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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D. C. 20555

May 1, 1979

MEMORANDUM TO: R. Mattson
V. Stello

FROM: NRR Technical Review Group

SUBJECT: TMI-2 PLANT MODIFICATIONS FOR COLD SHUTDOWN (SEE APPENDIX)

Attached for incorporation into the staff report on TMI-2 post accident cooling is the NRR Technical Review Group's evaluation of several proposed modifications for achieving natural circulation.

The report was prepared based on the information available to the staff prior to the plant being placed into a natural circulation mode of cooling on April 27, 1979. It is expected that because of those operations, a reevaluation will be made regarding the benefits to be gained from certain changes in plant design.

NRR Technical Review Group

Attachment:
Staff Review of TMI-2
Modifications

cc: NRR Technical Review Group Members
T. Lainas

7906280173

TMI-2 PLANT MODIFICATIONS

FOR ACHIEVING COLD SHUTDOWN*

MAY 1979

*This report was prepared based on information available to the staff prior to the plant being placed in a natural circulation mode of cooling on April 27, 1979. It is expected that certain plant changes different than those described herein may result and will be evaluated in a subsequent report.

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TMI-2 PLANT MODIFICATIONS

FOR ACHIEVING COLD SHUTDOWN

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I. Background

In order to bring TMI-2 to a cold shutdown condition, modifications will be made to various plant systems in phases to be carried out over a period of the next few weeks. These modifications will permit a gradual transition from the current plant operating mode to one which provides a stable long term cooldown mode of operation.

The first planned modifications will be made in conjunction with the transition from forced primary coolant circulation (by the reactor coolant pump) to natural circulation through the core. To accomplish natural circulation, the secondary side (shell) of the steam generators will be operated water solid. Water will enter through the main feedwater ring and exit through the main steam line.

In order to provide for water solid operation, certain modifications to each steam generator secondary flow loop will be required. The design of modifications to steam generator "B" has accounted for possible contamination as a result of suspected tube leakage in the steam generator.

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To backup pressure control capability of the primary system during natural circulation, a new pressure control and makeup system will be provided. This system is essential in the event of loss of pressurizer heaters and level indication. Criteria and procedures for letdown and overpressure control of the primary system will be established prior to going into this mode of operation.

Because of suspected leakage in the existing plant decay heat removal system, a program will be conducted to identify and correct leaks to provide as leak tight a system as possible. Also, an additional skid mounted decay heat removal train will be connected into the existing system as a backup. Connections will be provided to the new train for a possible addition of a dedicated decay heat removal and cleanup system located in its own permanent structure.

To facilitiate early completion of design and installation of these system modifications, system functional capability following a seismic event has not been a design requirement. If a seismic event should occur and damage the modified systems, the seismic Category I TMI-2 Decay Heat Removal System and Reactor Coolant Makeup System could be used to remove core decay heat and control primary system pressure as necessary.

More detailed descriptions of these modifications are included in the following pages.

2. <u>Steam Generator Modifications</u>

a. <u>Modifications to Steam Generator "A" for Water Solid Operations</u>

1) <u>Design Concept</u>

The short and long term concept for water solid operation of Steam Generator A have been consolidated. One concept will now be utilized. Water will be circulated by the new pump through the shell side of a new heat exchanger and into the secondary side of the steam generator A in a closed loop to remove heat from the steam generator. The tube side of the new heat exchanger will be cooled by the existing Nuclear Services River Water System (NSRWS) which supplies water from the river and returns it to the mechanical draft cooling tower. Refer to Figure 1 for a schematic of this flow path. Provisions will be made for system pressure and expansion control by utilizing the existing 3rd stage feedwater heater shell and its nitrogen supply as a pressurized surge tank. The design also includes provisions for sampling, demineralization and chemical addition capability. For the initial phase of operation, all valves, will be manually operated and instrumentation will provide local read out.

The new loop of heat removal equipment has been designed to operate at a pressure higher than the expected reactor coolant system pressure thus assuring in-leakage of secondary system

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liquid into the primary system in the event of steam generator tube leakage.

2) Modification

This scheme will involve installation of a new high pressure train consisting of a pump, heat exchanger, valves and piping located in the turbine building basement. The loop will be connected to the main steam turbine bypass line between the connection to the main steam lines and the condenser, and to the main feedwater line between the feedwater pump and 3rd stage feedwater heater FW-J-6A. Horizontal runs of piping will be supported for static loads and secured to supporting structures to prevent lateral motion. Vertical runs of piping will be secured to permanent structural members as required. Additional piping will be required for the surge tank (3rd stage feedwater heater FW-J-6A), chemical addition tank and demineralizer. In addition, the interconnections between the A and B feedwater heater trains will be broken and capped off.

Jumper pipes will be installed between the existing Nuclear Services River Water System (NSRWS) the existing Secondary Services River Water System (SSRWS) to provide cooling water to the tube side of the new heat exchanger. The safety classification of the Nuclear Services River Water System will be maintained by providing double isolation valves.

All piping connections will be welded. These modifications are expected to be completed and the system ready for operation by the middle of May.

3) Systems Evaluations

The system design as proposed will meet the necessary secondary side requirements for decay heat removal with either forced or natural circulation through the reactor core. All required modifications will be made to accomplish this purpose. This system is completely independent and separate from steam generator "B" during all intended modes of operation with the exception that the Nuclear Services River Water System and Secondary Services River Water System will be shared by both loops of steam generator cooling.

The system will not be provided with redundant active components. However, a single active failure within the system will not compromise natural circulation of the primary system in that the secondary cooling loop through steam generator "B" will continue to operate (see staff evaluation of TMI-2 natural circulation performance).

The flowrates predicted through each of the heat exchangers will provide adequate cooling based on an assumed heat load of 30×10^6 Btu/hr (RCP operating + 3 MWDH). Operating performance and design parameters for the system are as follows:

Steam Generator A Modified System

System Location	Operating Conditions: Pressure (psig)	Temp. (°F)	Flow (gpm)	Design Pressure (psig)
New Pump Discharge	670	100	3000-5000	800
New Pump Suction (New Heat Exch. Disch. Shell Side)	500	100	3000-5000	600
New Pump Recirc.	670	120	Pump min. flow	800
New Heat Exchanger Supply (Shell Side)	670	120	3000-5000	800
NSRW Supply to New Heat Exchanger (Tube Side)	100	85	6000	150
NSRW Return from New Heat Exchanger (Tube Side)	100	95	6000	150

The system flow arrangement has been selected to minimize fouling effects by maintaining Nuclear Services River Water on the tube side of the new heat exchanger. System operating temperature indicated for NSRW supply and return are design values.

5. Modifications to Steam Generator "B" for Water Solid Operations

1) Design Concept

The short and long term concept for water solid operation of Steam Generator B have been consolidated. One concept will be utilized. Water will be circulated by the new pump through the tube side of the new heat exchanger and into the secondary side of steam generator B in a closed loop to remove heat from the steam generator. The secondary side (shell side) of this new heat exchanger will be cooled by the Secondary Services Closed Cooling Water (SSCW) system which is, in turn, cooled by the Nuclear Services River Water System, by way of the mechanical draft cooling tower. Refer to Figure 2 for a schematic of this flow path. Provisions will be made for system pressure and expansion control by utilizing the shell side of the existing 3rd stage feedwater heater with its existing nitrogen supply and thus will operate as a pressurized surge tank. The design also includes provisions for sampling, demineralization, and chemical addition capability. For the initial phase of operation, all valves will be manually operated and instrumentation will provide local read out.

The first intermediate loop of this scheme (that portion of the secondary system which removes heat directly from the steam generator) has been designed to permit normal operation at a pressure higher than the reactor coolant system pressure, thus assuring inleakage of secondary system liquid into the primary system in the event of steam generator tube leakage. However, the expected mode of operation from which to initiate natural circulation would involve higher primary side pressures.

2) Modification

This scheme will involve installation of a new high pressure train consisting of a pump (LTB-P-1), heat exchanger (LTB-C-1), valves and piping located in the turbine building basement. The loop will be connected to the main steam line at a 10" drain pot between the main steam isolation valve and the stop valve, and to the main feedwater line between the feedwater pump and 3rd stage feedwater heater FW-J-6B. Piping supports will be similar to those provided for Steam Generator A modifications. Additional piping will be required for the surge tank (3rd stage feedwater heater FW-J-6B), chemical addition tank (LT3-T-2), and demineralizer. In addition, the interconnections between the A and B feedwater heater trains will be broken and capped off.

Connections will be made to the existing secondary services closed cooling water system supply and return lines for cooling the shell side of heat exchanger LTB-C-1.

Jumper pipes will be installed between the existing Nuclear Services River Water System and the existing Secondary Services River Water System to cool the tube side of the Secondary Services Closed Cooling Water System. The safety classification of the Nuclear Services River Water System will be maintained by providing double isolation valves.

All piping connections will be welded. These modifications are expected to be completed and the system ready for operation by May 7, 1979.

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3) <u>Systems Evaluation</u>

The system design as proposed will meet the necessary secondary side requirements for decay heat removal with either forced or natural circulation through the reactor core. All required modifications will be made to accomplish this purpose. This system is independent and separate from steam generator "A" during all intended modes of operation with the exception that the Secondary Services River Water System will be shared by both loops of steam generator cooling.

The system will not be provided with redundant active components. However, a single active failure within the system will not compromise natural circulation of the primary system in that the secondary cooling loop through steam generator "A" will continue to operate (see staff evaluation concerning TMI-2 natural circulation performance).

The flowrates predicted through each of the heat exchangers will provide adequate cooling based on an assumed heat load of 10×10^6 Btu/hr (RCP operating + 3 MWDH). Operating performance and design parameters for the system are as follows:

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Steam Generator B Modified System

System Location	Operating Conditions Pressure (psig)	Temp (°F)	Flow (gpm)	Design Pressure (psig)
New Pump Discharge	670	100	3000-5000	800
New Pump Suction (New Heat Exch. Disch., Tube Side)	500	100	3000-5000	600
New Pump Recirc.	670	100	Pump min. flow	800
New Heat Exchanger Supply (Tube Side)	670	120	3000-5000	800
SSCW Supply to New Heat Exchanger (Shell Side)	150	72	4000	
SSCW Return from New Heat Exchanger (Tube Side)	100	59	5000	
NSRW Return from SSCW Heat Exchanger (Tube Side)	100	71	5000	

The system flow arrangement has been selected to minimize fouling effects by maintaining Nuclear Services River Water on the tube side of the Secondary Services Closed Cooling Water heat exchanger.

C. Mechanical System Design - Steam Generator "A" and "B" Modifications

All components and supports of both nuclear class and non-nuclear will be designed or verified to have been originally designed for the maximum loads that they could be exposed to during testing startup, and expected operation of the system, i.e., pressure, temperature deadweight, pump vibration, etc. The component design structural information is listed in Table 1.

