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

April 16, 1979

MEMORANDUM FOR: R. J. Mattson

FROM: NRR Technical Review Group

SUBJECT: UPDATED STATUS OF PROPOSALS FOR ACHIEVING COLD SHUTDOWN FOR TMI-2

The NRC Technical Review Group has been interfacing with 1) General Public Utilities and Burns and Roe to review and assess the feasibility of achieving cold shutdown by going water solid on the steam generators, 2) Westinghouse to review and assess the feasibility of upgrading the existing decay heat removal systems and building a new (third train) decay heat removal system, 3) Burns and Roe for spent fuel pool modifications and 4) General Public Utilities and Burns and Roe on Primary Makeup and Pressure Control.

The attached enclosures identified below provide an update of the status of these proposals.

- Enclosure 1 - Cold Shutdown of TMI-2 Using Water Solid Secondary Steam Generators.
- Enclosure 2 - Decay Heat Removal System
- Enclosure 3 - Comments on Possible OTSG Design Limitations Relating to Water Solid Operation
- Enclosure 4 - Assessment of Radiological Consequences of Steam Generator B in Short Term Mode
- Enclosure 5 - Summary of B&W Submittal of April 10, 1979 Regarding Natural Circulation

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Enclosure 6 - Spent Fuel Pit Modification

Enclosure 7 - Memo J. Gilray to T. Novak, "QA Practices Associated with TMI-2 Modifications" (without enclosures)

Enclosure 8 - Primary Makeup and Pressure Control

T. Novak for
NRC TECHNICAL REVIEW GROUP

Attachments:
Enclosures (8)

cc: H. Denton
V. Stello
NRC Technical Review Group (8)

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COLD SHUTDOWN OF TMI-2 USING WATER SOLID SECONDARY IN STEAM GENERATORS

1. Short Term Steam Generator "B" (Approach 1)

   A. Design Concept

   This concept will utilize a balance of plant cooling water system to effect cooling of steam generator "B" during water solid operation of the secondary side of the steam generator and ultimately under natural circulation of the primary water.

   Water will be circulated by the secondary services cooling water pumps through the main feedwater line and into the steam generator. It will exit the steam generator through the main steam line and will flow to the secondary service cooling water heat exchangers where it will be cooled and returned to the steam generator by the secondary service cooling water pumps. The secondary services cooling water heat exchanger will be cooled by the nuclear services river water system. Refer to Figure 2 for a schematic of this flowpath.

   B. Modification

   This scheme will involve installation of a jumper pipe between the discharge of the secondary services cooling water pumps and the feedwater line just downstream of the feedwater control valve. This pipe routing will be between 500 feet to 600 feet long located in the turbine building. A jumper pipe will also be installed between the main steam turbine bypass line and the supply line to the secondary services cooling water heat exchangers. This pipe routing will be approximately 20 feet long also located in the turbine building. These jumper pipes will include connections for the long term scheme. Provisions will also be made for demineralization of a portion of the flowrate. These piping modifications will be run along the floor and attached to railroad ties and existing steel beams in the turbine building where feasible. These modifications establish the flowpath for the secondary side of steam generator "B".

Additional jumper pipes will be installed between the nuclear services river water system and secondary services river water system to cool the secondary services cooling water heat exchanger. The safety classification of the nuclear services river water system will be maintained. This water flows to the mechanical draft cooling tower basin. This modification establishes the flowpath for cooling the secondary side of the secondary services cooling water heat exchanger. All piping connections will be welded. These modifications are expected to be completed in early May.

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During the meeting with Burns & Roe on April 13, 1979, several shortcomings of the short term steam generator B (Approach #1) modification were brought out. These were:

1. The secondary services cooling water heat exchangers should be limited to 150°F inlet water from the steam generators to prevent damage to the tube sheets. The heat exchanger manufacturer would expect that 200°F water would be acceptable for a short period of time, however, the tubes are not welded into the tube sheets and the higher than design inlet temperature could cause separation of the tubes from the tube sheet resulting in leakage. While there should be adequate recirculation in the system to prevent this occurrence, provisions will have to be incorporated into the design to limit this temperature.

2. Since the tubes are not welded into the tube sheets, there will always be some leakage between the primary and secondary side of the secondary services cooling water heat exchangers.

3. The pressure is higher on the steam generator side of the secondary services cooling water heat exchangers, therefore leakage is always out. This could result in releasing radioactive coolant into the mechanical, draft cooling towers basin by way of the nuclear services river water system if steam generator B leaks. Blowdown from the cooling tower would carry potentially radioactive water to the river.

Based on the shortcomings of the steam generator B short term Approach #1 modifications identified and the relatively short difference in time to fabricate Approach #2, it appears that it may be possible to go directly to a more leak tight modification with little compromise in completion time.

2. Short Term Steam Generator "B" (Approach 2)

A. Design Concept

This approach will utilize a new high pressure loop to effect cooling of the secondary side of steam generator "B" during water solid operation and ultimately under natural circulation of the primary water. The advantages of this system would be 1) potential leakage would either be into the primary system instead of out or less than expected in Approach 1 and 2) high reliability for long term operation. If this approach is used the long term modifications to steam generator B described in section 3 would probably be eliminated. Whether approach 1 or 2 for short term steam generator B modifications is used is dependent

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on the schedule for completion of the modifications. If the time difference is small, which is expected, then approach 2 is obviously preferrable.

Water will be circulated through a new heat exchanger and pump and into the secondary side of steam generator B in a closed loop to remove heat from the steam generator. The secondary side of this new heat exchanger will be cooled by water from other the nuclear services river water or secondary services river water systems. Refer to Figure 3 for a schematic of this flowpath.

B. Modification

This scheme will involve installation of a new train consisting of a pump, heat exchanger and piping. The loop will be connected to the main steam bypass line and the feedwater line just downstream of the feedwater control valve. This pipe routing will be between 550 feet to 650 feet long located in the turbine building. Provisions will be made for system surge and expansion and for demineralization of a portion of the flowrate. The piping modification will probably be run along the floor and attached to railroad ties and existing steel beams in the turbine building in a manner similar to that for Short Term S.G. "B" Approach 1. This modification establishes the flowpath for the secondary side of steam generator "B".

Jumper pipes will be installed between the nuclear services river water system or secondary services river water system to cool the secondary side of the new heat exchanger. The safety classification of the nuclear services river water system will be maintained if it is utilized in this scheme. Nuclear services river water flows to the mechanical draft cooling tower basin, and secondary services river water is discharged directly to the river. This modification establishes the flowpath for cooling the secondary side of the new heat exchanger. All piping connections will be welded. These modifications are expected to be completed in early May.

3. Short Term Steam Generator "A"

A. Design Concept

This concept will utilize the normal flow path through steam generator "A" during water solid operation of the secondary side of the steam generator and ultimately under natural circulation of the primary water.

Water will be circulated by the condensate pump through the condensate booster pump, feedwater pump and main feedwater line into the steam generator. It will exit the steam generator through the main steam line and will flow to the condenser. In order to achieve adequate heat transfer in the condenser, the water will enter through the makeup line. This line consists of a spray header which will spray the water over the condenser tubes thereby utilizing the majority of the tube surface area for cooling. Normal water level will be maintained in the condenser hotwell. The water will be returned by the condensate pump from the condenser hotwell to the steam generator. Normal circulating water will cool the condenser. Refer to Figure 1 for a schematic of this flowpath.

B. Modification

This scheme will involve installation of only one jumper pipe between the main steam turbine bypass line and the condenser makeup line. A separate independent cooler will be provided for the condensate pump motor bearing oil and the impellers of the condensate booster pump and feedwater pump will either be removed or blocked. Other than the above modifications, the flowpath for the secondary side of steam generator "A" is the same as during normal plant operation. All piping connections will be welded.

These modifications are expected to take 2 days following isolation and cooldown of Steam Generator A.

