GPU NUCLEAR

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September 30, 1998
1920-98-20527

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
Generic Letter (GL) 96-06, Response to Request for Additional Information

Attached is the GPU Nuclear response to the NRC's July 13, 1998 request for additional information regarding Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions."

If you have any questions or comments on this matter, please contact Bob Knight, TMI Nuclear Safety and Licensing at (717) 948-8554.

Sincerely,

James W. Langenbach

James W. Langenbach
Vice President and Director, TMI

MRK
Attachment

cc: Administrator, NRC Region I
TMI Senior NRC Resident Inspector
TMI Senior NRC Project Manager
File# 96078

A072 1/1

NRC Questions Related to Potential Penetration Overpressization

In its submittal of June 2, 1997, the licensee identified eleven piping segments that are susceptible to thermally induced over pressure. The licensee indicated that the Reactor Coolant Pump Seal Return Line was demonstrated to be acceptable by analysis. The licensee also committed to install pressure protection for the remaining ten piping segments. In order to complete its review of the licensee's response to the GL, the staff needs additional information regarding the Reactor Coolant Pump Seal Return Line piping segment which was found acceptable by analysis. Please provide the following information:

Question 1

Provide the applicable design criteria for the piping and the valves. Include the required load combinations.

Response

The segment referenced is the 4" Reactor Coolant Pump Seal Return Line pipe that penetrates the Reactor Building through penetration 329. The inboard valve is MU-V-0025 and the outboard valve is MU-V-0026. Valve number MU-V-0233 is also contained within the penetration boundary. The pipe is a 4" schedule 120, A312 type 304 stainless steel. The design code for the pipe segment is USAS B31.1-1967. Fabrication, erection, and test is in accordance with USAS B31.7 Draft February 1968, including June 1968 Errata. MU-V-0025 is a 1500 lb globe valve that is designed, fabricated and tested to the ASME Boiler and Pressure Vessel Code Section I, USAS B31.7, USAS B31.1, USAS B16.5, USAS B1.1, USAS B18.2.2, MSS SP 25 and MSS SP 61. MU-V-0026 is a 1500 lb gate valve that is designed, fabricated and tested to USAS B31.7 Draft February 1968, USAS B31.1, USAS B16.5 and USAS B16.10. MU-V-0233 is a 1500 lb globe valve that is designed, fabricated and tested to USAS B31.1-1967 and USAS B31.7 Draft February 1968. The pipe segments are classified seismic class I. The following load combinations were considered:

Pressure + Dead weight $\leq 1.0\ S_h$
Deadweight + Seismic OBE + Pressure $\leq 1.2\ S_h$
Deadweight + Seismic SSE + Pressure $\leq 1.8\ S_h$
Thermal expansion $\leq S_a$ where $S_a = f\,(1.25\ S_c + .25\ S_h)$

Where:
S_h = B31.1 allowable stress for the hot condition
S_c = B31.1 allowable stress for the cold condition
f = function of cycles ($f = 1$ for TMI-1)

The design pressure is 2500 psig and the design temperature is 300°F.

Question 2

Provide a drawing of the piping run between the isolation valves. Include the lengths and thickness of the piping segments and the type and thickness of the insulation.

Response

See Figure 1

Question 3

Provide the maximum-calculated temperature and pressure for the pipe run. Describe in detail the method used to calculate these pressure and temperature values. This should include a discussion of the heat transfer model used in the analysis and the basis for the heat transfer coefficients used in the analysis.

Response

The maximum calculated temperature for the fluid in the piping run is 220.5°F at a pressure of 404 psi (Reference 1). To evaluate the temperature rise of the piping segments located within the Reactor Building, the GOTHIC computer model was used.