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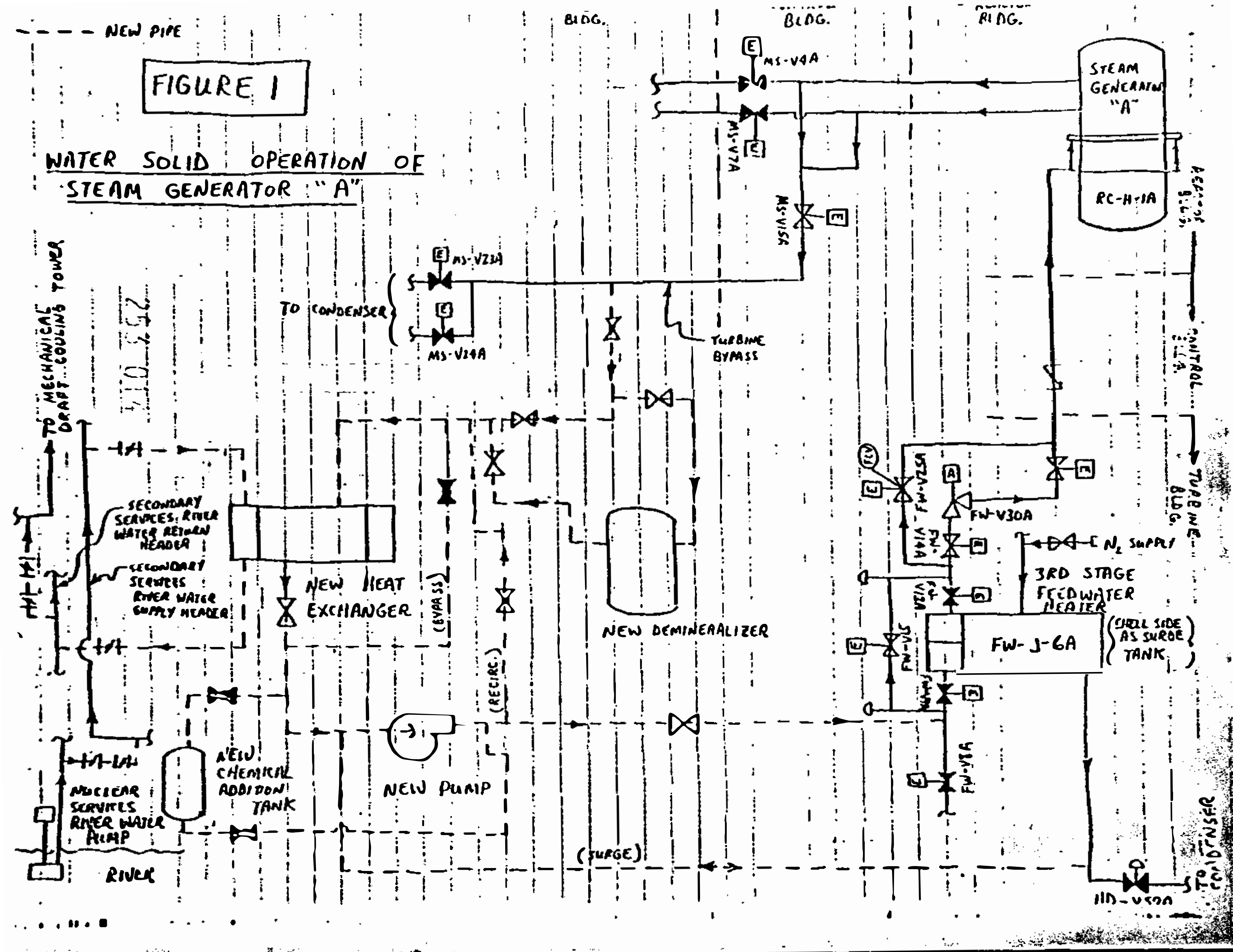
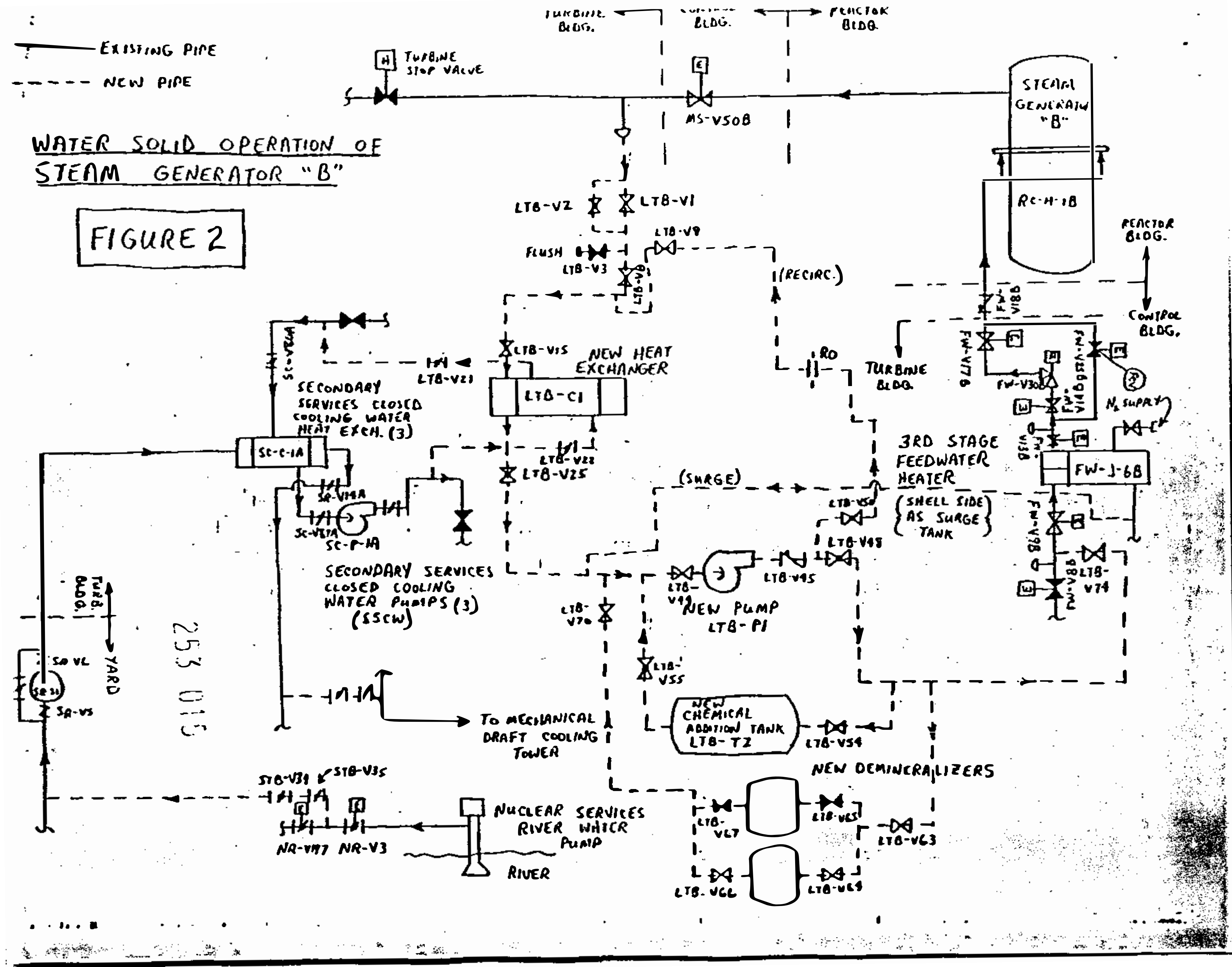


FIGURE 1

WATER SOLID OPERATION OF STEAM GENERATOR "A"

WATER SOLID OPERATION OF STEAM GENERATOR "B"

FIGURE 2



A specific concern that we have addressed in our review is the structural adequacy of that portion of the Main Steam piping system which as incorporated into the OTSG cooling system will contain solid water in lieu of the pressurized steam for which it was designed. Components in the system will not experience any significant dynamic loads. Special precautions will be taken during initial filling and startup of the system to minimize the potential for water hammer. Loads that will be experienced include pressure, deadweight of water, and thermal expansion. Since the system will be operated at a maximum pressure of about half the design pressure of the piping and its maximum operating temperatures will be considerably lower than the design temperature of the main steam piping; stresses resulting from these loads will be minimal.

After assembly and prior to initial operation of the plant, the existing piping was hydrostatic tested and at that time was water filled. Thus the piping and its supports have been demonstrated to be adequate for the weight of the water.

In order to minimize piping deflection, the licensee has specified that selected spring hangers be pinned. We concur with this requirement.

Design Bases Loads

It should be noted that all ASME Section III CL. 2 components used in the cooling system were designed for seismic Category I service. However all of these components, both those that are part of the original TMI-2 Main Steam and Feedwater piping system and those obtained from other nuclear sites to be incorporated into the OTSG cooling system, are being utilized in a system with different response characteristics from that for which they were initially designed or are operating with a fluid media different from that for which they were seismically qualified, i.e., some components designed for operation on steam during a seismic event as opposed to water filled as in the present system. Thus because of these differences from the original seismic design requirements, which can affect seismic response, these components should not be considered seismically qualified as installed in the proposed cooling system, solely on the basis of their original qualifications. Additional work would be required to evaluate the seismic capability of these components for this application. However, seismic capability of these system modifications is not a necessary acceptance criterion; therefore, no additional seismic evaluation of this system is planned.

Evaluation Conclusion

We have concluded that the Licensee has specified components designed and fabricated in accordance with acceptable industry codes or standards and will take into account the loads associated with startup, testing, and the planned system operation.

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The use of components that are in conformance with these criteria provides adequate assurance that structural integrity of the OTSG "A" and "B" cooling system will be maintained.

Table 1

Component Design Structural Information

Pumps A & B System

ASME Section III CL. 2

Design Temperature - 350°F
Design Pressure - 700 psig.

Heat Exchangers

A System

ASME Section VIII
Design Pressure: 150 psig (tube side)
600 psig (shell side)

B System

ASME Section III CL. 2

Design Temperature: 350°F (Both Shell and Tube Sides)
Design Pressure: 675 psig (ube side)
200 psig (shell side)

3rd Stage Feedwater Heater (shell side as surge tank)

ASME Section VIII

Design Pressure: 1000 psig

Steam Generator

Secondary Side - ASME Section III CL. 2

Piping, Valves and Mis. Tanks (minimum requirements)

Piping - ANSI B31.1
Misc. Tanks - ASME Section VIII Div. 1
Valves - ANSI B16.5 and B16.34

From feedwater check valves outside containment to the main steam isolation valves

ASME Section III CL.2

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Supports

ANSI B31.1 or as described in text.

d. <u>Structural Design - Steam Generator "A" and "B" Modifications</u>

Modification of Steam Generator "A" and Steam Generator "B" involves installation of one pump and one heat exchanger for each steam generator. The new equipment will be installed in the north-west end of turbine building at elevation 281'-8", between column lines TJ and TK in north-south direction and T41 and T44 in east-west direction. The enclosed Figure 1 shows the general area of location of the above equipment.