4. Long Term Steam Generator A and B Modifications

A. Design Concept

The long term steam generator A and B modifications have not been designed, however, the concept entails installing a high pressure loop with a heat exchanger and pump between the turbine by-pass line and the feedwater inlet. It should be noted that Approach #2 to short term Steam Generator #B is very similar to the long term concept. The advantages of this system would be:
1) potential leakage through steam generator tubes would be into the primary system instead of out and
2) high reliability for long term operation. Provisions are being incorporated into the short term modifications to tie in the long term when ready. Refer to Figure 4 for a schematic of this flowpath.

B. Modifications

The heat exchanger, pump and piping would be connected to the main steam turbine bypass line and feedwater line for

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each steam generator. This will provide the flowpath for the secondary side of the steam generator.

Additional connections will be made to the nuclear services river water system to provide cooling to the new heat exchangers. This water will flow to the mechanical draft cooling tower basin. All piping connections will be welded.

This modification is scheduled to be completed in 45 days.

Electrical-Power Requirements

1. Short Term "B" SG System (Approach 1)

   Power system requirements for Approach #1 are limited to supplying loss-of-offsite-power (LOOP) back-up to the Secondary Services Cooling Pumps. We understand this requirement will be satisfied by adding two new 2500 kw diesel generators to two existing non-safety 4160 volt busses. The primary system status has evolved to the point that quick-response fast-starting diesel generator performance is not required.

The entire system is to be a manually operated system (i.e. positioning of valves and starting pumps). The tertiary cooling loop for this modification (ultimate cooling) utilizes the Nuclear Services River Water Pumps which are already connected to the existing Class 1E diesel generators.

2. Short Term "B" SG System (Approach 2)

Power system requirements for approach #2 include the addition of two new 700 hp pump motors that will be powered off the two new 2500 kw diesel generators. We have no further information on the electrical requirements at this time. It has not been decided whether to use the secondary services river water pumps (which requires LOOP back-up from the new diesel generators) or the Nuclear Services River Water Pumps (which already receive Class 1E power) for ultimate cooling.

3. Short Term "A" SG System

Power system requirements to support this modification include supplying LOOP back-up protection for the Condensate Pumps and the Circulating Water Pumps. We understand that the LOOP back-up requirement for the condensate pumps will

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be satisfied by connecting these pumps to the new 2500 kw diesel generators. Consideration has been given to using a construction power line that comes into the site for supplying LOOP back-up to the circulating water pumps (2250 hp each). The use of this 13.8 kv line will not provide total LOOP protection if the entire grid in the area of the plant should experience a black out. However, we have approved this design concept on the following bases: a) the time window for needing the circulating water pumps is limited to the time in which the steam generator A short term system will be in operation, b) the starting requirements of the circulating water pumps demands a diesel generator much larger than has been located to date, c) Burns and Roe has stated that this back-up can be operational within 3 days of the approval, and d) this line will have remote connections to the 230 kv system and four 115 kv system lines which gives a reasonable degree of station blackout protection.

The remainder of the system is all manual and primary system conditions are such that no prompt response is required given system interruption due to such occurrences as LOOP.

4. Long Term "A" and "B" SG Systems

Long term modifications for each steam generator are proposed to be the same. The design calls for all new equipment

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including motor operated valves and high pressure pumps. These loads will be assigned to the two new diesel generators busses to assure LOOP back-up protection.

This system will have the provision for remote operation but will not require prompt response to such events as LOOP. Ultimate cooling will utilize the Class 1E Nuclear Services River Water Pumps which are already on the existing plant 1E diesels.

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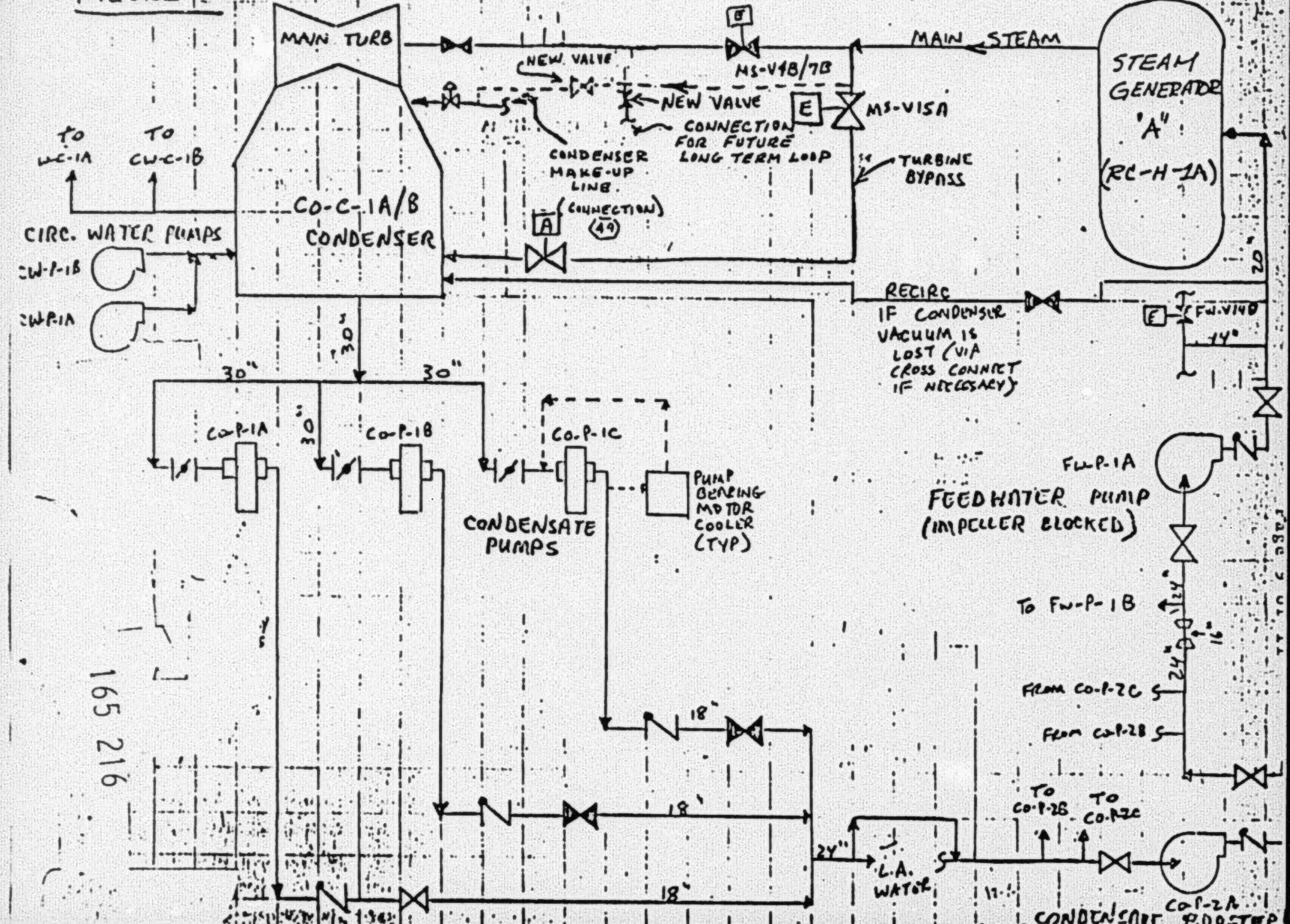
SHORT TERM - STEAM GENERATOR A

FIGURE 1.