The pipe segments were modeled as cylindrical heat conductors and exposed to the bounding EQ temperature and pressure profiles for TMI-1; credit was not taken for pipe insulation. The convective heat transfer coefficient for the end of the segment inside containment was modeled by combining both the condensation and convection heat transfer values. The Uchida correlation was used to assess the condensation heat transfer while the convection was assessed using the following correlation:

$$h = (k/L)\mathrm{Max}(0.53Ra^{0.25}, 0.126Ra^{1/3})$$

Where:
k = Thermal Conductivity
L = Outer Diameter of pipe
Ra = Rayleigh Number

The Uchida condensation correlation used in the heat transfer calculation is widely recognized as an appropriate heat transfer correlation when in condensing mode.

The convection correlation represents natural convection over a horizontal cylinder. This correlation was selected to be representative of the physical situation in the plant.

The internal convection term is treated as having a Nusselt number of 1.0, which is equivalent to a pure conduction heat transfer mechanism. The basis for this is described as follows. During the design basis LOCA, each piping section of concern is being symmetrically heated. Therefore, the entire surface area of the pipe is at the same temperature and the heat transfer is in the radial direction. Because of the symmetrical heating of the fluid, it was felt that there was insufficient buoyancy to drive a natural convection process. This assumption was evaluated and determined to be conservative in comparison with a natural convection correlation as described below.

From the Handbook of Heat Transfer Fundamentals (Rohsenhow, Hartnett and Ganic, Second Edition, 1985, page 6-67) a correlation was obtained for quasi-steady convection in an enclosure following a step change in wall temperature. The quasi-steady regime becomes established after an initial transient in which the heat flow is dominated by conduction. The time period required to establish the natural convection mode of heat transfer is not specified in this reference. To verify the peak temperatures calculated using the conduction method described above, an alternate method was applied using guidance from NAI (GOTHIC Code developer), and the aforementioned correlation for natural convection inside a cylinder. One of the piping systems under evaluation was modeled as a control volume with a constant low pressure (100 psia) boundary condition. This boundary condition is needed because GOTHIC cannot be used to calculate the high internal pressures (potentially several thousand psi) expected for this analysis of the volume. The boundary condition prevents the pressure from becoming too large. This approach will slightly under-predict the temperature rise because the water specific heat decreases slightly at high pressure and low temperature. It is estimated that the error in temperature rise due to this neglected effect is less than 1.0%. Therefore the conduction method was proven to be conservative and thought to be appropriate. The peak temperature using the alternate method was slightly lower than the temperature calculated using the conduction method.

The heating of the fluid within the containment will cause the system pressure to rise. The pressure rise was not evaluated using the GOTHIC code because of the pressure limitation previously described. In the piping systems under evaluation, there are two isolation valves (one inside and one outside containment); each is closed. Only that portion of the piping inside containment is heated, however the fluid outside containment is compressed by the expanding fluid that is trapped within the containment. The resulting pressure rise will cause the piping material to expand under stress, and the piping within the containment will also expand thermally. These phenomena were coupled in a solution method used to establish the system pressure that results from the temperature rise.

The approach used was to first assume that the system is rigid and when heated the fluid specific volume remains constant. This will provide an initial pressure value to be assumed in the analysis. Using this initial pressure, the pipe volume expands as a result of mechanical and thermal expansion of the materials and the volume used in the pressure calculation is modified. The fluid pressure is recalculated and again the pipe volume is modified. This process is repeated until there is no change.

NRC Questions Related to Potential Waterhammer and Two-Phase Flow

GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996, included a request for licensees to evaluate cooling water systems that serve containment air coolers to assure that they are not vulnerable to waterhammer or two-phase flow conditions. The licensee provided its assessment of the waterhammer and two-phase flow issues for TMI-1 in letters dated February 14, 1997 and February 13, 1998. The licensee has determined that since a minimal amount of voiding will occur (<1%) in the reactor building emergency cooling (RBEC) system during the worst-case scenario, waterhammer and two-phase flow do not present a concern for TMI-1. The RBEC system provides cooling water for the containment air cooling units. In order to assess the licensee's resolution of these issues, the following additional information is requested:

Question 4

Provide a detailed description of the "worst case" scenarios for waterhammer and two-phase flow, taking into consideration the complete range of event possibilities, system configurations, and parameters. For example, all temperatures, pressures, flow rates, load combinations, and potential component failures should be considered. Additional two-phase flow considerations include:

- *the effects of void fraction on flow balance and heat transfer,*
- *the consequences of steam formation, transport, and accumulation,*
- *cavitation, resonance, and fatigue effects; and*
- *erosion considerations*

Response

Initially the system is assumed to be at 130°F and 130 psig. Upon Loss of Offsite Power (LOOP), the RBEC fan coolers will trip and start coasting down. The heated containment atmosphere will transfer heat to the fan cooling coils. This heat transfer will cause the water contained in the cooling coils to heat and expand. Expansion will cause the pressure in the coils to rise to the relief valve (RR-V-0011) setpoint. The relief valve will lift and is assumed to stick open. The stuck open valve will cause the system pressure to decrease due to the loss of inventory. Cooler inlet valve (RR-V-0003) is normally open by procedure. If the RR-V-0003 valves are not already open, they and the discharge (RR-V-0004) valves, as well as the pump discharge valves (RR-V-0001), will begin to open when the first block is loaded on the diesel generator. Also the cooler fan will restart on slow speed as part of the first block loading. The RBEC System cooling water pumps, (RR-P-0001A and B) will start when block two is loaded on the diesel generators. The starting of the pumps will terminate the pressure loss and collapse the small amount of voiding. The warm water slug contained in the coolers will be swept out by the pump flow and the coolers will operate as designed. Because the warm water slug is transient, the effects of cavitation, resonance, fatigue, and erosion are insignificant.

In this worst case scenario for the TMI-1 RBEC Coolers, the voiding in the upper 1/5 of the coolers was approximately 0.51% by volume; the overall void fraction was 0.12% by volume. The assumptions and a description of the model is as follows:

Initial conditions:

- Initial system pressure is at 130 psi.

  Basis: Our analysis found that a higher initial pressure will lead to an earlier relief valve, RR-V-0011, actuation. Assuming a failure of the relief valve (explained later) this actuation will cause a loss of system inventory and a subsequent pressure decrease. Other cases were run at lower initial pressures (0 psig, 55 psig) but resulted in a smaller void fraction or no voiding at all. This was due to the ability of the water to expand and pressurize the system prior to lifting the relief valve and loosing system mass. The delay in pressurizing the system to the relief valve setpoint lets less mass escape the system prior to the actuation of the coolant pumps.

- The RBEC cooler relief valve (RR-V-0011) fails open upon initial actuation.

  Basis: The relief valve becomes an active component when required to actuate. For this case the relief valve is assumed to fail upon initial actuation as the single failure. The high initial system pressure combined with the relief valve failure results in a large amount of system mass exiting the coolers. The mass decrease results in a pressure drop to saturation which results in the voiding. A correctly operating relief valve would maintain the elevated system pressure which would prevent voiding.

- The cooler discharge valve (RR-V-0004) is initially closed.

  Basis: If the RR-V-0004 valve is not initially closed the RR-V-0006 valve will control the system pressure to greater or equal to 55 psig and the initial 130 psig pressure could not occur. The initial position of this valve has very little impact due to the pressure loss caused by the stuck open relief valve, RR-V-0011. Other cases were run with the valve initially in the open and closed position.

- The Nuclear Services Closed Cycle Cooling Water (NS) cross tie is closed and leak tight.

  Basis: The NS cross tie is assumed closed to permit operation with or without the cross tie. A closed cross tie does not permit makeup to the fan cooling system from the NS system. This is conservative because the NS system will not keep the RR system pressurized. The valves are assumed leak tight so they do not provide pressure relief and will not leak makeup water. The system also physically contains isolation and check valves which would prevent its use as a pressure relief path.

- RR-V-0006 controls system pressure to greater than or equal to 55 psig.