The details of the equipment is as follows:

The heat exchanger for steam generator "B" weighs 35 kips wet and is supported on two saddles, 2'-0" by 4'-0" each. The pump for this steam generator weighs 13.7 kips.

The heat exchanger for steam generator "A" weighs 135.3 kips wet. It is approximately 33 ft. long and is supported on two saddles 8 in. wide and 4'-3" long located 13'-0" apart.

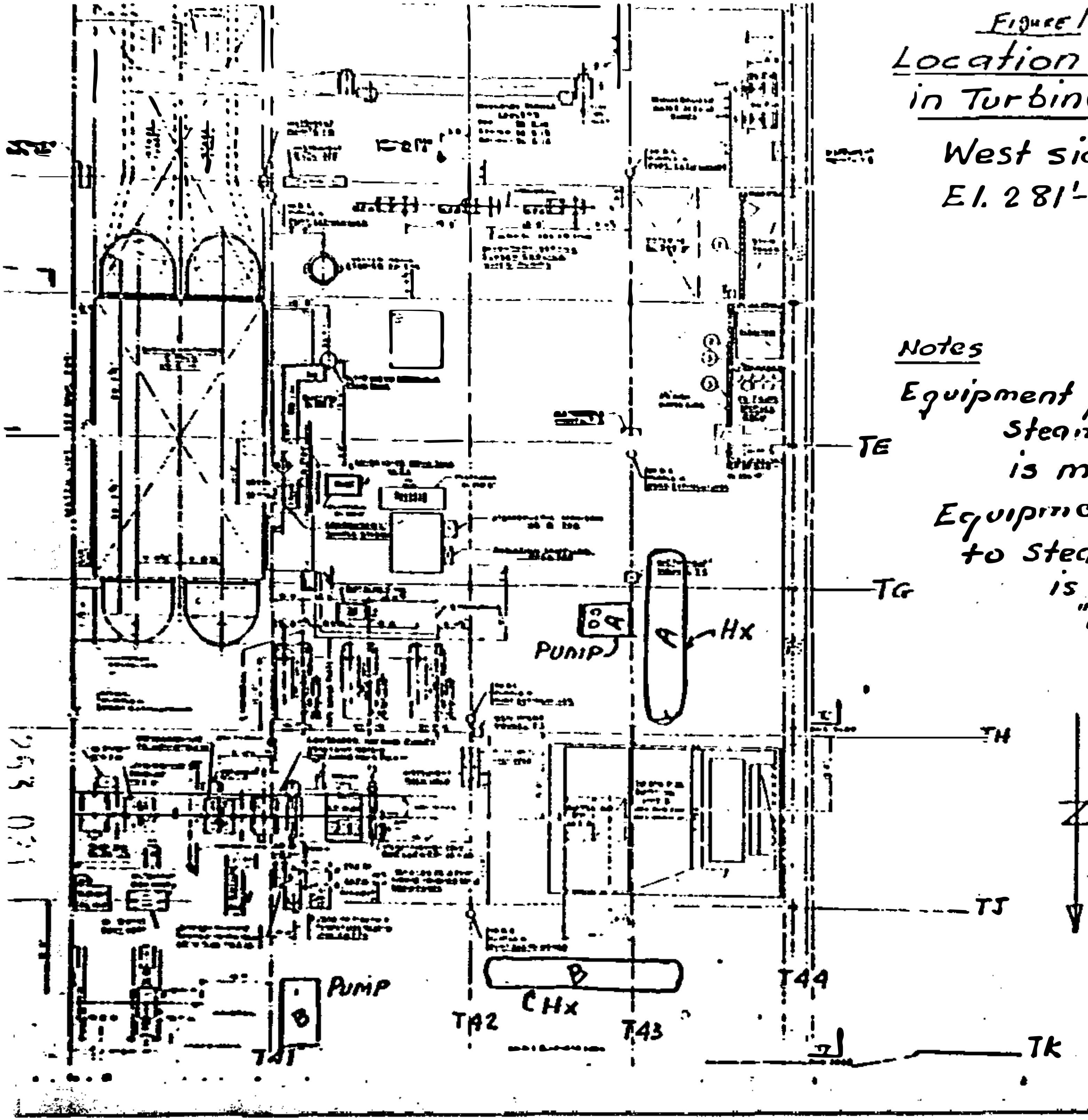
Examination of the existing structural drawings of the turbine building area, where this equipment is to be installed, reveals that the base slab is four feet thick, with #11 rebars at 12 in. spacing, each way, top and bottom. Top of the structural concrete base mat is at the elev. 277'-0". The base mat is covered with a 3'0" layer of lean concrete fill plus approximately 3 in. wearing surface, bringing the top elev. to 281'-3". Assuming 45 degree distribution of the load, and on the basis of the allowable concrete bearing pressure, the

Figure 1

Location of Equipm. in Turbine Bldg

West side

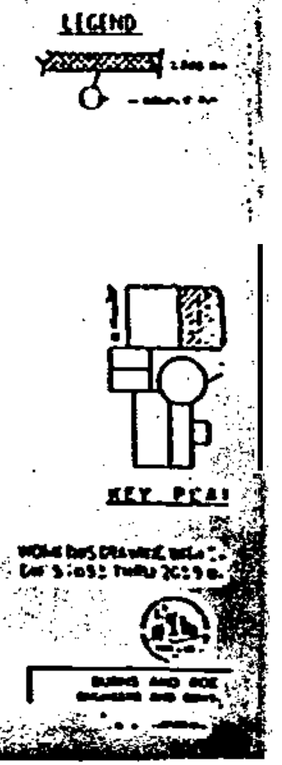
El. 281'-8"



Notes

Equipment pertaining to Steam Gen. "A" mod. is marked "A"

Equipment pertaining to Steam Gen. "B" mod. is marked "B"



licensee estimated that the maximum load that can be placed over one square foot of the area is 90 kips. Since the heaviest piece of equipment to be installed in this area is the heat exchanger, 135.9 kips, which when distributed over the area of the supporting saddles (2 x 4.67' x .67' = 6.1 sq. ft) delivers the head to the floor of 22 kips, the licensee concluded that the slab is capable to support the load. The analysis was performed for the dead and live loads only, under static conditions.

We have concluded that the licensee has performed his analysis in accordance with the methods and procedures which are specified by the appropriate codes and standards. The use of these methods provide a reasonable assurance that the structural elements affected by this modification will perform their intended function.

e. <u>Instrumentation and Control (OTSG "A" and "B" Cooling)</u>

The following new instrumentation (listed below) is being provided on the OTSG "A" and "B" Long Term Cooling Systems.

1. Radiation Monitor (* 'B' only)
2. Loop flow Indication*
3. Steam Generator Heat exchanger, Shell Side Temperature - T_{in} (*A only), T_{out}
4. Heat exchanger, Tube Side Temperature - T_{in}, T_{out}
5. Pump Pressure - Suction and Discharge
6. Loop Temperature*
[illegible]. Loop Pressure

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*Redundant sets of instrumentation are marked with an asterisk.

Control room alarms will be provided for radiation as a result of high S/G out-leakage. For initial installation and system operation, only local indication will be available mounted near the sensors in the turbine building. Similarly, all controls will be local.

S/G leakage into the cooling loops is to be sensed by victoreen model 355 area monitors strapped to the piping just down stream of the tie-in to the main steam line. Loop flow rate is to be sensed by 2 (two) barton model 200 differential pressure mechanical indicators across a single permutit orifice plate. All new temperature sensors/indicators are specified to be ashcroft (5 - inch code 50 El) bi-metal, liquid-filled thermometers. Loop pressures are to be sensed by ashcroft 1279 (bourdon system) mechanical pressure gauges. Table 1 provides additional instrumentation characteristics.

The system design criteria include the requirement to provide control and sensor read-out to the main control room on an expedited basis. Due to the time constraints placed upon initial system operation, we find the above design criteria to be acceptable. The specific details of the design associated with providing control and sensor read-out to the control room have not been developed at this time.

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OTSG "B" COOLING LOOP - TABLE I

VARIABLE	SENSOR TYPE	LOCATION
S/G "B" Leakage (primary to secondary)	Strap-on gamma Detectors Victoreen	1 monitor on existing main steam turbine bypass header ("A" loop only)
		2 monitors on new pipe just downstream of existing main steam header ("B" loop only)
Loop Flow		
Total Flow	Orifice Plate-Permutit	2 Indicators in new pipe just downstream of existing main stean header ("A" loop)
		2 indicators in new pipe just upstream of existing eedwater header ("B" loop)
	Bimetallic Thermometer-Ashcroft	In new piping just off mainsteam header
Heat Exchanger		
Shell Side, Temperature	Bimetallic Thermometer-Ashcroft	At heat exchanger
T_{in}, T_{out} Tube Side, Temperature T_{in}, T_{out}	Bimetallic Thermometer-Ashcroft	At heat exchanger
Pump Pressure		
Suction and Discharge	Pressure Gauges-Ashcroft	At discharge and suction of pump
Loop Temperature	Bimetallic Thermometers-Ashcroft	Downstream of pump recirc.
Loop Pressure	Pressure Gauge-Ashcroft	Upstream of heat exchanger

f. Radiological Evaluation (Steam Generator A/B)

The radiological considerations which are integral to the water solid steam generator secondary side cooling method will assure that radioactive effluents from contaminated systems will be controlled and minimized. Additional precuations will also be made to minimize occupational radiation exposures to operating personnel.

The secondary coolant presently in the "B" steam generator is contaminated due to the initial primary to secondary leakage which occurred on March 28, 1979. The measured radioactivity concentration and curie content is estimated in Table 1. Under normal conditions the secondary coolant pressure will be maintained at a value greater than the primary system pressure such that if steam generator leakage flow paths are available, the highly contaminated primary coolant will not enter the secondary coolant. However, it is expected that transients of short duration may occur such that a reverse pressure gradient could introduce additional radioactivity into the secondary coolant. To alert the system operators of such a condition, indicators and alarms for pressure and radioactivity in the secondary coolant have been provided. These indicators will alert the operator of an adverse condition so that corrective action can be taken prior to significant additional contamination of the secondary coolant.

The steam generator secondary coolant system and secondary services closed cooling water system will be periodically sampled and analyzed to determine if heat exchangers are starting to leak. Samples will be taken at a frequency of at least weekly or at any time there are indications that possible leakage may be occurring, e.g., increase in the steam generator closed cooling loop surge tank levels.