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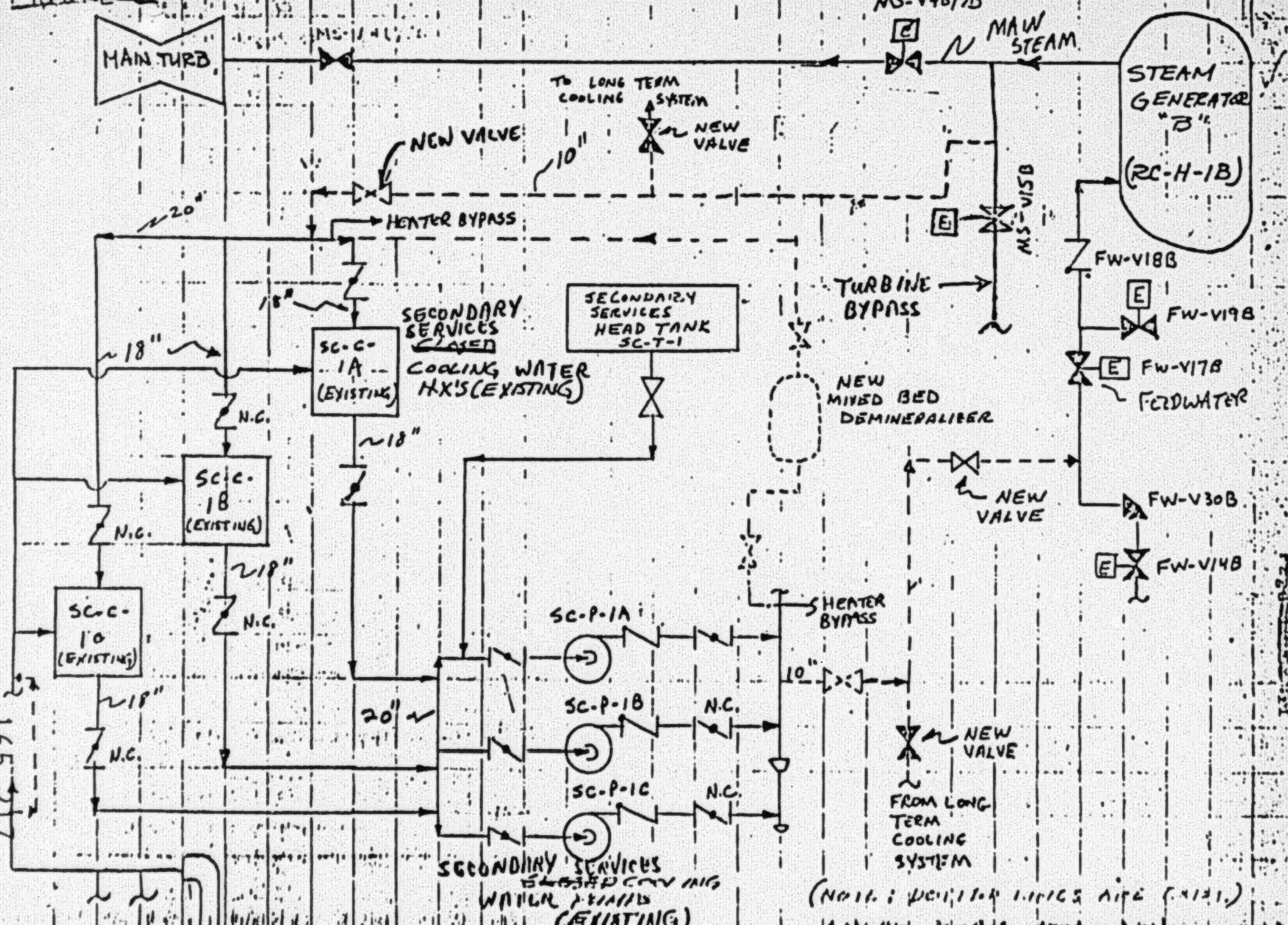

(NOTE: DOTTED LINES ARE (NEW))

SHORT TERM—STEAM GENERATOR "B"

FIGURE 2

APPROACH #1

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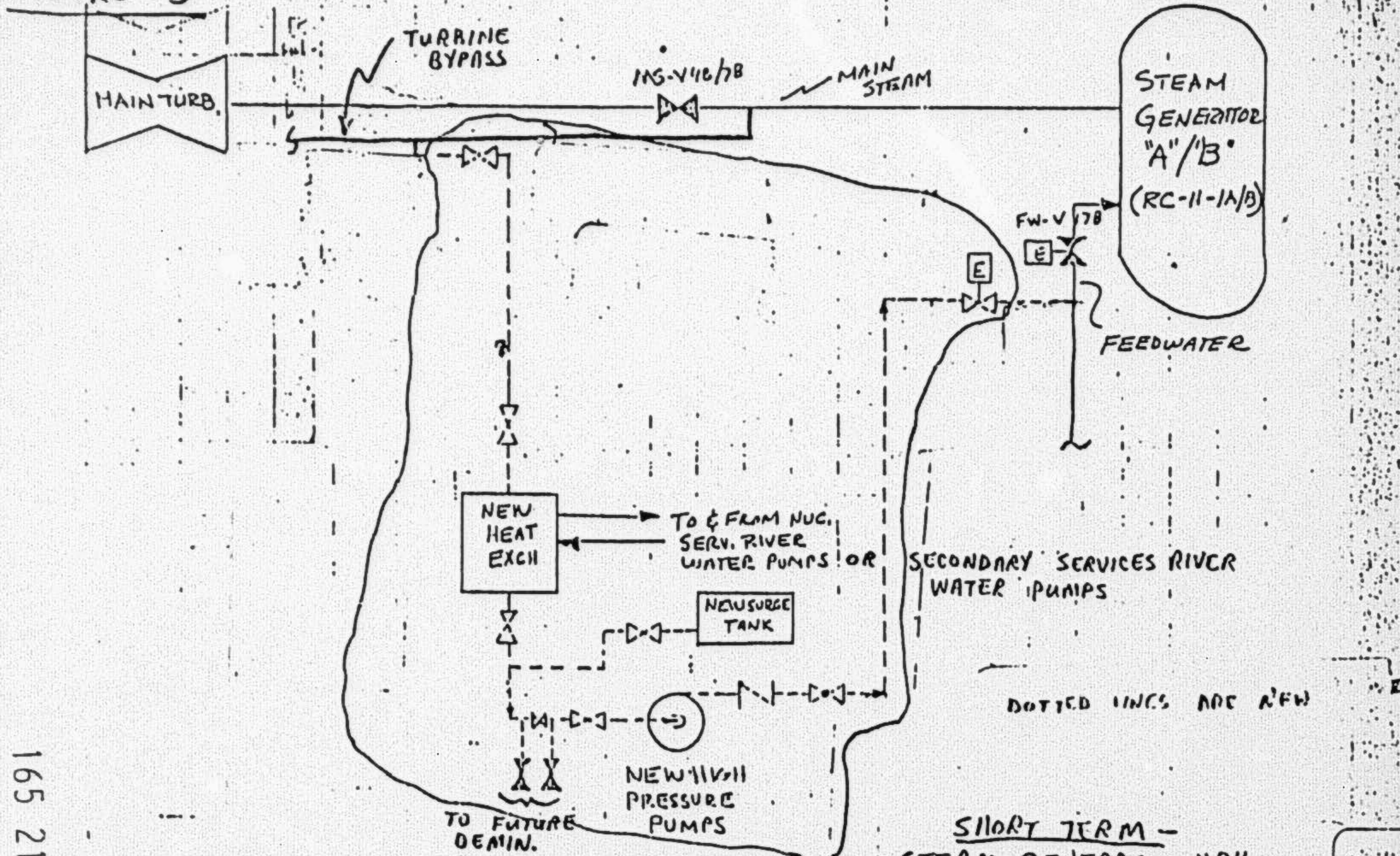
FIGURE 3