  Basis: The RR-V-0006 valve is designed to provide a 55 psig back pressure for the system. The valve is connected to both the instrument air system and the two hour air system (safety grade) to provide reasonable assurance of operation. For this case the valve is assumed to operate properly with the RR-V-0011 as the single failure. Other cases were run that assume RR-V-0006 fails open.

- Initial cooler water temperature is 130°F.

  Basis: TMI-1 Technical Specification 3.17.1 states "Primary containment average air temperature above Elev. 320 feet shall not exceed 130°F and average air temperature below Elev. 320 feet shall not exceed 120°F." The coolers at TMI-1 are located on the 281 foot elevation. According to the technical specifications, the maximum air temperature is permitted to be 120°F. Our analysis conservatively assumes that the cooler water is in thermal equilibrium with the maximum technical specification bulk containment temperature for elevations above 320 feet.

- RB temperature profile follows the bounding RB EQ temperature profile.

  Basis: The EQ temperature profile provides the bounding containment conditions.

- RR-V-0011 is assumed lift at 145.5 psig.

  Basis: The valve is currently set at 150 psig. The valves are tested as part of our IST program and are set to lift within 3% of their setpoint. This makes the lower limit 145.5 psig. A higher setpoint would delay the opening and assumed failure of the valve and the subsequent loss of pressure which is less conservative.

- The RBEC fans are initially operating at the maximum design flow rate.

  Basis: The fans are operating as designed. Sensitivity studies were run to determine the effect of higher system flows. The difference was not noticeable.

- The fans trip with the loss of power and coast down until powered by the diesels which will restart the fans on low speed.

  Basis: The fans are assumed to coast down in 30 seconds. They will trip during the LOOP and are designed to be repowered by the diesels at low speed. A sensitivity case was run to determine the effect of this variable and there was no noticeable change in the conclusion regarding void fraction.

- The Accident is concurrent with the LOOP.

  Basis: This was determined to be the worst case scenario. The LOCA provides the energy necessary to void the cooling coils. The LOOP delays the start of the cooling pumps and provides heat up time required to cause voiding.

Other scenarios were run making different assumptions for the initial system pressure, RR-V-0003 and RR-V-0004 valve positions, NS cross tie, and RR-V-0006 and RR-V-0011 failures.

The GOTHIC computer code (version 5.0e) is used to provide the modeling of the cooling coil and its response to elevated containment temperatures prior to RR pump flow. The approach taken is to use the qualified fan cooler model provided in the computer code and link it with a network of volumes and heat structures designed to represent the cooling water side of the fan coolers. The link is provided using GOTHIC control variables in a feedback loop. Specifically, the portion of the volume heat structure network which represents the cooling coil portion of the river water system is used to drive the fan cooler tube side inlet temperature and convective heat transfer coefficients. From these inputs the fan cooler model calculated the heat transfer rate to the cooling fluid. The calculated heat transfer value is then fed back to the heat structure that represents the cooling coils and heats the fluid in the coil volume. See Figure 2.

The remainder of the river water piping within the containment is modeled as volumes and heat structures. These additional volumes are linked together with the cooling coils by junctions. The heat structures are connected to the containment volume from which heat is absorbed. A relief valve is provided at the outlet to the system upstream of RR-V-0004 representing RR-V-0011 and a valve is provided at the cooler outlet to represent RR-V-0004.

Question 5

Describe and justify all assumptions and input parameters (including those used in any computer codes) that were used in the waterhammer and two-phase flow analysis. Confirm that these assumptions and input parameters are conservative and are consistent with the existing design and licensing basis of the plant. Any exceptions should be explained and justified.

Response

See Question 4 for assumptions and input parameters. The input parameters and assumptions used were chosen to maximize potential for voiding in the system. The initial system pressure and temperature were conservatively elevated to maximize the fluid loss through the relief valve and minimize the heating required to create voids. The containment environment was chosen to bound the MSLB and LBLOCA containment conditions. The impact of various initial valve positions was considered. The impact of various fan flow rates was studied. Piping outside of containment was conservatively not credited in the analysis because it could reduce the system pressure by allowing for the expansion of hot fluid. In our opinion the fan cooler model is a conservative representation and the plant design and licensing basis is met.