Leakage may also occur at various mechanical connections in the secondary cooling system. To the extent practical, locations where leakage is likely, e.g., valves with a known leakage history will be evaluated and a program to minimize leakage implemented. Even with a leak minimization program leakage of contaminated liquids to the floor drain system may still occur. This leakage should flow into the floor drain system and be collected in the turbine building sump, the turbine building sump pumps will be operated in a manual mode with an analysis of the sump water radioactivity content being made prior to sump pump operation. If the radioactivity content is low, the water can be discharged through its normal flow path to the river. If the radioactivity content is high, such that discharge would exceed technical specification limits, the sump water will be pumped to an appropriate radwaste system for treatment. Adequate free volume in the radwaste system will be provided for such contingencies.

Isolation between the secondary cooling system and services systems, e.g., nitrogen and demineralized water supplies to the surge tank, will be provided to prevent back flow of contaminated water.

Gaseous effluents from this system should be negligible. The noble gas inventory in the "B" steam generator is negligible because the steam generator was vented. Airborne radioiodine releases should also be negligible because the secondary cooling system is not vented (a nitrogen blanketed surge tank) and the low secondary cooling system temperature (100°F) results in a low air/water partition factor which reduces the volatility of the radioiodine.

The licensee is providing a demineralizer system to maintain secondary water chemistry and to reduce radioactive contamination. Details on this system are not available at this time. The demineralizers will be shielded and designed to permit resin bed disposal as radioactive waste. The shielding and spent resin handling system will also be designed to minimize occupational radiation exposures, e.g., the use of disposable demineralizers.

Following the TMI-2 incident there has been no indication of primary coolant leakage past the "A" steam generator tubes. Since the primary to secondary system pressure differential will result in in-leakage only, precautions similar to those evaluated above for the "B" steam generator modification are not required.

Table 1

"B" Steam Generator Radioactivity Content
(Activities based on 4/19/79 sample reported by B&W)

	Half-Life	April 19, 1979 Radioactivity concentration $\frac{(uCi)}{gm}$	(1) System Inventory (Ci)	May 9, 1979 Radioactivity (3) concentration $\frac{(uCi)}{gm}$	(2) System Inventory (Ci)
I-131	8.05d	.93	50.8	8.9×10^{-2}	9.9
Cs-134	2.1y	9×10^{-3}	.59	5.3×10^{-3}	0.59
Cs-136	13d	$8.x \times 10^{-3}$	.54	1.6×10^{-3}	0.18
Cs-137	30y	3.4×10^{-2}	2.2	2.0×10^{-2}	2.2

(1) S.G. water level = 368 inches = 6.5×10^{7} cc

(2) S.G. full at 625.5 inches = 1.1×10^{8} cc

(3) Assumes decay from 4-19 (21 days) plus dilution from filling the steam generator.

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3.0 STANDBY REACTOR COOLANT PRESSURE CONTROL SYSTEM

A. System Evaluation

1) Description

A standby reactor coolant pressure control and makeup system has been proposed by the licensee. This system would serve as a backup to the CVCS and maintain reactor coolant system pressure with the pressurizer filled solid with water. Prim , coolant system pressure will be maintained even with the loss of pressurizer instrumentation and inoperative pressurizer heaters. Also, the pressure control system will be designed to provide adequate NPSH to the reactor coolant pumps if they are needed.

The stanby reactor coolant pressure control system will consists of passive components (a series of water storage tanks and a surgetank with nitrogen blanket for pressure control) and active components (charging pumps). The system will control reactor coolant pressure over the range of 100 psig to 750 psig.

The passive reactor coolant pressure system which would be operated initially with local control. Additional instrumentation and remote control will be incorporated to permit automatic operation of the system. The active pressure control portion would resupply water to the surge tanks with added capability of providing additional makeup water directly to the RCS if needed.

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2) Modifications

The standby reactor coolant system pressure control and makeup system will involve installation of two 900 gallon capacity water tanks and one surge tank (all will be of the Westinghouse Boron Injection Tank design), nitrogen bottles, two 40 gpm positive displacement pumps, a degassed borated water supply tank, valves, and piping. This system will be connected between the discharge side of the high pressure makeup system downstream of valve MU-V-144C and upstream of valve MU-V16C. All the components will be placed in the fuel handling building.

The above modification will establish a flowpath of makeup water and pressure control through the normal makeup lines that connect with the reactor coolant loop cold legs. Chemical control of the degassified borated water used in the pressure control system will be provided by the present chemical addition system. Connections will be provided to accommodate the addition of boric acid, H_2, demineralized water and hydrazine, LiOH and NaOH. The degassed water tank will be replenished via piping connection from borated water transfer pump and boric acid batching tank. The boron concentration will be maintained in the range of 2200 to 4000 ppm. Figure 1 depicts the proposed pressure control system and interface connections to the existing systems.

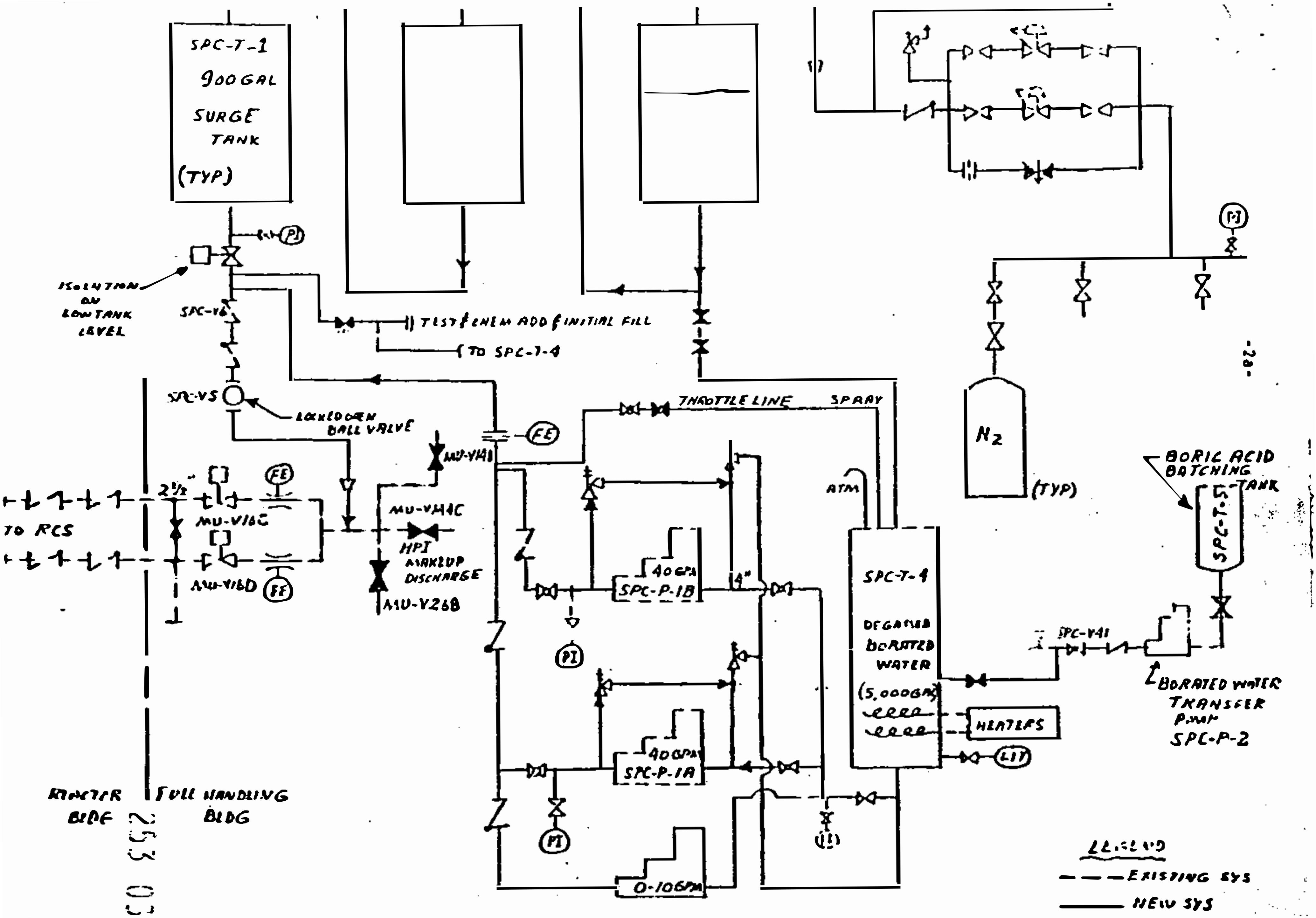

FIGURE 1. STANDBY REACTOR COOLANT PRESSURE CONTROL SYSTEM

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3) <u>Evaluation</u>

If required, the passive portion of the standby pressure control and makeup system is designed to provide peak initial 500 gpm injection rate to the reactor coolant system. The 500 gpm injection rate will be adequate to provide primary system makeup for certain transient events that can cause considerable shrinkage in the RCS. Because of the finite inventory (1300-2000 gal) this injection will decrease as the discharge proceeds. Also, the passive portion of the system will be designed to provide continuous makeup of 4 gpm for 8 hours.

Sufficient makeup volume requirement can be met by the proposed RCS pressure control system for moderate system perturbations and for the following postulated transient event: loss of natural circulation cooling due to a loss of all secondary side cooling with restart of one secondary cooling loop following a hot leg temperature rise of 50°F.

For this event the licensee has shown a total volume change of 1900 gallons which can be made up by the proposed system. However, the pressure control system is not designed for makeup requirements of more severe transients such as a sudden complete loss of natural circulation for a period of 2 hours, followed by an RC pump start. Procedures which permit reactor coolant pump restart following loss of natural circulation would require the availability of other makeup systems such as the HPI in addition to the proposed pressure control system.

The reactor coolant pressure control and makeup system will not provide letdown capability of reactor coolant caused by overpressurization events. Reactor coolant fluid expansion will be relieved by one or a combination of the following current components and systems: (1) normal letdown line (through the letdown coolers); (2) maintaining letdown with concurrent termination of makeup/seal injection; (3) continued reactor coolant pump seal return flow; (4) opening of pressurizer vent valve; and (5) lifting of the pressurizer relief valve.

Piping integrity of the reactor coolant pressure control and makeup system has been examined for postulated overpressurization events such as inadvertent startup of an HPI pump. This system will have a design pressure of 1000 psig except for the section from the HPI makeup line back through the second isolation check valve which will have a design pressure of 1500 psig. Overpressurization protection of the latter piping section when the HPI/makeup pump is started will be provided by installation of a relief valve about the HPI/makeup pump set at 1000 psig and a relief flowrate of 525 gpm. Check valves located downstream of the HPI/makeup pumps that are inside the Reactor Building will also provide protection to this system from inadvertent overpressurization of the RCS due to any other causes. The criteria and procedures for letdown and overpressure control of the RCS will be established prior to the operation of this system.