SHORT TERM – STEAM GENERATOR "B" APPROACH #2

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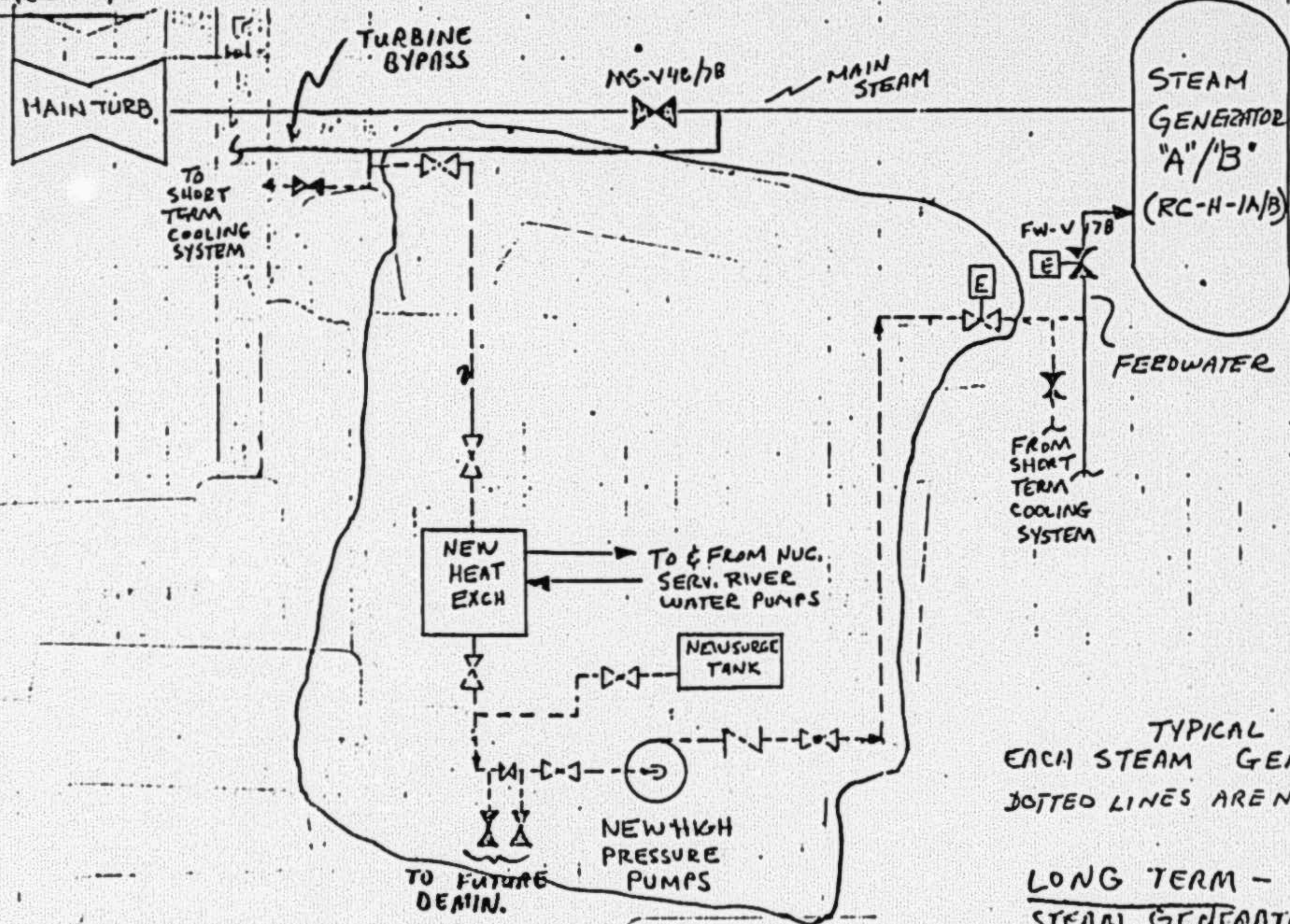
FIGURE 4



TYPICAL FOR EACH STEAM GENERATOR

DOTTED LINES ARE NEW

LONG TERM -

STEAM GENERATORS "A" & "B"

NEW HIGH PRESSURE SECONDARY SIDE COOLING SYSTEM

KEY: 1/1/80 17:00

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ENCLOSURE 2

DECAY HEAT REMOVAL SYSTEM

1. SCOPE

Westinghouse has been given the responsibility for direct decay heat removal from the primary system. Their work will be done in two parts as follows:

Upgrading Existing Decay Heat Removal System

Westinghouse will make provisions for and conduct a preoperational test of each loop of the existing DHR system. Locations of system leakage will be identified using television cameras installed at key locations. Once leakage paths are identified, they will be corrected if possible, thereby providing as leak tight a system as practical. Leakage collection capability will also be added to the system where feasible (i.e. collection of leakage around valves). Instrumentation to detect pump vibration will also be installed.

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Design of New Decay Heat Removal System

Westinghouse has proposed to install a third train for decay heat removal. This will involve a tie in to the decay heat removal system drop line downstream of valve DH-V-3 located in the fuel handling building (Aux. Building) and tie ins to the two return lines to the cold legs also located in the fuel handling building (See Figure 1). New lines will be run through the penetration room to an opening cut in the fuel handling building wall and out to a skid located outside the building at grade level (304'-6"). This skid will contain a new decay heat removal heat exchanger and two pumps. The discharge line for the heat exchanger will return through the opening in the fuel handling building to the return line penetrations. The tie in to the decay heat removal drop line will be made by welding an 8 inch weldolet to the pipe with a full penetration weld, dye penetrate testing the weld, then cutting the hole in the pipe using a plasma arc cutting process to minimize debris and finally welding the new pipe to the weldolet. A similar procedure would be used for the tie-ins to the two return lines. This procedure should minimize the time that the decay heat removal system will be out of operation. All valves will have two seals with provisions for collecting leakage and will be electric motor operated. Additional connections will be provided in the new piping outside the fuel handling building for future use in the installation of a long life, hardened structure which will contain heat exchangers, pumps, demineralizers and filters for long term decay heat removal and cleanup of primary water.

The secondary side of the new decay heat removal heat exchanger will be cooled by a new separate decay heat closed cooling water system with its own pump, piping and valves (See Figure 2). This system in turn will be cooled by a new decay heat service cooler. This cooler will be cooled by water from the nuclear services river water system. New connections will be made to this system.

The design of this new decay heat removal system will require that the pressurizer level be maintained half full at all times. Alternatively, a backup makeup and pressure control system as described in Enclosure 8 would serve to satisfy this requirement.

2. Structural

Fuel Handling Building Wall Penetration (Aux. Bldg)

Penetration would have to be made through the west wall of the Fuel Handling Bldg., between column lines AC and AF and across column line A68, at elevation 297'-0". (See B&R Drawing 2075). This is approximately seven feet below grade. Excavation outside of the structure will be done by pick and shoven to minimize a possibility of damaging any piping or electrical conduit. The outside wall at that location is reinforced concrete, 5'0" thick. The compressive strength of concrete in this wall is 5000 psi. The opening will be rectangular in shape, 3' horizontal by 4' vertical. The reinforcing bars will be cut on each side of the opening providing sufficient length exposed to install Cadwell type splices with the new reinforcing after the installation of the new pipe is completed. A concrete pour to reclose the penetration will occur following installation of piping runs.

Possible Problems

Because we have not had access to structural as-built drawings we have not been able to arrive at a definite conclusion regarding the concept of penetration of the wall, however, some problems can be foreseen. These are outlined below:

1. The exposed reinforcing ends must be of sufficient length to enable splices to be installed.

2. The exposed surface of concrete inside of the opening must be properly cleaned and prepared to receive the new concrete to fill the opening after installation of the new piping is completed, so that the joint between the new and the existing concrete remains leak-tight.

3. The safety margin of the structure after the wall is restored should be examined.

3. Electrical Power Requirements

This new system will have motor operated 480 volt valves arranged in such a manner that there will be two sets of isolation valves on each of the three DHR lines that will be tapped. These valves will be assigned power sources in a manner that assures isolation capability given a single power source failure. The remaining motor operated valves will be arranged on a "per loop" basis to allow selection of either of the two new 4160 volt pumps. These valves and the associated pumps will be powered from different busses to assure system function given a single power source failure.

The two busses that will be selected for powering this system are to receive diesel generator back-up power from the two new 2500 kw diesel generators. Westinghouse will also provide two 480 volt motor control centers. All electric powered equipment (valves, pumps, motor control centers and cabling) will be Class 1E system quality. System installation and power sources preclude this system from being a fully qualified Class 1E system. System functional requirements of the primary reactor coolant system preclude the need for a fully qualified Class 1E system due to the following considerations:

1. There are two existing DHR systems that are fully Class 1E.

2. Because of the low decay heat levels, it is expected that sufficient time would be available for manual operator action.