Question 6

Confirm that the waterhammer and two-phase flow analysis included a complete failure modes and effects analysis (FEMA) for all components (including electrical and pneumatic failures) that could impact performance of the cooling water system and confirm that the FEMA is documented and available for review, or explain why a complete and fully documented FEMA was not performed.

Response

No formal documented FEMA was performed for this analysis. The failure modes were compared to FSAR Table 6.3-1 "Single Failure Analysis – Reactor Building Emergency Cooling System" and no unreviewed configurations were found. The FSAR analysis considers the following:

1. The fan unit fails to start or fails to start at low speed.
   A sensitivity case was run to determine the effect of higher containment atmosphere flow rates through the coolers. This study determined that there is no change in the results due to a higher fan flow rate. If the fan failed to restart at all there also is very little change because the fan will still coast down from full speed over 30 seconds.

2. Cooling water supply and return lines or emergency cooling coils ruptures.
   A rupture or major leak of the cooling supply and return lines or emergency cooling coils is capable of being detected and isolated by the Control Room. A rupture or major leak would render the cooling unit inoperable so the effects of waterhammer or two-phase flow are inconsequential.

3. Supply or return valve fails to operate (RR-V-0001A/B, RR-V-0003A/B/C, RR-V-0004A/B/C/D).
   A single failure of a RR-V-0001 valve would be identical to a pump failure. The valve would isolate one pump and the other pump is capable of delivering the required system flow. A single failure of a RR-V-0003 valve would close the inlet of the cooler and render that particular cooler inoperable (failed open has no effect). Cooling will still be provided by the remaining units. If the valve were fixed, waterhammer and two phase flow would not be a concern because the system pressure in the cooler would have cycled up to and remained at the relief valve setpoint which is well above the saturation pressure for the maximum containment temperature. A single failure of an RR-V-0004 valve would not be of concern because the system will not void before the coolant pumps are activated and providing an overpressure (failed open has no effect, RR-V-0006 would then control cooler pressure to greater than or equal to 55 psig). This overpressure is sufficient to prevent void formation because the corresponding saturation temperature is greater than the maximum containment temperature.

4. One Reactor Building Emergency Cooling Pump Fails to Start.
   The other Reactor Building Emergency Cooling Water pump starts automatically and is capable of supplying full system flow.

5. <u>Back pressure valve in return water line from emergency cooling coils fails closed (RR-V-0006).</u>
   The system pressure would be maintained at the pump discharge dead head pressure. This pressure is well above the saturation pressure for the system under the accident conditions. The event and void formation would terminate when the pressure is supplied to the system. The back pressure bypass valve (RR-V-0005) is capable of restoring system flow with operator action.

6. <u>One diesel generator supplying power to the handling unit fails to start.</u>
   The remaining diesel generator will start and supply power to one pump and cooling unit train. A second cooler may be switched to the operable diesel bus. This assures that two coolers can remain operable with one diesel is inoperative. The coolers which are not supplied by the diesel will either pressurize due to the operating coolant pump or pressurize due to thermal expansion of the coolers water depending on initial valve positions. In either case the pressure of the coolers is sufficient to prevent or collapse voids.

In addition to the above considerations, a failure of the RBEC fan cooler relief valve RR-V-0011 to close after initial actuation was considered the worst case scenario.

<u>Question 7</u>

Determine the uncertainty in the waterhammer and two-phase flow analyses that have been completed, explain how the uncertainty was determined, and how it was accounted for in the analyses to assure conservative results.