253 055

The proposed reactor coolant pressure control and makeup system may not meet makeup demand following a depressurization event of the primary system such as inadvertent opening of the pressurizer PORV (with downstream isolation valve open). If not isolated in time, the system would be drained resulting in entrainment of nitrogen into the reactor coolant system. To prevent such an occurrence the system will automatically isolate on low level in the water tank. In the event of loss of offsite power this valve will fail in the "as is" position. Since this position is the preferred position for normal operation and it would also be the preferred position in the event of loss of offsite power. Also, an alarm will be annunciated locally and in the control room when the isolation valve is not fully open.

We have examined for single failures that can disable the pressure control and makeup system. The discharge valve SPC-V5 is a single failure point, however it will be a manually operated ball valve positioned to an open position, and then locked in that position. For simplicity of design and installation, we have not required redundant valves to meet single criterion to insure system isolation capability. A redundant charging pump will be available to fill the water tanks in response to tank level reduction.

253 034

Requirements for preoperational flow test of the RCS pressure control system will be determined pending a review of the margin suggested by analysis for peak makeup flow demand and what the system is designed to provide.

We conclude that the proposed system will maintain reactor coolant pressure control for normal water solid conditions and provide sufficient makeup water for a wide range of expected transient events that would cause shrinkage in the RCS. Emergency procedures will be u ed for additional makeup capability that may be required to mitigate more severe and less probable transient events.

B. <u>Mechanical Design</u>

1) <u>Description</u>

The Reactor Coolant Pressure Control System is madeup of several water supply tanks, positive displacement pumps, valves, and piping. The applicable design codes and standards used for the design of these components are provided in the Table 1.

Table 1

Applicable Design Codes or Standards

Water Supply Tank (Passive System)

ASME Section III CL. 1 - S.S.

Charging Pumps

ASME Section III CL.2

Piping

ANSI B31.1 (minimum requirement)

Degassified Water Supply Tank

ASME Section VIII Div. 1

Component Supports

ANSI B31.1

2) Design Loads

Pressure, pump vibration, component and fluid deadweight, and maximum anticipated pressure surge forces were considered in developing design loads for the RCS makeup system. The staff considers these to be acceptable load considerations for this application. The staff has not required the RCS pressure control and makeup system be evaluated for seismic load capability.

3) Other Considerations

Welded construction will be used wherever possible to minimize the potential for system leakage. Components will be fabricated from stainless steel or carbon steel clad with stainless steel.

4) Evaluation Conclusion

We conclude that the licensee has specified components that will be designed and fabricated in accordance with acceptable industry codes or standards and will taken into account the loads associated with startup, testing, and expected system operation.

The use of components that are in conformance with these cirteria provides adequate assurance that structural integrity of the Reactor Coolant Pressure Control System will be maintained.

C. Structural Design

The system includes the following major equipment:

a) Three water tanks, 900 gal. capacity. Each weigh 20 kips empty and 27.5 kips when full of water. Each tank will be supported on four 12 in. by 12 in. plates.

b) Borated water tank weighing 60 kips supported on four 6 in. plates.

c) Two - 100 HP pumps weighing 5.4 kips each.

The enclosed Figure 2 (2 sheets) show the conditions of the original structure. The area is located between columns AP and AT in south-north direction and columns A65 and A67 in east-west direction. The slab is three feet thick and the reinforcing steel is #8 at 9 in. top and bottom in north-south direction and #8 at 6 in. top and bottom in the east-west direction. The compressive strength of the concrete is 3000 psi and the yield stress of the reinforcing steel is 60 ksi. The licensee has analyzed the slab for the additional loads resulting from the new equipment and concluded that the stresses will be within the allowables using generally accepted codes. The analysis was performed for the static conditions only. We conclude that the licensee has performed his analysis in accordance with the methods and procedures which are specified by the appropriate codes and standards. The use of these methods provide reasonable assurance that the structures affected by this modification will continue to perform their intended function.

253 038

Fig 2 (Sheet 1) AUXILIARY & FUEL HANDLING BLDG EL 323'-0" W.O: 2555

RCS PRESSURE CONTROL SYSTEM

BURNS & ROE DWG # 4250

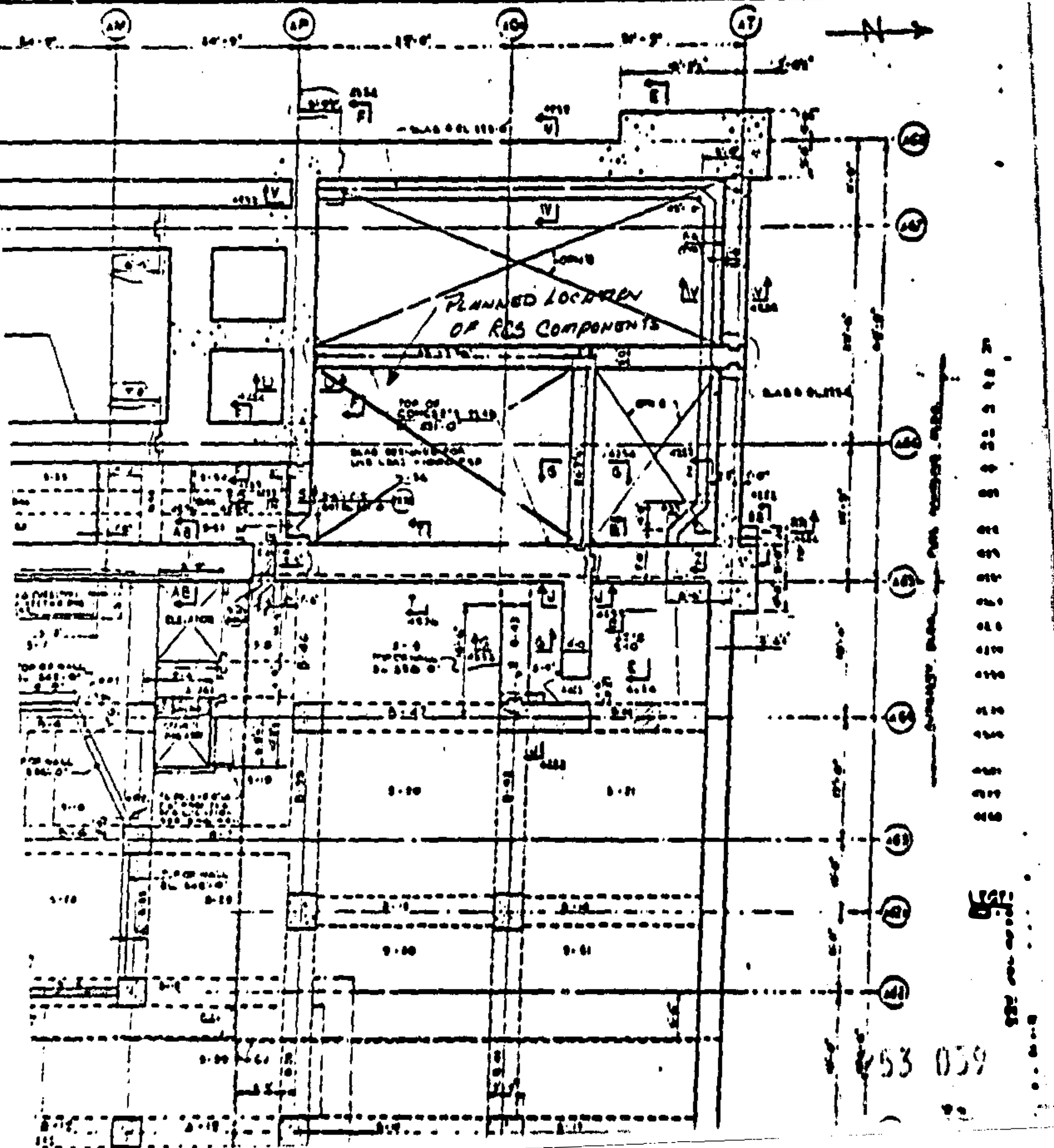
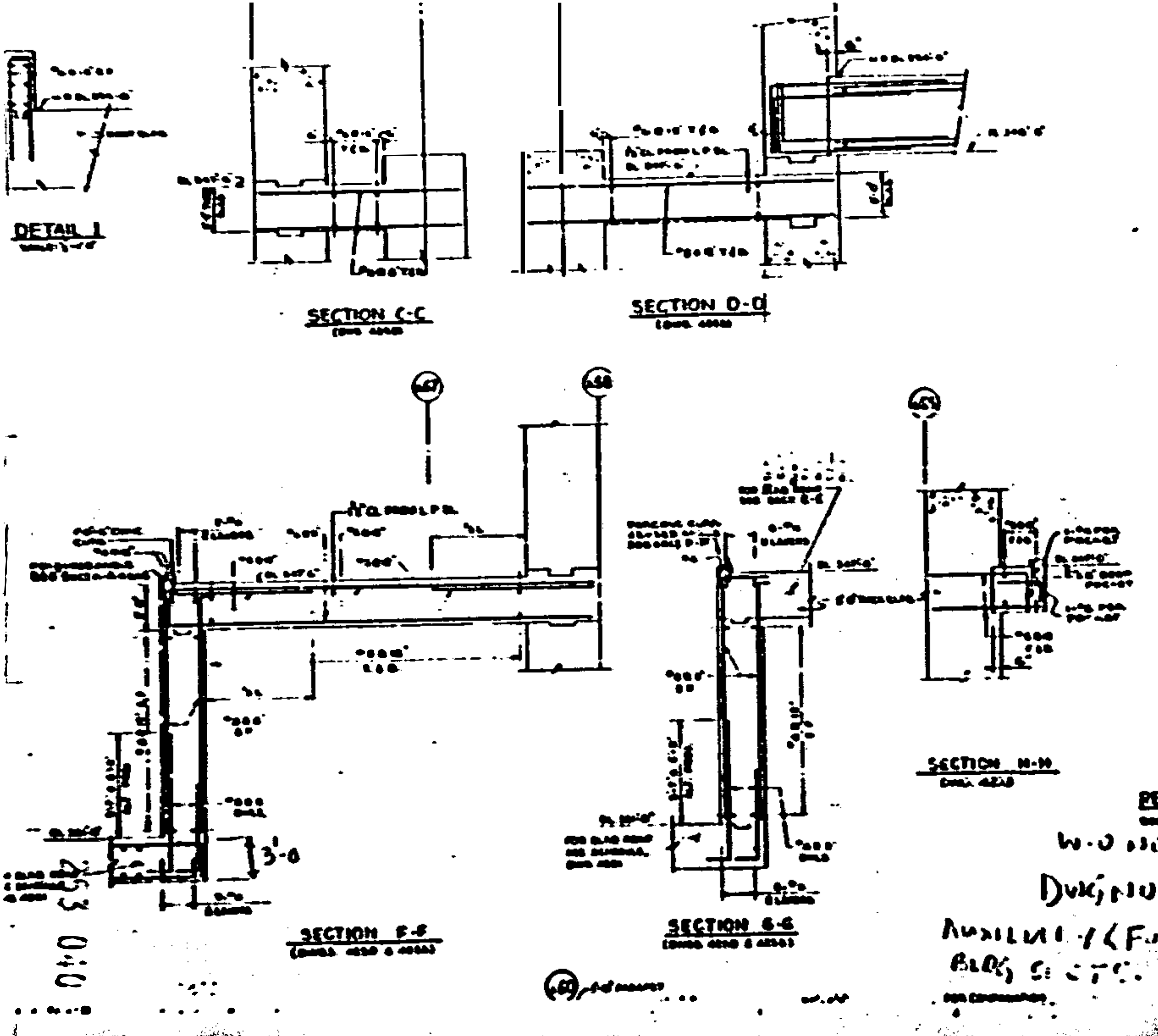


Fig. 2 (Sheet 2)



REFERENCE DWG

W.O. NO 2555.

