control of valves for any one vent point will be independent of the control for valves for any other vent point.

Instrumentation will be provided to allow the operator to determine when venting is required. Also, instrumentation will be provided to indicate the presence of flow in the vent lines.

2.1.2.2.9 Reactor Vessel Head Vent

As described in section 2.1.2.2.5 the reactor vessel head vents do not contribute significantly to the mitigation of an accident. In addition the head vent lines will provide additional leak patterns and higher manrem exposure for takedown and installation during refueling outages. Also additional testing and maintenance will result in extended refueling time.

GPU does not recommend the installation of Reactor Vessel Head Vent System and concurs with Babcock & Wilcox's position that high point vents on the hot leg are adequate.

However, GPU has initiated the engineering, procurement and installation plans, consistent with Restart schedules, for the reactor vessel head vent System, as directed by NRC.

To reduce the potential for RCS leakages, a study is being conducted to determine feasibility and advantages/disadvantages of venting the Reactor Vessel head to the Pressurizer Steam Space (see figure 2.1-12). A conceptional design of the System will be submitted to NRC by July 15, 1980, and the detail design description will be provided by September 15, 1980.

It should be noted that the requirement to install reactor head Vent System is beyond the design basis requirements.

Valve position shall be derived from stem operated limit switches or comparable means.

Annunciation (visual and audio), for the not closed position, will be provided in the main control room.

Control of valves for any one vent point will be independent of the control for valves for any other vent point.

Instrumentation will be provided to allow the operator to determine when venting is required. Also, instrumentation will be provided to indicate the presence of flow in the vent lines.

2.1.2.3 Plant Shielding Review

2.1.2.3.1 Design Review of Pla Shielding

The Section 2.1.6b in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", requires Met-Ed 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 and also to identify locations of vital areas and equipments 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.

A report, TDR-121, "Plant Shielding Design Review Report", has been prepared and will be submitted separately to provide greater detail concerning the above requirement. The content of the report is based on the following guidelines:

a) The post-accident release of radioactivity in the evaluation should be equivalent to the source terms recommended in Regulatory Guide 1.4 and NUREG-0578.

b) The post-accident dose rates in areas requiring continuous occupancy should not exceed 15 mn/hour.

c) The post-accident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 Rem whole body or its equivalent.

The report has utilized the well-accepted cyllindrical source methodology for the calculation of dose rates at different vital aras. The doses to be received during ingress and egress and while performing a given operation in those areas have been determined by multiplying the dose rate by the travel time and the time assumed to perform the operation respectively.

As a result of the review, several areas requiring post-accident access have been identified as areas that would exceed the guidelines mentioned above. The report addressed mitigating measures to be taken for the problem areas so that the functions can be performed in those areas without exceeding the safety guidelines. The Table 2.1-6 depicts the required manual actions, areas that have been considered, radiation doses assuming no corrective action, and the corrective action under consideration. The report does not, however, identify the equipment requiring improved environmental qualification, which would be delineated in a separate report.

2.1.2.3.2 Design Basis

The source term to be used for shielding calculations shall be as follows:

Liquid Systems:

Noble Gas - 100% of core inventory
Halogens - 50% of core inventory
Others - 1% of core inventory

Containment Air:

Noble Gas - 100% of core inventory
Halogens - 25% of core inventory

The criteria for limiting general area radiation levels in order to assure personnel access to vital equipment will be as follows:

Areas requiring continuous occupancy - <15 mr/hr

Control Room
Operation Support Center (TMI-1 Health Physics)
Technical Support Center (Mod/comp room and cooldown from outside control room panel)

Areas requiring possible frequent access - <100 mrem/hr (Once or more per each 8 hour shift)

Radiochemistry Laboratory
H2 Recombiner Control Panel
Liquid Waste Disposal Panel

For all other areas, shielding will be provided as required to keep personnel exposures less than 10CFR20 and to maintain the integrated dose to vital equipment below that for which the equipment has been qualified. The integrated does to vital components and equipment will be determined using the calculated radiation levels and the required length of service of each component and piece of equipment post accident.