4. Mechanical Equipment Design

Pump and Heat Exchanger

ASME Section III Cl. 2

Valves

ASME Section III Cl. 1

Piping

Mixture of Type 304 and 316 10" Sch. 40 Stainless Steel sections.

ASTM Material Certification

All welds fully radiographed except for weld-o-let connections to existing DHR piping.

Design Criteria

Loads Considered - Normal plus OBE

Normal stress limits will be met for all piping and components including loads from OBE.

Design Stress Limits Used:

Pumps, Valves, Heat Exchangers - as specified in ASME, Section III for applicable Code Class.

Piping - Stress limits per ANSI B-31.7.

Valves including operators have been seismically qualified for worse seismic environment than would be experienced at TMI site in an SSE.

Pumps and Heat Exchanger - W to review B&W supplied Design Specification for specific seismic qualification and advise NRC.

Design Information: Specific For System to Existing DHR Piping, Weld-o-let Connection

Reinforcement area of fitting provides a 240 percent margin over the area of DHR piping it replaces.

Pipe supports will be arranged so maximum stress levels from the normal plus OBE load combination at the weld-o-let to DHR pipe interaction will be held to about one third of the B-31.7 stress limit for normal loads.

Weld-o-let to DHR pipe welds will be made using a qualified procedure and by welders qualified on weld-o-let to pipe connection mockups.

Because of time constraints, the weld-o-let to pipe welds will not be radiographed; however, the root of the weld will be ground and dye penetrate inspected and the final surface will be dye penetrant inspected. Additionally, the design is being qualified by hydrostatic pressure tests and bending moment tests which apply loads until the simulated DHR pipe exceeds its yield strength.

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All welds and the cut into the DHR pipe will be performed using the plasma arc method. The plasma arc was chosen for the combination of small heat-affected zone and minimum resulting slag which can be cleaned up.

Miscellaneous

Valves - Line valves and relief valves will have leakage or discharge fluid piped to a drain tank in the auxiliary building.

Decay Heat Closed Cooling Water System Components

ASME Section III CL.3

For all components

Materials: C.S. piping
S.S. Pump, Valves, Heat Exchanger

All connections welded except for piping to component interfaces which will be flanged.

5. Instrumentation for New Decay Heat Removal System

The following identifies the instrumentation to be provided by Westinghouse for the third DHR System train.

A trailer will be used to provide remote control room operation.

1. 4 DHR System Pressure Transmitters with remote control room indication. Two have low pressure alarms.

2. 2 T_{HOT} Sensors for each pump leg with remote control room indication.

3. 2 Pressure Transmitters Upstream of DHR cooler with remote control room Indications.

4. 2 Pressure Transmitters downstream of DHR cooler with remote control room indications.

5. 2 DHR System Flow, 2 Delta P transmitters from a single orifice with remote control room indications with low flow alarms.

6. 2 T_{COLD} sensors with remote control room indication.

The instrumentation supplied by Westinghouse will provide measurements of the important system variables - pressure, flow, and temperature. The pressure measurement provides system pressure at the suction of the pumps. From the pressure measurement, estimates of pressurizer water level can be made. The temperature measurements provide reactor T_{HOT} and T_{COLD} values.

No information has been provided regarding overall reliability, diversity, and quality of the sensors and transmitters (i.e. fully qualified safety grade or non-safety process instrumentation). It has been indicated by W that the instrumentation will be the best available within the perceived time constraints.