DWG NO 4254

AUXILIARY (FUEL HANDLING) BLDG SECTS.

253 040

3. Instrumentation and Control

The following instrumentation is to be provided for the Reactor Coolant Pressure Control System:

1. System Water Pressure
2. Water Level in each tank
3. Nitrogen Pressure
4. Make-up Flow
5. Make-up Pump Discharge Pressure
6. Borated Water Storage Tank Level
7. Borated Water Supply Temperature

Redundancy is provided for: Water level (in each tank) and system water flow. Nitrogen pressure is sensed at the cylinders and just downstream of the N_2 pressure regulators.

The design of the system is not predicated on the availability of the pressurizer instrumentation and controls. The ranges of instrumentation have not been provided.

The control aspects of this system are as follows: The first make-up pump will be cut in by a "Low" level signal from the nitrogen pressurized surge tank; the second pump will be cut in by a "Low-Low" level signal; and both pumps will be cut out by a "High" level signal. The controls for the heaters have not been described.

253 041

We have required that the design include automatic isolation capability to preclude the addition of nitrogen into the Reactor Coolant System. The licensee has responded by providing an automatic close signal to the motor operated isolation valve initiated by low level in the tank nearest the Reactor Coolant System. The isolation valve will be signaled to close when water in the tank is depleted to 300 gallons. Tank pressure taps will be utilized for level indication. (One pressure tap in the tank and the other in the 6-inch piping upstream of the tank). Because of the location of pressure taps and uncertainty of level control to actuate the isolation valve under transient conditions we require functional testing of this control system that would require tank blowdown prior to installation of the pressure control and makeup system. If the test results turn out to be unsatisfactory other means of preventing N_2 insurge into the RCS such as automatic isolation and venting of N_2 supply will have to be considered. We have further required that an alarm be provided to the operator when the differential pressure of the Reactor Coolant System and the pressure control system exceeds a set value. The licensee has complied and the alarm will be annunciated whenever the differential pressure is greater than 50 psi. We find the instrumentation and control aspects of this system as described above to be acceptable.

4. Decay Heat Removal System (Existing and Skid-Mounted)

Decay heat removal capability will be provided for the plant for its current operating modes. To accomplish this, the following work and modifications will be undertaken: 1) upgrade the leak tightness of the existing decay heat removal system as required, 2) install an additional skid mounted train of decay heat removal equipment, and 3) consideration of a dedicated long term decay heat removal and primary coolant cleanup system in a permanent structure.

A. Upgrading Existing Decay Heat Removal System

Provisions will be made for conducting 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 is practical. Leakage collection capability will also be added to the system where feasible (i.e., collection of leakage around valves). Instrumentation to detect DHR pump vibration will also be installed. The availability of the existing decay heat removal system will depend on the final leak tightness of the system considering the contamination levels present in the primary coolant.

B. Design of New Skid Mounted Decay Heat Removal System

A third train for decay heat removal will be provided. This will involve a tie into the existing decay heat removal system drop line downstream of valve DH-V-3 located in the fuel handling building and ties into the two return lines to the cold leg 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 elevation (304'-6"). This skid will contain a new decay heat removal heat exchanger and two full capacity pumps. The discharge line for the heat exchanger will return through the opening in the fuel handling building to the return line tie-ins. The tie into the decay heat removal drop line will be made by welding an 8 inch weldolet to the pipe with a full penetration weld, dye penetration 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 be electric motor operated and will have two seals with provisions for collecting leakage and directing it to the existing rad-waste system. Additional connections will be provided in the new piping outside the fuel handling building for future use in the installation of a dedicated, long life, hardened structure which will contain heat exchangers, pumps, demineralizers and filters for long term decay heat removal and cleanup of primary coolant 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 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 be compatible with the current primary system decay heat levels and operating parameters. It will require that the pressurizer level be maintained half full at all times to insure adequate NPSH to the DHR pumps. Alternatively, the backup makeup and pressure control system could serve to satisfy this requirement. The new skid mounted DHR system and closed cooling system is completely separate and indepent from the existing DHR but is not designed against single active failure. However, the existing DHR system would be available in this event.

The plan is to have the skid available but not make the connection to the lines penetrating the fuel handling building wall at this time.

In the event this sytem would be required for immediate service, the skid and piping would be shielded to provide all reasonably achievable radiation protection.

NEW WESTINGHOUSE DECAY HEAT REMOVAL SYSTEM

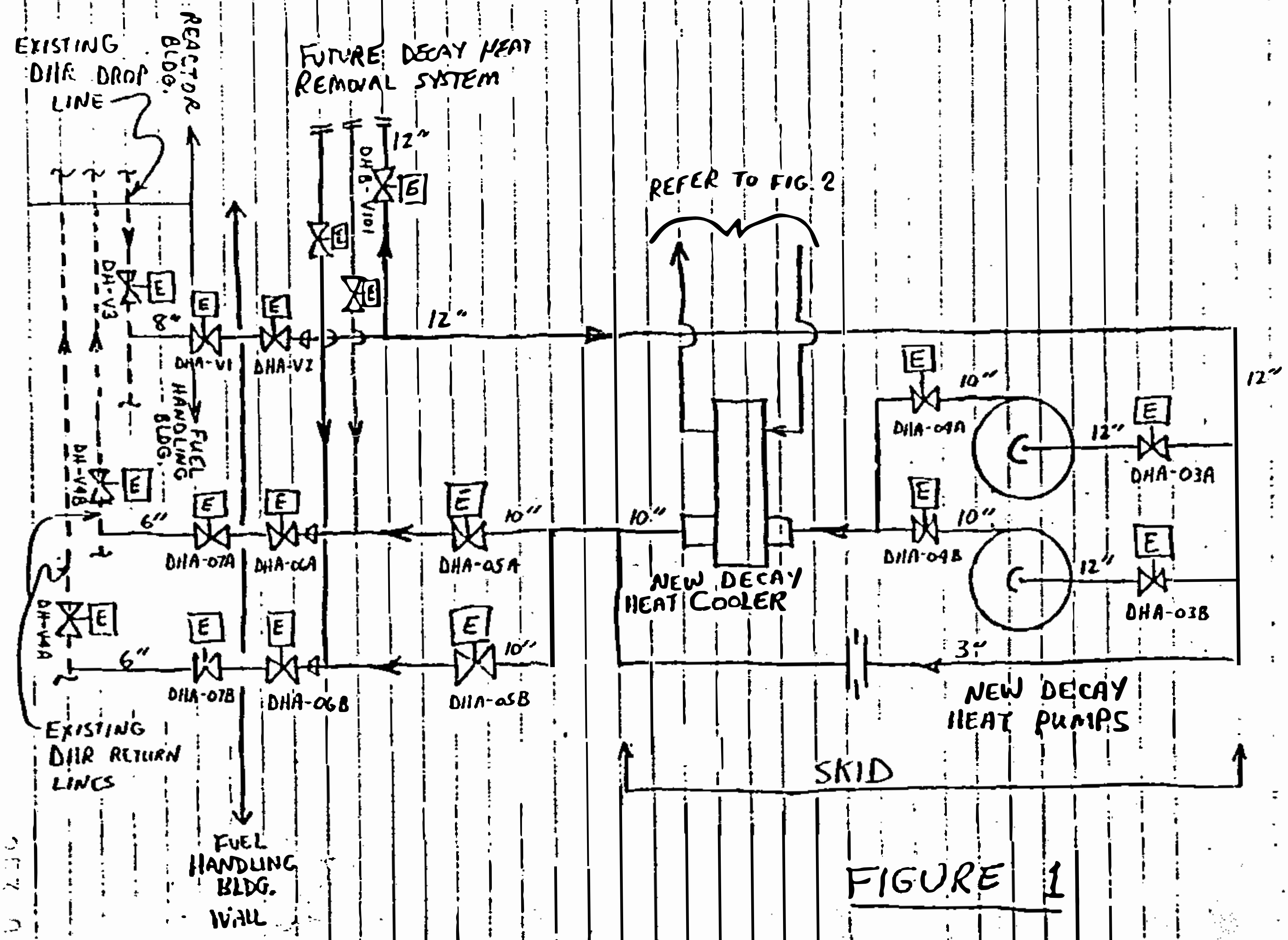
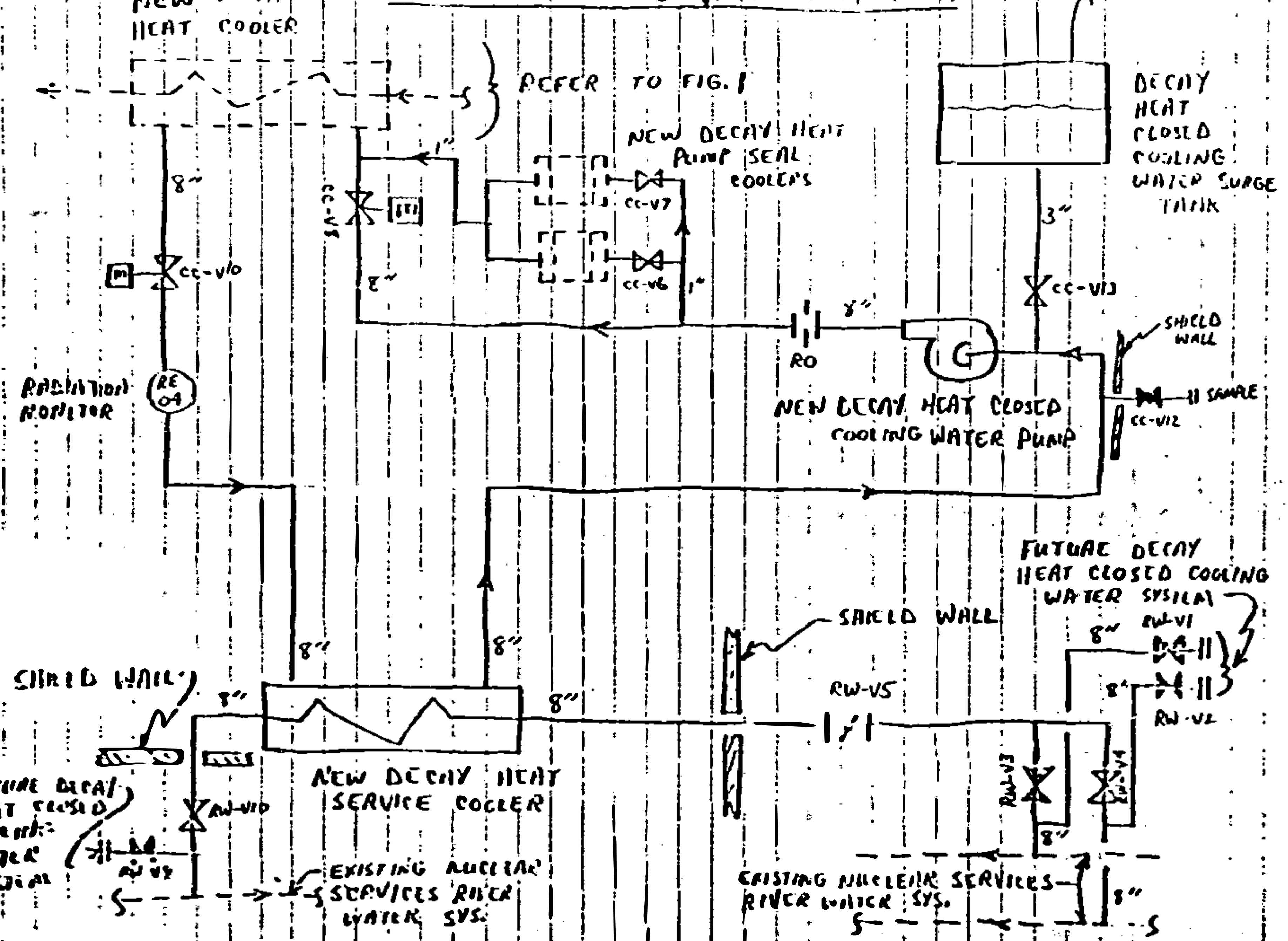