2.1.2.4 Post Accident Sampling Capability

2.1.2.4.1 System Description

Post accident analysis of reactor coolant samples and the containment atmosphere is recognized as a means to better define core damage and anticipate the need for remedial actions. The TMI-1 capabilities for post accident sampling will be modified as necessary to provide key sample results on an on-line basis and to provide backup confirmatory sampling capability within 2 hours of directing that a sample be taken.

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 boronometer will be installed. The conceptual design and schedule for installation will be forwarded to the NRC by August, 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 has been performed. This review has been based on the radiation levels experienced during the TMI-II accident. The results of this review are as follows.

The containment atmosphere sampling capabilities at TMI-I are basically similar to those which existed at TMI-II. A description of this sampling system is contained in Supplement 1 Part II of the TMI-I FSAR. No radiation exposure problems were experienced in obtaining atmosphere samples of the TMI-II atmosphere. As a result it is concluded that containment air sample can be obtained using the existing TMI-I system.

Originally, TMI-I and TMI-II utilized the same reactor coolant sampling facilities. These facilities are located on the 305 foot elevation of the TMI-I Control Building. Subsequent to the TMI-II accident, a separate facility has been provided for TMI-II sampling. However, based on the experience gained during the TMI-II accident, special procedures, long handle tooling and portable shielding has been developed for TMI-I. With these special procedures and equipment, a reactor coolant sample can be obtained within approximately one hour following determination of the need for the sample.

Once the reactor coolant sample is obtained it will be taken to the chemistry laboratory located adjacent to the sample room. TMI-II experience has demonstrated that the shielding afforded by the concrete shield walls separating these rooms provides sufficient shielding for personnel to prepare samples for spectrum analysis and/or chemical analysis. However, based on TMI-II experience, the Germanium Detector/Multi-Channel Analyzer has been relocated to the TMI-I Turbine Building to ensure background radiation levels will not interfere with analysis of the sample.

Based on the above, it is concluded that using the existing sampling facilities:

a) RCS sampling can be accomplished within approximately one hour and analyzed in an additional one hour following determination of the need for the sample.

b) 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.

c) The equipment and procedures available for sample analysis are of sufficient sensitivity and resolution to permit identification of gamma emitting isotopes in the reactor coolant system.

d) Chemical analysis capability will include the capability for boron and chloride analysis.

As further assurance of adequacy of the existing TMI-I sampling facilities a detailed shielding analysis utilizing the NUREG criteria will be performed. This analysis will be coupled with the dry run sampling times to verify radiation exposures will be less than those discussed above.

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 saturation margin has been lost.

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 desirable. The conceptual design described in this section is being submitted for NRC review and comment and will be implemented subject to concurrence of the NRC Staff.

2.1.2.5.2 Design Bases

See response to Question 11 of Supplement 1, Part 3.

2.1.2.5.3 System Design - To be provided later.

2.1.2.5.4 Design Evaluation - To be provided later.

2.1.2.6 Auxiliary Feedwater System

Auto start of the emergency-feedwater (EFW) System is being implemented in two phases: 1. Control Grade Auto Start - This is a non-safety related initiation as described in paragraph 2.1.1.7 and it is a short-term approach, 2. Safety Grade Auto Start - This will be a long-term modification where the initiation will meet the requirements for Class 1E system and the system is functionally described below.

1. The safety grade EFW auto start when implemented will automatically initiate the system on presence of the following conditions with or without the availability of the off-site power:

   - Loss of both normal feedwater pumps, or
   - Loss of all four reactor coolant pumps, or
   - Low differential pressure between the normal feedwater and main steam lines at either steam generator,

   The system initiation on low steam generator level will eventually be added. This will be done after the necessary analysis and engineering has been completed to insure that this signal will give a satisfactory actuation and will not interact with other plant functions. Loss of normal feedwater pumps is detected by differential pressure switches across each pump (two switches per pump, i.e., one switch per train).