<u>Response</u>

Uncertainty analysis was not performed for this evaluation. The scenario was selected to conservatively represent the design basis of the plant. All assumptions made in the analysis were designed to maximize the potential for void formation. In particular the cooling coil relief valve is assumed to fail open and makeup water from the Nuclear Services surge tank is assumed to be isolated. The initial pressure in the system is assumed to be equivalent to that associated with operation to cause an early relief valve opening maximizing inventory lost. The initial system temperature is assumed to be 10°F greater than the Technical Specification containment temperature for the elevation. The accident environment selected is for a bounding event. These assumptions will maximize the coil fluid heating, the fluid lost from the system and eliminate any makeup to the system. Piping located outside the containment is not credited in the analysis. Such piping would help to reduce the system pressure somewhat by allowing the hot fluid inside containment to expand slightly.

<u>Question 8</u>

Provide a complete description of any limitations and non-conservatisms that exist in the fan cooler model that was developed, including use of the GOTHIC computer code in this particular application. Explain how the fan cooler model and use of the GOTHIC code were validated such that conservative results are measured.

Response
It is our opinion that the fan cooler model is an accurate representation and there are no non-conservatisms in the model. The one limitation is that the model cannot be used to assess water hammer events, however, the model is not used to evaluate the impact of water hammer on the system. The GOTHIC model of the fan cooler is used to establish the amount of voiding within the system. For this purpose the GOTHIC code is appropriate.

The GOTHIC containment analysis code is issued through EPRI with a Qualification Report. Within the Qualification Report is an evaluation of Fan Cooler models. To evaluate the models performance data was obtained for several fan cooler units (one of the coils is typical of those installed at Three Mile Island Unit I). The data was taken from tests conducted in a pressure vessel that allowed vapor to be recirculated through the coil. The coils were tested at pressures of approximately 25, 45 and 70 psia with temperatures of approximately 185, 245 and 285F. For all three tests the humidity was set at 100%. When evaluated with the GOTHIC code the TMI fan cooler compared very well with the test data. Based upon these bench mark results we feel that the GOTHIC computer code and the fan cooler model are appropriately used in evaluating the possibility of void formation.

The application of the fan cooler to assess void formation at TMI-1 was developed by GPU Nuclear following the guidance provided by the code vendor (NAI) in a report provided for EPRI ('GOTHIC Analysis of a Containment Fan Cooler Unit and Piping Under LOCA & Loss of Offsite Power Conditions' Prepared by Numerical Applications, Inc.). In addition, NAI was contracted by GPU Nuclear to perform a design verification of the model developed for TMI-1. The conservative implementation of the model is ensured primarily by the bounding assumptions used in the analysis. These are described in the answer to question 7.

Question 9
Depending on the context, 1% voiding can be a substantial amount. Clarify what was meant by the characterization that was made (e.g., <1% mass fraction of vapor in the fluid stream; <1% vapor formed when compared to the total fluid mass of the system). Explain why this amount of voiding does not constitute a waterhammer or two-phase flow concern.

Response
When discussing void fractions in our calculations we are referring to a volume fraction. The worst case analysis demonstrated an overall maximum void or volume fraction of 0.0012 (0.12%) in the fan cooling coils. That is 0.12% of the fan cooler fluid volume contains vapor while the remainder is liquid. The small amount of voiding will not produce water-hammer loads that will endanger the system integrity. The primary reason for our conclusion is that the volume fraction is so small that any collapse would not result in water acceleration sufficient to produce water hammer damage. In addition, the void is surrounded by saturated water and is not stratified making condensation induced water hammer unlikely.

Question 10
Discuss specific system operating parameters and other operating restrictions that must be maintained to assure that the waterhammer and two-phase flow analysis remain valid (e.g.,

pressures, temperatures, surge tank level, system alignment), and explain why it would not be appropriate to establish technical specification requirements to acknowledge the importance of these parameters and operating restrictions. Also, describe and justify use of any non-safety related instrumentation and controls in maintaining these parameters and operating restrictions.