NEW WESTINGHOUSE DECAY HEAT REMOVAL SYSTEM

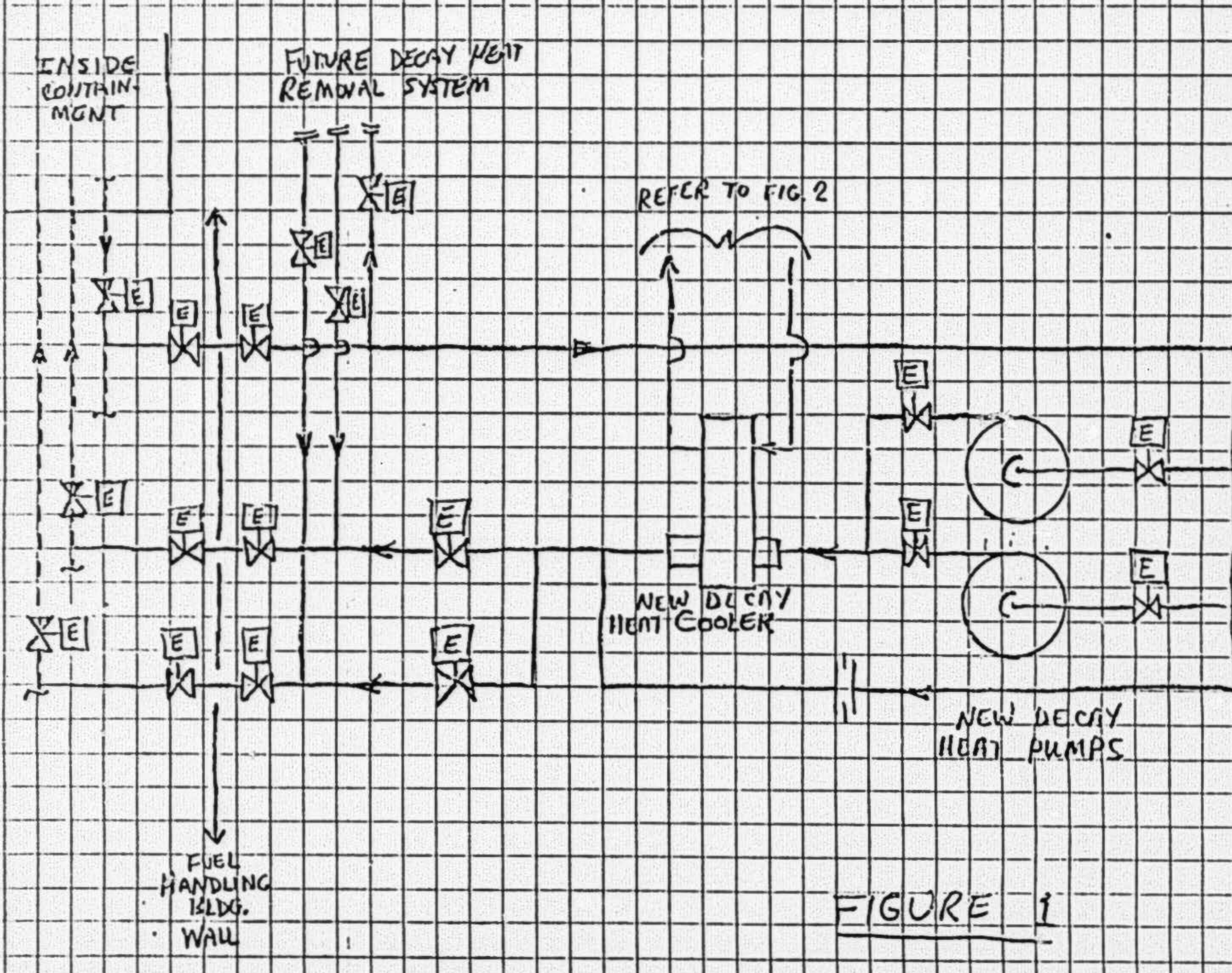

FIGURE 1

Dotted lines - existing system

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NEW WESTINGHOUSE DECAY HEAT CLOSED COOLING WATER SYSTEM

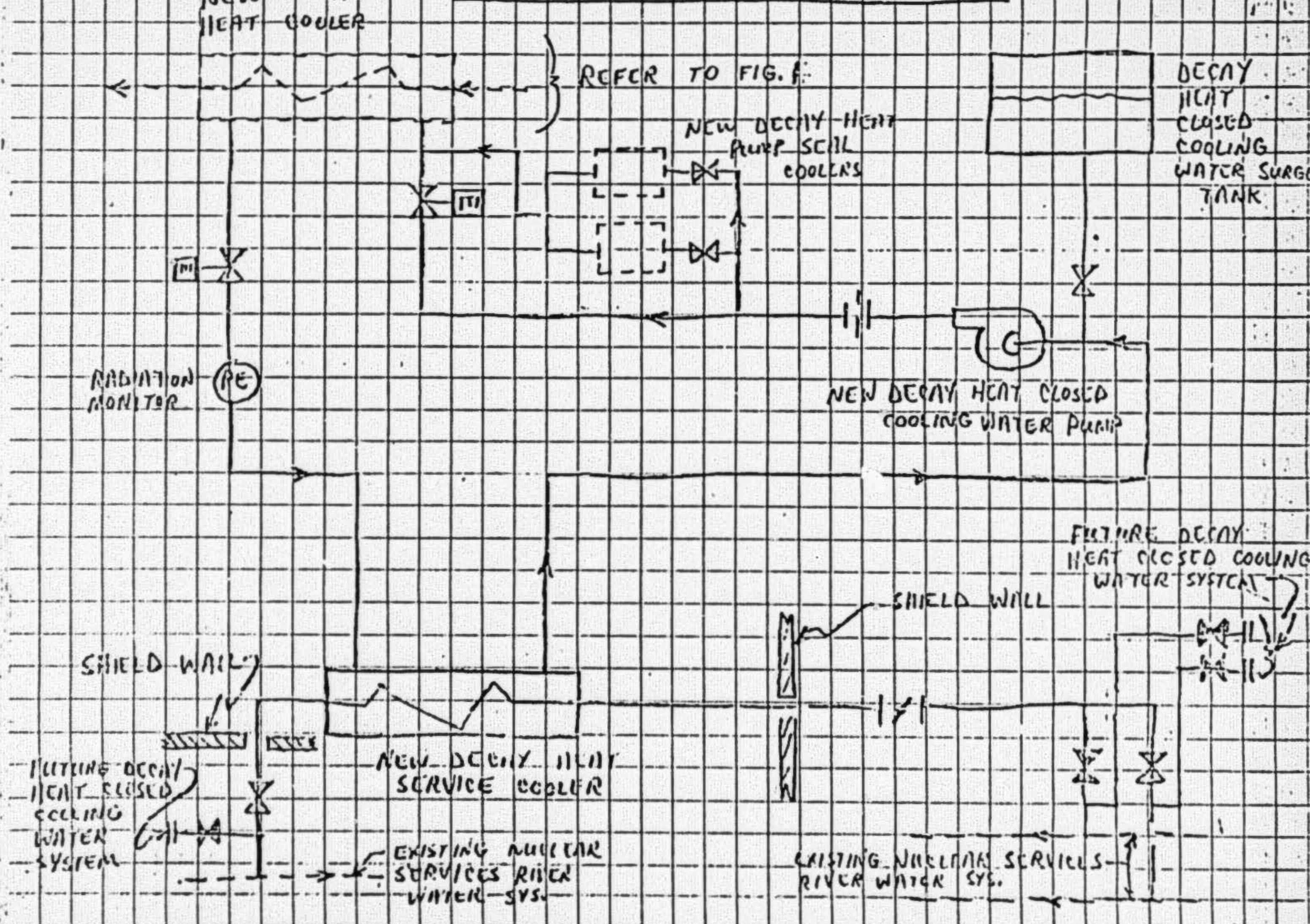

FIGURE 2

Dotted lines - existing system in reference to Fig. 1

ENCLOSURE 3

COMMENTS ON POSSIBLE OTSG DESIGN LIMITATIONS RELATING TO WATER SOLID OPERATION

Water Solid Steam Generator Flow Considerations

Prior to the accident, B&W had established an upper limit of 1000 gpm for secondary side fluid flow under water solid conditions. This limitation is apparently related to concerns over possible tube vibration. The steam generators are completely filled with water during normal operation of the plant for performing hydrostatic tests.

Although subject to confirmation from B&W, we believe that when such tests are performed the inside of the tubes are dry. When the OTSG is used water-solid for core cooling, the tubes will be filled. Because the water filled tubes will have considerably more mass and stiffness, it is unlikely that vibration would be a concern in the water solid shutdown cooling mode of operation.

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ENCLOSURE 4

ASSESSMENT OF RADIOLOGICAL CONSEQUENCES OF OPERATION OF STEAM GENERATOR B IN SHORT TERM MODE

The procedure for operation of Steam Generator B in the solid mode must consider radiological effects of this operation on radiation levels which may develop in the turbine building as well as the possibility that some leakage may occur across the Secondary Services Cooling Water Heat Exchanger and be released to the basin of the mechanical draft cooling tower. Calculations can be performed to assess the radiation levels expected assuming a source term based upon the latest known water activity levels and assuming this is in the existing inventory. These calculations can then be extended to evaluate a range of primary-to-secondary leakage rates. The need to include the latest primary water activity levels should be considered. The results of these analyses should suggest requirements pertaining to:

1. The need to address possible use or demineralizers, e.g. condensate demineralizers, to reduce secondary side contamination prior to use of secondary side of OTSG-B. This could offer a clue as to possible additional primary to secondary leakage.

2. The need to specify the intervals for sampling secondary water.

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3. The need to specify criteria for isolation of the Steam Generator B and return to Steam Generator A or some other mode of decay heat removal.

4. The need to specify levels of anticipated contamination in the turbine building, e.g., liquid leakage, and airborne contamination.

5. The need for additional shielding of any hot piping or demineralizers.

6. The need to control discharges of turbine sump water to a radiation controlled location for sampling or treatment prior to discharge.

7. The need to address locations and release points of all vents, drains, traps, and pressure relief valves and mechanisms for treatment if necessary.

8. The need to check valve line ups to assure no unnecessary contamination of "clean" areas.

9. The need to verify operability of the radiation monitoring pit monitor RM-C7 which reads out in Unit 1 control room and monitors the mechanical draft cooling tower blowdown.

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Steam Generator "B" Tube Leakage

There has been discussion about the fact that the proposed short term cooling system loop that would utilize the B Steam Generator may be vulnerable to leakage through the steam generator. It is not clear at this time how much this generator leaked during the initial portion of the transient and whether such leakage would occur at the lower temperature and lower pressure difference conditions of long term cooling. However, it is known there is now contaminated water on the secondary side of this generator. B&W has suggested that there are mechanisms other than tube leakage by which the contaminated fluid could have entered the secondary side during the March 28 accident.

Currently, there is about a factor of 10^4 between the primary coolant iodine concentration and the OTSGB secondary side concentration. This means it would take 10000 gallons of clean water on the secondary side to dilute one gallon of primary coolant to current secondary side activity levels. This could be construed to imply the leakage has stopped since the iodine activity in samples taken of OTSG-B appears to be decreasing with decay.

Leakage Across the SSCW Coolers

The potential for leakage across the SSCW coolers into the river water appears high as these coolers are not designed for high temperatures or pressures. We suggest that the possibility of pressurizing the cooling side above the shell side with the secondary water be considered. If demineralizers are not used to clean up OTSG-B secondary water and no additional primary to secondary leakage occurs, the tolerable leakage across the coolers is less than 0.1 gpm to restrict the river water outfall to Part 20 unrestricted area levels on iodine. This assumes dilution in the OTSG-B by flooding up to the MSIV and a mechanical draft cooling tower blowdown flow of 40000 gpm.

ENCLOSURE 5

SUMMARY OF B & W SUBMITTAL OF APRIL 10, 1979
REGARDING NATURAL CIRCULATION

Comments on the B&W submittal regarding Natural Circulation:

- The B&W recommended primary coolant temperature for initiation of natural circulation core cooling is 100°F. The reactor cooling system would be put into a water solid condition and with reactor coolant system pressure between 20 and 50 psia.

- B&W recommends both steam generators operating in a water solid mode at the time natural circulation is initiated with a 3000 gpm minimum (5000 gpm preferred) flowrate provided t o each steam generator.