   The model of differential pressure switches used for this application has been seismically tested. These switches have temperature limits of -60 to 200°F. Since they will be located in Turbine Building which is a non-seismic building, the switches will be tied into their respective EFW initiating circuits (Train A&B) through buffer devices and thus the switches will be treated as safety grade items to the extent possible. The buffer devices are relays similar to those described in the TMI-2 FSAR Section 7.3.2. This application is similar to the approved application described in that section.

2. All cables associated with the initiating logic will be qualified for Class 1E application and the initiations will be designed to meet single failure criteria. All circuits will meet the regulatory criteria for separation of Class 1E circuits.

3. The initiating logic will include hardware for the following purposes:

   - Latching mechanism to seal-in the actuation
   - Manual Override Capability
   - Testability of the initiating circuit

4. Indication will be provided in the control room to identify the source of the initiation.

5. Annunciation will be provided in the control room to alarm:

   - Auto start of the EFW system. This will be a common alarm for both the trains.

2.1.2.7 Increased Range of Radiation Monitors (2.1.8.b)

2.1.2.7.1 The existing Radiation Monitoring System provides in-line monitoring capability for effluents from:

a) Auxiliary and Fuel Handling Building (RM-A8)
b) Reactor Building Purge (RM-A9)
c) Condenser Off-Gas (RM-A5)

Discharge from Waste Gas Decay Tanks are monitored by RM-A7 prior to combination with other exhaust and after dilution by RM-A8. The Reactor Building Hydrogen Purge System discharge is monitored by the normal purge system monitor RM-A9.

The monitors, RM-A8 and RM-A9 are manufactured by Victoreen, Inc. and consist of:

a) A fixed filter particulate monitor; Beta scintillation detector; sensitivity approximately $1.5 \times 10^{10} \frac{cpm/min}{\mu Ci/cc}$ based on SR-90; full range 1×10^{6} cpm.
b) A Fixed Charcoal Filter Iodine Monitor; NaI detector with fixed window; Sensitivity approximately $1.3 \times 10^{9} \frac{cpm/min}{\mu Ci/cc}$; fill range 1×10^{6} cpm.
c) A gross gaseous monitor; Bata Scintillation detector; Sensitivity approximately $4 \times 10^{7} \frac{cpm}{\mu Ci/cc}$; full range 1×10^{6} cpm.
d) Air sampling pump with normal sample flow of approximately 1 cubic foot per minute.

Radiation monitor RM-A5 has only a gross gaseous monitor (c above) situated on the discharge of the condenser vacuum pumps, exhausting to the suction of the vacuum pumps. Flow through the monitor is regulated to maintain approximately 500 cc/min. All monitors have Control Room readout and recording.

2.1.2.7.2 Long Term Modifications

Increased range capabilities will be furnished for each of the effluent monitors described above (RM-A8, RM-A9, RM-A5) and the Main Steam lines. For the Long Term Modification additional monitoring ranges will be provided utilizing ionization chambers for the Reactor Building Purge Exhaust, the Condenser OFF-GAS Exhaust, the Main Steam Lines. The Auxiliary and Fuel Handling Building Exhaust will have extended monitoring ranges incorporating a G.M. device. The sensitivity of the individual units will be determined by standard volume source calculations.

The sensitivity will assure that release rate of:

5,600,000μCi/sec from Auxiliary & Fuel Handling Bulding.

2,300,000 μCi/sec from Reactor Building Purge.

1400 μCi/sec from Condenser Off-Gas based on maximum flow rates from each release path.

2500μCi/sec from a single steam generator can be detected.

The installation of each monitor will include evaluation of the position of the monitor relative to other potential radiation sources and shielding necessary to minimize the effect of sources other than sample lines on the response of the monitor and recording.

For each of the monitors described, the following applies:

Each will be powered from vital power, thereby providing redundancy in power supply.