Response
No operating restrictions are required to assure that the analysis remain valid and therefore no technical specification requirements are required. The system has been modeled to assume a variety of valve positions and initial pressures. The valve positions allow for operation of the coolers with the NS cross tie open or closed as well as the cooler inlet and discharge valves open or close or any variety thereof. As stated in our February 13, 1998 submittal the NS cross tie is not required to maintain system pressure or provide makeup flow and therefore no operating restrictions are required. Initial system temperatures are adequately controlled by the RB temperature technical specification. The relief valves are tested and set in accordance with the plant IST program.

Question 11
Implementing measures to assure that waterhammer and two-phase flow will not occur, such as maintaining system alignment requirements and surge tank parameters, is an acceptable approach for addressing these concerns. However all scenarios must be considered to assure that the vulnerability to waterhammer and two-phase flow has been eliminated. Confirm that all scenarios have been considered, including those where the affected containment penetrations are not isolated (if this is a possibility), such that the measures that have been established are adequate to prevent the occurrence of waterhammer and two-phase flow during (and following) all applicable accident scenarios.

Response
No special measures or system configurations are required to prevent waterhammer or two-phase flow in the fan cooling unit. As stated in our February 13, 1998 submittal, the surge tank is not required to prevent this event. All known scenarios for waterhammer and two-phase flow have been considered for the containment coolers as specified in the Generic Letter.

Question 12
Explain and justify all uses of "engineering judgement" that were credited.

Response
The fans are assumed, based upon engineering judgment, to coast down in 30 seconds. The fans will trip as a result of the LOCA conditions or the LOOP. In either case the fans will begin to coast down while the containment atmosphere is at a high density. This heavy atmosphere will tend to slow the fan speed considerably. Discussions with plant personnel indicated that this was a reasonably long period of time for the coast down of the fans. However, in the analysis the fans never reach zero flow since they will receive power from the diesel generators prior to reaching a full stop. In addition, a sensitivity study was done where the fan speed was held constant and there was no noticeable change in the conclusions regarding the void fraction.

The check valves on the discharge of the river water pumps are assumed to seat tightly and do not allow reverse flow while the system is pressurized by heat addition from the accident conditions. This assumption is also based upon engineering judgment. It is reasonable to expect that these valves will not leak a significant amount of fluid. Any minor leakage prior to the pump start would only reduce the amount of fluid leaving through the relief valve. In other words they would provide a portion of the pressure relief.

Question 13
Provide a simplified diagram of the system, showing major components, active components, relative elevations, lengths of piping runs, and the location of any orifices and flow restrictions.

Response
See Figure 3.

Question 14
Describe in detail any plant modifications or procedure changes that have been made or are planned to be made to resolve the waterhammer or two-phase flow issues.

Response
The RBEC system relief valves (RR-V-0011) have been moved outside of containment. This was a result of a flood concern due to failed relief valves as described in GL 96-06 and not to mitigate any waterhammer or two-phase flow concerns. This modification was completed in the fall of 1997 during the 12R outage.