FIGURE 1

Dotted lines - existing system

NEW WESTINGHOUSE DECAY HEAT CLOSED COOLING WATER SYSTEM



Dotted lines - existing system or reference to Fig. 1

FIGURE 2

The staff has reviewed the expected system performance of this system and concludes that its heat removal capacity is sufficient for decay heat removal from the TMI-2 reactor.

Mechanical Design Considerations

The skid mounted decay heat removal system consists of components such as heat exchangers, pumps, tanks, valves and piping. The applicable design codes and standards used for the design of these components are provided in the table below.

Table I

APPLICABLE DESIGN CODES OR STANDARDS

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 operating conditions, - to include operating pressure, dead weight, pump vibration, thermal expansion, and maximum anticipated pressure surges.

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.

Design Information: Specific for System Tie-in to Existing DHR Piping, Weld-o-let Connection

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

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

Weld-o-let to DHR existing 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 existing DHR pipe exceeds its yield strength.

All welds and the cut into the existing 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 with relative ease.

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.

Evaluation Conclusion:

We have concluded that the Licensee has specified components designed and fabricated in accordance with acceptable industry codes or standards and will take into account the loads associated, with startup testing, and planned system operation.

The use of components that are in conformance with these criteria provides adequate assurance that structural integrity of the Decay Heat Removal System will be maintained.

Structural

Fuel Handling Building Wall Penetration (Aux. Bldg)

In order to provide the skid mounted decay heat removal system a 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 A69, 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 shovel 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 psig.

253 051

The opening is to be made 21" wide 5'1" high on the outside tapering to 4'-1" at the inside face. One reinforcing bar, #18 is to be cut at each face. Figure 3 shows the front view and the cross-section at the opening.

Procedure

At the time of writing of this report various methods of penetration are being studied. One possibility is using the oxygen lance process. A trial penetration is currently being performed by this method. Drilling through the well is also considered. The final method of the wall penetration will be described upon pending results of the test and consultation with the other experts in the field of construction materials.

Instrumentation for New Decay Heat Removal System

The following identifies the instrumentation to be provided for the third DHR System train. A trailer will be used to provide remote control room operation.

The following instrumentation will be provided:

1. DHR pump suction pressure
2. DHR pump suction temperature (DHR cooler inlet temperature)
3. DHR Cooler inlet pressure
4. DHR Cooler discharge pressure
5. DHR Cooler discharge temperature
6. DHR flowrate

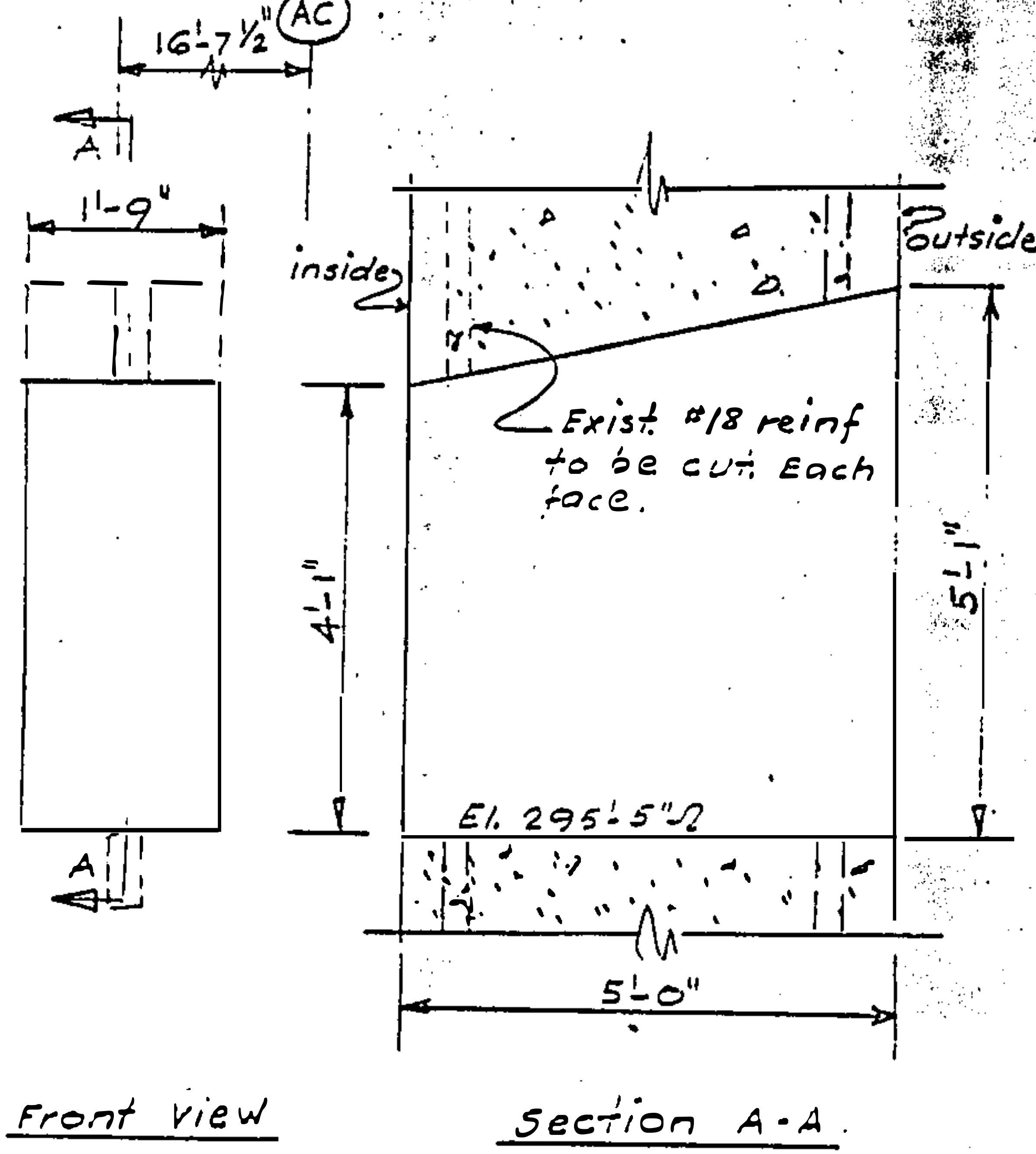


FIGURE 3
Sketch Showing Penetration through Auxiliary Bldg Wall.
4/26/1979.

Redundant measurements and display are provided for each parameter and separated to be consistent with the A or B pump train. Independent power supplies are provided for each train of instrumentation. No automatic control or interlock features are provided.

To the extent practical, the nuclear instrumentation which is safety related has been purchased to Class 1E requirements. The instrumentation selected was based on, to the extent available, the same manufacturer, model, and principle as the instruments used in the existing DHR system.

The staff concludes that the instrumentation and controls to be provided for the skid mounted decay heat system are appropriate for their intended function.

253 054

5. Electrical Power System Modifications

A. General Description

The licensee has identified a number of modifications to the Unit 2 power system in order to accommodate a loss of offsite power. Backup power to the two existing safety related load groups (designated Red and Green) is supplied by the two existing Class IE diesel generators. No loading changes have been proposed for these busses and the staff position is that all new loads be powered from other busses leaving the original Class IE system intact. The staff has further stipulated that the loading of these Class IE busses be on a "manual only" basis due to the time available for operator action following any perterbation and the complex nature of trying to coordinate which loads are needed and which would be detrimental if actuated for each potential plant mode of cooling. Approved written emergency procedures are required to cover the above contingencies. We find the above described modifications to the existing Class IE system to be acceptable.

For all electrical loads that previously did not require loss-of-offsite power back-up protection and all newly added loads which now require this protection, the licensee is installing two new diesel generators. These diesel generators will be assigned to busses 2-3 (breaker position 3-3) and 2-4 (breaker position 4-13).

253 055

These load groups are now designated "Gray" and "White" respectively. These diesels are self-contained skid-mounted outdoor units rated nominally at 2500 kw each. The convention for bus assignment has been odd numbers for the A loop and even numbers for the B loop and this has been retained for the modifications. We have required that the new load groups be maintained independent and that no interlocks shall be incorporated into the power system design that couples the load groups in any manner.