Establishing sensitivities will be correlated to solid source calibrations. Procedures defining calibration method and frequency will be written to assure proper response of the instruments.

Emergency procedures will be written to the use of the radiation instrumentation in conjunction with flow information to determine release rate.

Emergency Plan implementing procedures describe the dissemination of information obtained from monitors.

Procedures and evaluations will be available for NRC review prior to 1 January 1981.

2.1.2.7.3 Short Term Modifications

Increased range capabilities will be furnished for the Reactor Building Purge Exhaust, the Condenser Off-Gas Exhaust, and the Main Steam Lines as a short term modification. This Short Term Modification will consist of G.M. Tubes or ionization chambers affixed to each of the effluent release paths described in 2.1.2.1.1 (only one detection system will be provided for each OTSG). Remote readout will be provided to areas, which are habitable during an accident. The Long Term Modification for the Auxiliary & Fuel Handling Building is projected to be completed by October, 1980. If a Long Term Modification is not available by start-up, a Short Term Modification utilizing a G.M. tube or ionization chamber will be incorporated. All devices will have necessary shielding if background effects are considered excessive.

The installation of each monitor will include evaluation of the position of the monitor relativve to other potential radiation sources and shielding necessary to minimize the effect of sources other than sample lines on the response of the monitor.

The sensitivity will assure that release rates of:

5,600,000 μCi/sec from Auxiliary & Fuel Handling Bldg.

2,300,000 μCi/sec from Reactor Building Purge.

1400 μCi/sec from Condenser Off-Gas based on maximum flow rates from each release path.

2500 μCi/sec from a single steam generator can be detected.

The range of these monitors is identical to the range capability of the long term modification.

For each of the monitors described, the following applies:

Each will be powered from normal power with battery backup.

Established sensitivities will be correlated to solid source calibration. Procedures defining calibration methods and frequency will be written to assure proper response of the instruments.

Emergency procedures will be written to the use of radiation instrumentation in conjunction with flow information to determine release rate.

Emergency Plan implementing procedures describe the dissemination of information obtained from the monitors.

Procedures and evaluations will be available for NRC review prior to restart of Unit I or 1 October 1980, whichever occurs first.

FIGURE 2.1-12

(To be provided later)

handle the waste generated from a single operating unit and TMI-1 operated successfully in the single unit mode in excess of three years during TMI-2 construction. The TMI-2 design utilized the conservation of the TMI-1 system to provide radwaste treatment capabilities for liquid radwaste generated from its operation. Therefore, all TMI-2 liquid radwaste had to be transferred to TMI-1 for treatment. Therefore isolation of the units will increase the TMI-1 liquid radwaste capability relative to that available during the pre-accident period. The Miscellaneous Waste Evaporation is the primary process equipment used to treat liquid radwaste. The evaporator has been used successfully in the past for a single unit waste treatment requirements. Because of interferences in TMI-1 caused by the storage of TMI-2 waste waters, the evaporation is not currently in use. At the time of startup of TMI-1 (or within the month previous to starting) the evaporation will be shown to be operational or replacement equipment shall be available to supplement evaporation operation.

Although, TMI-1 radwaste processing capabilities are currently limited by the presence of TMI-2 liquid radwaste stored in TMI-1 process vessels, this limitation will be completely removed by transferring and flushing all TMI-2 waste stored in TMI-1 to TMI-2 facilities prior to startup. TMI-2 miscellaneous liquid radwaste will be processed in systems installed (or to be installed) in accordance with the recovery effort. The systems include Epicor 2, the submerged demineralizer system and an evaporator/solidification system.

A complete description of the TMI-1 radwaste treatment system capability is provided in section 7.3.1.1.1 of this report.