- B&W also notes that a much lower reactor coolant pressure can be maintained with solid water secondary side as opposed to steaming on the secondary side.

- B&W suggested that operating both steam generators in natural circulation mode in a water solid condition provides an increase between 10 and 20 percent in natural circulation flowrates. For steady-state operation two steam generators operating in natural circulation water solid would not be required. The ease of transition, etc., suggests the need for two steam generators operating instead of only one generator in operation at the time when natural circulation is initiated.

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- B&W proposes a 100°F subcooling margin for defining acceptable natural circulation performance.

Comments on Other Natural Circulation Conditions

- 2-loop operation with both steam generators steaming at 230°F with secondary side water level at the 95 percent level on the operating range. This mode of operation has been verified by operation at the Oconee Station.

- Steam Generator A steaming at 230°F and Steam Generator B isolated with water at the 30 foot level. Natural circulation is estimated to be 10 to 20 percent less for this condition than with both steam generators in service.

- Single loop operation with Steam Generator A operating with a solid secondary. A minimum secondary flowrate of 3000 gpm will provide similar natural circulation to a single loop steaming with a 30 foot secondary side water level.

- 2 loop operation with both steam generators operating with solid secondary and 3000 gpm feedwater being provided to each steam generator. This is the preferred mode of operation.

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The following B&W comments on the transition from forced circulation to natural circulation are provided:

- It was assumed that the transition would be made with both Steam Generator A and B operating in a solid mode. The B&W analyses predict a minimum core flow occurring about 1 minute into the transient. Stable natural circulation flow is expected to occur after about 10 minutes into the transient. Hot leg temperature changes should be observed at about 5 minutes into the transient.
- Considering the 100°F sub-cooling criterion, the B&W recommendation is made to maintain reactor coolant system pressure as high as practical at the time the reactor coolant pump is tripped. For example, operating at 500 psig permits the reactor coolant temperature to increase 340°F before the temperature limit would be reached.

The following recommendations were made by B&W regarding the overall transition to natural circulation.

1. Reduce Reactor Coolant System temperature to 230°F with Steam Generator A steaming.

2. Slowly fill Steam Generator B solid with water and begin heat removal until stable temperatures are reached in the 200-230°F range and then proceed to isolate Steam Generator A.

3. Reduce reactor coolant temperature to approximately 100°F by increasing feedwater flowrate to Steam Generator B.

4. Fill Steam Generator A to a water solid state. Open feedwater flow to Steam Generator A and obtain stable 2 steam generator flow operation.

B&W also recommended a feedwater throttling and opening procedure timed with reactor coolant pump tripping as a procedural consideration for enhancing the establishment of natural circulation.

B&W recommends that an alternate decay heat system be available and ready to be placed in operation prior to attempting the transition into natural circulation.

ENCLOSURE 6

SPENT FUEL PIT MODIFICATION

Figure 1 shows the proposed Spent Fuel Pit Modification. The purpose of the modification is to dispose of high level auxiliary building waste. Two tanks will be located at the bottom of the pit, 25,000 gal each. Their elevation will be adjusted by means of additional beams so that there is a clearance of approx. 1 foot between the outside of the tank and the concrete structure. Four additional tanks 15,000 gal. each will be suspended from beams (A) (See Figure 1), supported by the existing rails. The cross bracing, consisting of structural L's will be provided to stiffen the suspension structure.

The tanks were originally intended for middle and low level waste at another nuclear plant (see Q/C discussion for further details). The tanks will be prefabricated together with their hangers and their supporting beams and the entire assembly consisting of one 15,000 gal. tank, its cross-bracing and the supporting beams will be lowered into the pool. The tanks are fabricated with their saddles consisting of stiffened steel plates. At this time no information is available regarding piping and electrical connections.

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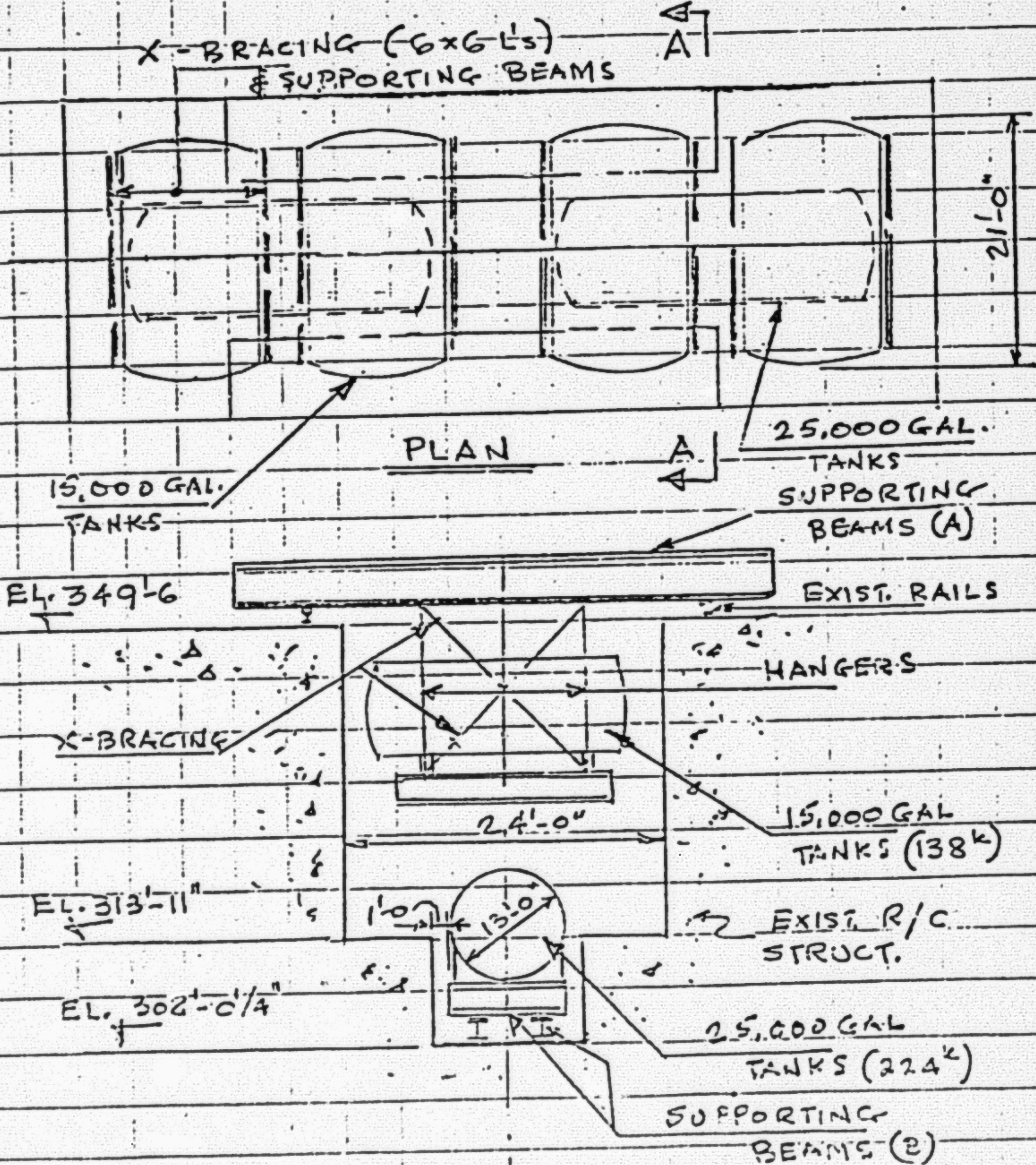
SPENT FUEL PIT MODIFICATION (FIG. 1)



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

MEMORANDUM FOR: T. Novak

FROM: J. Gilray

SUBJECT: QA PRACTICES ASSOCIATED WITH TMI-2 MODIFICATIONS

In regards to reviewing proposed modifications by the NRC Technical Review Group QA considerations has been given as to the extent QA practices, normally associated with 10 CFR Part 50 Appendix B (Seismic Category), should be applied to each modification. Due to the uniqueness of the TMI situation (i.e. urgency to get modifications complete and to within a short time frame) trade-offs to the traditional QA practices have been determined necessary but not to the extent that confidence is lost in meeting end use requirements. Accordingly the following is a QA summary of the QA practices associated with the modifications that have been reviewed to date.