Each new diesel will have its own switchgear, control battery, and fuel tank installed adjacent to it. The switchgear will be connected (via underground cabling) to busses 2-3 and 2-4 (gray and white respectively) through the existing circuit breakers (previously used for condensate booster pumps Co-p-2A and Co-p-2B) with modified relaying to accommodate the power sources. Loss of offsite power will be detected by new undervoltage schemes which will automatically start the diesels. The breakers at busses 2-3 and 2-4 will be normally closed and the new diesel generator circuit breakers will be automatically closed when the units have attained rated speed and voltage. Motor loads are automatically tripped upon loss of power as part of the existing design. The staff requires that these busses be manually loaded by approved written procedure. These actions can be performed from the control room. The basis for this requirement is the time available for operator action and the different loads required for the various potential plant conditions. The licensee's design is in conformance with this requirement and is acceptable.

Busses 2-3 and 2-4 have existing bus transfer schemes that allow continuity of power supply, should one of the two offsite power circuits through the auxiliary transformers be lost, by fast-transfer to the other transformer. This scheme is to be left intact and the new undervoltage detection schemes will have a 10 second delay to accommodate the fast transfer if one circuit remains available. We find the above modifications to be acceptable.

We will require that testing requirement of the Gray and White diesel generators and their associated D.C. control power supplies (batteries) be comparable to those that now apply to the Class IE diesel generators and batteries.

B. Steam Generator A and B Short Term Solid Water Operation

The 900 horsepower condensate pumps are the single largest loads that will receive diesel generator back-up power source protection. The circulating water pumps rated at 2250 horsepower each have placed an additional restraint on the power system. This size load is too large for any of the four diesel generators now on site (red, green, gray and white). The licensee has proposed to provide a 13.2 kv line from the Middletown Substation to accommodate the motor starting requirements of these large loads. This line will have the capacity to start a second circulating water pump while supplying power to the first. Cooling system requirements are fulfilled by one pump operation.

The new 13.2 kv offsite transmission line will originate at the Middletown Junction Substation. This substation has a 230 kv and a 115 kv bus tied together by four auto-transformers. This modification utilizes auto-transformer no. 2 and connects to the 13.2 kv tertiary winding rated at 25 MVA. The group operated air break switches on the 230 kv side will be opened to isolate the transfer from a direct connection to the 230 kv system. There is a spare 230 kv line that terminates on the tower at the entrance to the Middletown switchyard on one end and terminates on the tower at the entrance to the Three Mile Island Switchyard on the other end. This line shares a double-circuit line of transmission towers with an existing 230 kv line to the plant.

At the Middletown Junction Substation, a short line section from the existing capacitor breaker on the 13.2 kv tertiary of the no. 2 transformer back to the spare 230 kv conductors must be constructed. The capacitor bank will be disconnected. The necessary additional relaying will be provided for this capacitor breaker to protect this new feed.

At the Three Mile Island Substation, a 13.2 kv underground cable supply from the spare 230 kv conductors will be run around the southern side of the natural draft cooling tower. This underground portion of the cable run protects the line from any of the other incoming lines falling on it. Once around the cooling tower, the line goes overhead for one span and terminates on a 10 MVA 13.2/4.16 kv transformer. A spare 10 MVA transformer will also be in place with manual switching capability for energization should the first one fail. These transformers are located adjacent to the circulating water pump house and the electrical switchgear bay. The 4 kv portion of the line ties into breaker cubicle 5-4 on bus 2-5 located within the switchgear bay. This breaker was originally used to supply power to circulating water pump CW-P-1E. This breaker will have modified relaying to accommodate the new power source. We have required this alternate power source scheme to be "manual-only" by approved written procedure. The bus must be cleared of all connections following loss of offsite power prior to reenergization and subsequent loading of the circulating water pump. There is a normally open bus tie between busses 2-5 and 2-6. Closing this bus tie gives access to five of the original six pumps.

We will require that the 13.2 kv line be tested weekly by energizing busses 2-5 and 2-7 for a short time interval to assure continued functional capability. There are no phase angle differences between the existing system and this new line so that the connection may be made on a live bus followed by tripping circuit breaker 23-52 to prevent tieing the 115 kv system to the 230 kv system through the plant distribution system. We further require that circuit breakers T-56-2 and T-73-2 be verified open on a daily basis.

1) the application is limited to the circulating water pumps.

2) these pumps are only required for the time window necessary to complete the modifications to steam generators A and B for long term core cooling.

3) the time required to have the 13.2 kv line operational is consistent with the requirement to provide back-up power as soon as practicable.

4) alternative proposals such as the use of diesel generators were not considered feasible due to the size requirements for starting 2250 horsepower loads and indoor service would require a new building that could not be built in time based upon existing schedules.

5) the new line has been isolated to the extent practicable from the 230 kv system and therefore may survive a local 230 kv system disturbance.

6) given a total grid blackout, there are six combustion turbines in the close proximity of Three Mile Island. These units are rated 23 MVA each and have black-start remote supervisory control. The system dispatcher in York has supervisory control of this system with the one exception that the system dispatcher in Lebanon would have to be consulted as to the position of certain intervening circuit breakers.

7) transmission line tower failure could not redner the 13.2 kv and the 230 kv systems inoperable.

253 060

We find the addition of the 13.2 kv transmission line as a backup power source to be acceptable given the operational needs and time restraints present at Three Mile Island Unit 2.

C. Steam Generator A Modifications

These modifications provide a new 700 horsepower high pressure pump for circulating water through the secondary side of the A steam generator. This new loop is in cooled by the Nuclear Services River Water System and the Nuclear Services River Water Purges. The new high pressure pump will be powered from bus 2-3 (Gray), the Nuclear Services River Water Pumps are powered from the existing Class 1E power system. We find the power source allocations to be consistent with the separation of the A and B steam generator modifications, to capable of supplying the practical requirements of the system and to therefore be acceptable.

D. Steam Generator B Modifications

These modifications provide a new 700 horsepower high pressure pump for circulating cooling water through the secondary side of the B steam generator. This new loop is in turn cooled by the Secondary Services Closed Cooling Water Systems which utilizes the Secondary Closed Cooling Water Pumps. The final cooling loop uses the Nuclear Services River Water Pumps for circulation. The new high pressure pump will be powered from bus 2-4 (white), the secondary service closed cooling water pumps SC-P-1B and SC-P-1C are powered from 480 motor control center (MCC) 2-41B which in turn connects to 480 volt bus 2-41 and through a 4160/480 volt to bus 2-4. This arrangement allows a back-up pump (one pump operation meets cooling system requirements) and is part of the white power system. The Nuclear Services River Water pumps are existing loads on the Class 1E diesels. We find the above power assignments to be consistent with the separation requirements of keeping the B steam generator cooling system on the even numbered separate busses, to be capable of supplying the functional requirements of the system and to therefore be acceptable.

E. Skid-Mounted Decay Heat Removal System

This new system will have 480 volt motor operated 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 (Gray and White) in a manner that assures isolation capability given a single power source failure. The remaining 480 volt motor operated valves will be arranged on a "per loop" basis to allow selection of either of the two new 4160 volt 400 horsepower pumps. These valves and the associated pumps will be powered from separate busses (Gray and White) to assure system function given a single power source failure.

Two new 480 volt motor control centers will also be provided. All electric powered equipment (valves, pumps, motor control centers and cabling) will be Class 1E system quality. Because of the low decay heat levels, sufficient time is available for manual operator action assuming loss of power or equipment.

253 0[illegible]

The secondary cooling loop for the DHR heat exchanger includes an additional 250 horsepower pump connected to 480 volt bus 2-44. All associated motor operated valves will receive normal power from the white power system. Should the white power system fail, backup power can be provided from the gray system by closing the normally open bus tie breaker between buses 2-44 and 2-34. We require this tie breaker be racked-out at all times and only closed upon the failure of the white power system by approved written procedure.

This system will not be used concurrently with the steam generator cooling modes describes above and therefore does not affect diesel generator capacity. We find the electrical power aspects of this design as described above to be acceptable.

F. Reactor Coolant Pressure Control System

All electrical equipment and instrumentation required to operate the system are powered from the gray and white power systems. The charging pumps (A and B) are rated 100 horsepower and are powered from 480 volt motor control centers 2-32A (gray) and 2-42A (white) respectively. The charging water storage tank heater is rated at 100 kw and will be powered from bus 2-45. There are a number of small loads associated with this system that have not been assigned power sources.

The system will be automated as soon as possible but this may not occur before initial operation. A motor operated isolation valve will be provided to automatically close when the water level in the tank nearest the reactor coolant system is approximately one third full. This is to preclude the introduction of nitrogen into the reactor coolant system.

6. Quality Assurance Program for TMI 2 System Modifications

The NRC Technical Review Group has reviewed and evaluated the Quality Assurance (QA) Program of GPU/MetEd and of their major subcontractors for the TMI-2 system modifications. These QA Programs recognize the uniqueness of the TMI-2 plant condition and balance the schedule urgency for completion of system modifications against the extent of application of traditional QA program practices consistent with maintaining assurance that specified system requirements are met.

The GPU/MetEd QA Program has been specifically tailored for the TMI-2 system modifications. The Program will apply QA criteria of 10 CFR Appendix B commensurate with the specified system requirements and will be compatible with the MetEd Operations QA Program previously accepted by NRC. GPU/MetEd has established a QA organization at the TMI-2 site specifically responsible for the system modification QA activities. This staff is experienced in all QA disciplines and associated technical fields, including mechanical, electrical and civil engineering as well as welding and non-desctructive examination. The GPU/MetEd QA Manager and QA engineers were brought in from the Forked River facility. GPU/MetEd is the lead responsible QA organization for the TMI-2 system modification program. Their QA Program will provide surveillance over the activities of their subcontractors, including Westinghouse and Burns & Roe. Westinghouse has established a QA Program to control QA activities associated with design, procurement and vendor component fabrication related to the skid mounted Backup DHR system modification. Westinghouse has organized an experienced QA staff specifically responsible for this task. Westinghouse will also provide system installation and pre-operational testing procedures and on site technical supervision which will be conducted subject to the GPU/MetEd QA Program.

Burns & Roe is responsible for establishing designs for the other TMI-2 system modifications, including the designs for the Reactor Coolant Pressure Control System, the A and B Steam Generator Cooling System.

Burns and Roe is implementing design control QA practices which assure that appropriate quality standards are specified and included in design documents that provide for verifying or checking the adequacy of design and that will control design changes. The GPU/MetEd QA Program will provide QA surveillance for the follow-on activities for these systems, including procurement, fabrication, installation and testing.

The NRC Regional Office of Inspection and Enforcement has available qualified QA staff experienced in mechanical, electrical and civil engineering disciplines and welding and non-destructive examination to provide surveillance of system modification activities at TMI 2 site and at equipment vendor facilities as necessary.

Based on our review and evaluation of the QA practices, controls, and organization of GPU/MetEd and their major subcontractors, we conclude that these QA Programs will assure meeting the criteria of 10 CFR 50 Appendix B commensurate with the TMI-2 system modification requirements and are acceptable.

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Availability	Publicly Available
Microfiche Address	00253:002-00253:067
Sensitivity	Non-Sensitive
Package Number	7906280173A