7.2.4 Solid Waste Disposal

Prior to the accident, TMI-2 radwaste system was completely dependent on TMI-1 facilities for solid waste processing and disposal of evaporation concentrates, spent resin and compactible trash. Isolation of the two units will enhance the ability of TMI to process solid radwaste. TMI-2 had installed separate trash compacting equipment and will obtain the equipment to solidify liquid and wet solid waste as part of the recovery program. TMI-1 capabilities are not required for the TMI-2 recovery program. TMI-1 currently has an opeating trash compactor used exclusively for TMI-1 trash. Solidification facilities will be available through the use of a contractor for the short term and subsequently through the use of a permanently installed system discussed more thoroughly in Supplement 1, Part 2, response to Question 53 and Section 7.3.1.3. There are two options available for short term use for radwaste solidification. Both os these short term options, the portable Hittman solidification system and the portable Dow solidification system, are technically acceptable for use.

It has been decided to utilize the Hittman solidification system for short-term use on TMI-I. This system is currently in use at the Arkansas and Zion stations. The Hittman system will be installed and operational by July 1, 1980, pending approval of the Process Control Program and system Operating Procedures.

The uncertainties of present technology and changing regulatory requirements make the selection and installation of a permanent facility inappropriate prior to the restart of TMI-I. For these reasons the interim facility is projected to be in use for a period that will allow a refinement of technology to meet regulations in change at present; and engineering evaluation, procurement and installation of the permanent facility. This period is projected to be five years.

7.2.5 Sanitary Facility Drains

By design, sewage and sanitary drains from TMI-2 would join those from TMI-1 at the sewage pumping station and then be treated in a common sewage treatment plant. Operationally, the sewage from the individual unit currently remain separate 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 Baronometer 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 Environmental Protection Agency's NPDES program. These systems remove oils, greases, suspended solids and trace quantities of copper and iron from secondary plant waste water. (Figure 7-1 indicates the flow paths for specific waste water sources.) The systems are common to both operating units and will continue to be used. Because of the increased possibility of radioactive contamination from TMI-2, the transfer of waste water to the facilities from TMI-2 will be performed under strict procedural control. Additional procedural controls will be implemented when pumping from Unit 1 to the IWFS or ISTS should the TMI-1 secondary plant sumps become contaminated.

7.3 Supplemental Topics

7.3.1 Radwaste Capability

7.3.1.1 Liquid Radwaste Processing

7.3.1.1.1 General

Approximately 20,000 gallons of liquid radwaste related to the TMI-2 recovery is currently stored in TMI-1 process vessels. This liquid will be transferred to TMI-2 and the TMI-1 liquid radwaste processing system will be returned to a state of full operability prior to the TMI-1 startup. No system modifications are considered necessary to upgrade the system as a result of isolating the units.

Because the tank used for storage of TMI-2 waste blocked the main process path to the Miscellaneous Waste Evaporator, that process unit has been out of service. The Epicor I system is currently in operation to treat TMI-1 wastes in lieu of evaporation and to process TMI-2 waste with low concentrations of radioactive material. Following the physical separation of the units low activity TMI-2 waste will be processed using TMI-2 facilities. Epicor I will be retained until both evaporators in TMI-1 have been demonstrated to be operable to within 75% of their nominal design capacities. Descriptions of the TMI-1 liquid radwaste processing system and the Epicor I system are provided in Section 7.3.1.1.2 and 7.3.1.1.3 respectively.

7.3.1.1.2 Liquid Waste System and Equipment

TMI Unit 1 provides two strings of equipment for the processing of radioactive liquid wastes. One string of equipment is associated with the processing of reactor grade water including spent fuel pool

water and water on recycle through the decay heat system. The other string of equipment is associated with the processing of miscellaneous radioactive liquid wastes produced within the auxiliary and fuel handling buildings as a result of the processing of reactor and spent fuel pool liquids plus slightly radioactive liquids that occasionally occur in the secondary system as a result of steam generator tube leaks.

The reactor grade water processing string includes the following components:

One 780 ft.3 (5,500 gallons) Reactor Coolant Drain Tank: for suppression and collection of pressurizer relief and collection of process liquid from valve stems leak offs in the reactor coolant system and from the reactor coolant system and from the reactor coolant pump seals.