1. Westinghouse Backup DHR Systems

   A. During 4/7/79 meeting with Westinghouse I met with QA personnel from GPU (T. Scalaite) and W (W. Meyer and J. Cowfer) to discuss QA controls that would apply to the DHR backup system. It was determined that a practical QA program approach is being applied to this effort. Highlights are as follows:

      1. Pullman (a code shop) is fabricating the piping and welding components to ASME - Section III, Class 2; welds will be nondestructive tested in accordance with this code. Welders will be qualified to Code; work will be procedurally controlled.

      2. In most cases there will be material test reports and tracability to heats. Where quality standards cannot be met, these events will be documented as nonconforming and a disposition made as to acceptability.

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3. Receiving inspection and storage will be controlled by QA personnel.

4. GPU QA will be involved with the DHR backup system activity as an overview QA control.

5. As design and specifications are matured they will be provided to the NRC Review Group for evaluation and comment.

II A. GPU System Criteria for the Steam Generator Cooldown System

Quality Standards will not be to ASME Section III due to the activity being confined to secondary side and schedule constraints. Controls applied will be to ASME Section VIII, ANSI B31.1, ANSI B16.5 and .34. QA practices will be coupled with this modification which will include controlled procedures and disciplines in the areas of design, fabrication, installation and testing of systems.

II B. Modifications for Rad Waste Storage Tanks

- A detailed evaluation of the Modification for additional Rad Waste Storage Tanks is not complete as yet.

- As a result of a preliminary review it appears that these tanks will meet:

  - Nuclear waste storage tanks standards
  - ASME Section VIII

- During the coming week of 4/16/79, a detailed evaluation and inspection of this modification will take place including welds and installation of pipes.

- The modifications are under close GPU QA control. The tanks were found unacceptable by GPU QA because of lack of a hydrotest and incomplete ~~poor~~ welds. The tanks were returned to supplier for additional work. I understand they are now back onsite and are acceptable.

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III. I&E Inspection

At the request of the NRC Technical Review Group, I&E was brought in on 4/13/79 from Region I to inspect the activities associated with implementing the TMI-2 Modifications. Special attention will be given by I&E to welding practices including the nondestructive testing of welds.

IV. Meetings

Meetings were held on 4/13/79 with Westinghouse (W) and GPU QA personnel to discuss their QA practices applied to modifications and controls. I&E attended these meetings. Summary of the two meetings are as follows:

1. Meeting with Westinghouse on 4/13/79 - 9AM
   Attendees: J. Wright, GPU, QA Mgr; B. Bruce, W, QA Mgr; D. Cowfer, W, QC Engr; L. Tripp and G. Walton, NRC - I&E; J. Gilray, NRC, NRR.

   - The QA scope and efforts of Westinghouse in the Design, Procurement, Fabrication, Inspection, Receiving, Installation and Tests associated with the DHR modification were discussed.

   - Westinghouse has established a QA program plan to control the above effort and will be commensurate with 10 CFR 50 Appendix B. A draft copy is attached as Enclosure #1. I find this document providing practical QA guidelines for the DHR modification acceptable.

   - Significant aspects of the above QA program were also discussed. Highlights are described in attached Enclosure #2.

   - The W QA organization responsible for DHR modification was discussed. Presently there are 4 W QA Engineers assigned to this Mod. qualified in 4 disciplines (Design and Design Control; Material and Component Traceability; QA for the Skid and Skid Components; and Welding and Installation). An additional QA Engineer will be assigned to this staff in the near future qualified in QA for I&C and Electrical systems. I find this organization and staffing acceptable.

   - The QA activities accomplished to date were discussed. Highlights are described in the attached Enclosure #3.

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- Conclusion: Based on a review and evaluation of W's QA practices, controls and organization it is concluded that the W QA program for the DHR system is sufficient to assure adequate QA controls and practices will be applied to this modification.

2. Meeting with GPU on 4/13/79 - 1PM
Attendees: J. Wright, GPU, QA Mgr; Earl Allen, GPU, QA Mgr; George Trofer, MetEd, QA Mgr; Terry Mackey, MetEd, QC Supervisor; T. Scalaite, MetEd, QA Engineer; L. Tripp and G. Walton, NRC - I&E; and J. Gilray, NRR, NRC.

- The QA Scope and Effort of GPU and MetEd associated with the TMI-2 modifications were discussed.

- The GPU/MetEd QA organization responsible for the TMI-2 modifications were discussed. A team of 12 QA engineers managed by J. Wright of GPU were brought in from the Forked River facility to be specifically responsible for the QA activities for the TMI-2 modifications. This team has talent in all QA disciplines including welding and nondestructive testing. I find this organization and staffing acceptable.

- This team has established a QA program specifically tailored for the TMI-2 Modifications and is expected to be approved by GPU and MetEd for use by 4/20/79. This program will be compatable with the MetEd Operational QA program previously accepted by NRC.

- GPU QA has demonstrated their QA involvement by inspecting all incoming structures systems and components designated for the modifications to determine their classification and quality. Those items where the quality is unknown are tagged accordingly and if used controls are implemented to maintain tracability. Nuclear waste storage tanks were returned to supplier because they were not hydro-tested and all welds were not completed.

- QA activities on TMI-2 modification being performed by GPU/MetEd QA personnel were discussed. Highlights are described in the attached Enclosure #4.

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- Conclusion: Based on a review and evaluation of GPU/MetEd's QA practices, controls, and organization, it is concluded that the GPU/MetEd QA program for the TMI-2 modifications is sufficient to assure adequate QA controls and practices will be applied.

V. Conclusions

Based on the above discussion, the QA programs and implementation of these programs by W and GPU/MetEd are considered acceptable to assure compliance with 10 CFR 50 Appendix B and the design and specification requirements for TMI-2 modifications.

J. Gilray

cc: R. Mattson

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Primary Makeup and Pressure Control System

Burns and Roe has a conceptual design for providing a standby primary makeup and pressure control system in the event further degradation occurs to the existing systems. The system consists of five 900 gallon tanks in series connected to the pressurizer spray piping. The last tank (furthest from spray piping) will be pressurized with nitrogen. The multiple tank concept minimizes the diffusion of nitrogen into the primary system. Two positive displacement pumps of variable capacity (10-90 GPM) will take suction from the first tank (closest to spray piping) to provide makeup. Level controls are installed on the tanks for cycling the pumps on and off. This system is scheduled to be operational in 20 days.

Burns and Roe recognizes the difficulty in identifying an acceptable tie in for overpressurization protection. They are currently proposing relieving into the 18 inch line that goes to the reactor building sump. We do not believe that this line should be opened for this purpose. Based on B&W analyses, the maximum vessel pressure for NDT considerations should not exceed 1000 psi.

Several alternative paths for manual relief were identified during a meeting with B&R, B&W and GPU on April 15, 1979. They included:

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1. Letdown to makeup tank (outside containment)
2. Relief valve on DHR system (inside containment)
3. Electromagnetic pressurizer relief valve (inside containment)
4. Pressurizer vent line valve (inside containment)

Letdown to the makeup tank and the pressurizer vent line appears to be feasible but will require further analyses. In addition, an analysis has to be performed to determine the most limiting overpressurization transient in the natural recirculation mode to establish the time before operator action is necessary.to respond to these transients. The limiting transient is probably one of of the following:

1. Loss of electrical power on secondary side.
2. Loss of natural circulation
3. Inadvertent makeup pump operation.

Sensitivity studies would also have to be performed with the primary system operating at various pressures to establish limiting operator action times. GPU and B&W plan to perform these calculations.

Two questions requiring further consideration were identified. They were:

1. Would the letdown system be operational during the natural recirculation mode?

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2. Does the operability of this system effect the schedule as to when the plant goes into the natural circulation mode of operation?