Two 11,000 ft.^{3} (80,000 gallons) Reactor Coolant Bleed Tanks: for the collection of letdown from the reactor coolant system and accept condensate from the secondary system. Either of these tanks can also be used for injection of boric acid solution into the reactor coolant system.

One 11,000 ft.^{3} (80,000 gallons) Reactor Coolant Bleed Tank: for injection of feed solution into the reactor coolant system. This tank can also accept letdown from the reactor coolant system.

Two 150 gpm precoat filters: for removing suspended and ionic solids from reactor grade water. One may be used for treating from miscellaneous wastes.

Two 70 gpm demineralizers (normally cation resion only): for removing ionic solids from reactor grade water.

One 12.5 gpm reactor coolant evaporator: for concentrating the reactor grade water for reuse and producing a purified distillate.

Two 920 ft.^{3} (6,500 gallons) reclaimed boric acid tanks: for storage of concentrated reclaimed reactor coolant grade water.

The miscellaneous waste string provides the following items of equipment:

One 3124 ft.^{3} (20,000 gallons) miscellaneous waste storage tank: for accumulating waste liquids from various sumps, vents and drains within the auxiliary and fuel handling buildings.

One 666 ft.^{3} (4,900 gallons) neutralizer feed tank: for accumulating and storing solutions to be neutralized.

One 194 ft.^{3} (1,000 gallons) neutralizer tank and mixer: for neutralizing solutions from the regeneration of the deborating demineralizer resins and adjusting the pH or adding antifoam agent to miscellaneous and laundry wastes prior to their evaporation.

One 779 ft.^{3} (5,500 gallons) neutralized waste storage tank: for accumulating neutralized wastes prior to their evaporation.

One 1,120 ft. ^{3} (8,000 gallons) spent resin storage tank: for the accumulation and storage of radioactive spent resin generated throughout the unit.

One 590 ft. 3 (4,000 gallons) used filter precoat storage tank: for the accumulation and storage of used precoat material.

B. The Auxiliary and Fuel Handling Building Ventilation Systems and the Reactor Building Purge System

The auxiliary and fuel handling building ventilation systems and the reactor building purge systems serve primarily to distribute to and exhaust ventilation air from the areas indicated.

7.3.1.3 Solid Waste System

7.3.1.3.1 General:

Five general types of waste are produced, processed and shipped from the TMI-1 as solid radioactive waste. These wastes are:

a. Concentrated Liquid Waste (evaporator bottoms)
b. Used Precoat (spent powdered resin)
c. Spent Resin (bead type)
d. Dry Compactible Trash
e. Dry Non-Compactible Trash

Dry trash is shipped offsite directly (i.e., without solidification) following compaction (to reduce the volume) where possible. TMI-1 has a trash compactor for use with 55 gallon drums dedicated for the exclusive use of the unit. Trash is segregated during collection to insure that TMI-1 and TMI-2 wastes remain separate. Appropriate packaging of dry trash is performed in accordance with applicable shipping and disposal regulations.

Concentrated liquid waste, used precoat, and spent resin will be solidified prior to being shipped offsite for disposal where required by the TMI-1 Technical Specifications and applicable regulations. Permanently installed plant equipment does not current exist to solidify radwaste.

A two part program has been initiated to solidify these wastes. For the short term, until the permanent system is available, TMI-1 will use contractor supplied mobile solidification systems. See also discussion in Section 7.2.4.

7.3.1.3.2 Mobile Solidification Systems

Two specific mobile solidification systems are being considered for use in TMI-1. Neither of the systems considered use urea formaldehyde (UF) as a solidification agent and neither system is subject to the free water problems associated with UF. The systems considered are the Hittman Nuclear and Development Corporation cement in-cask system and a demonstration system from Dow Chemical. A Process Control Program (PCP), approved by the USNRC is required to be available prior to the operation of any radwaste solidification system. Systems other than the two specifically noted may be considered depending on their capability, availability, and approval of an applicable PCP.