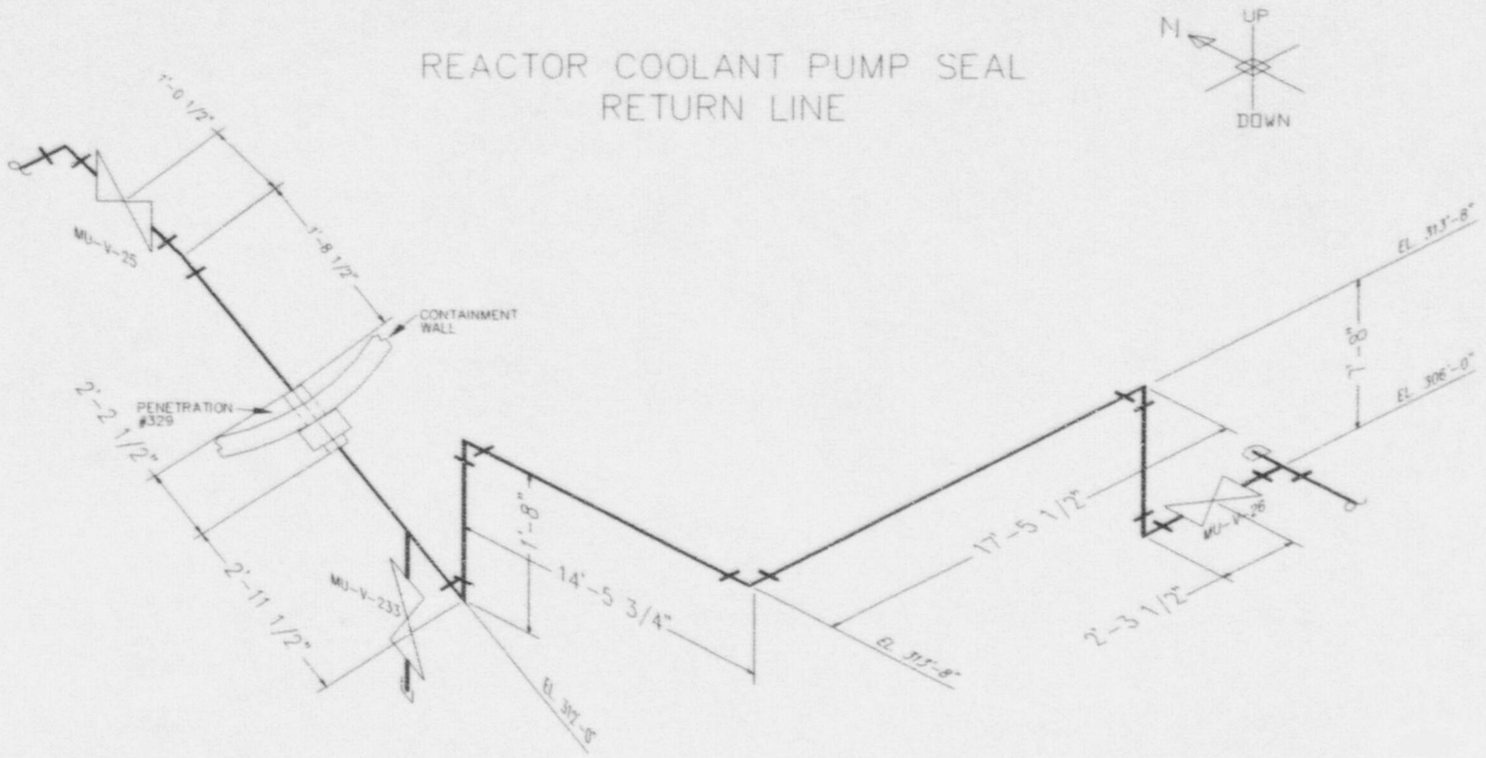
In addition the cooler discharge back pressure control valve (RR-V-0006) was connected to the two hour backup air system. This safety grade air supply will provide additional assurance that the valve will be able to function during a loss of instrument air. This modification was completed in the fall of 1997 during the 12R outage.

No other plant modifications or procedure changes were required to resolve the waterhammer or two-phase flow concerns.

REFERENCES:

1. GPU Nuclear Calculation, C-1101-104-E610-024, Rev. 1, "Pipe Stress Analysis for TMI In Response to GL 96-06," dated September 29, 1998.
2. GPU Nuclear Calculation, C-1101-534-E610-019, Rev. 1, "TMI Fan Cooler Response to GL 96-06 condition with consideration given to RR-V-0003, RR-V-0004, NS-V-0084 and NS-V-0085 Initial Position," dated January 7, 1998.
3. GPU Nuclear Calculation, C-1101-823-E610-010, Rev. 0, "TMI Fan Cooler Response To GL 96-06," dated May 28, 1997.

Figure 1



PIPE INFORMATION

SIZE	4"
SCHEDULE	120
MATERIAL	ASTM A312, TYPE 304
INSULATION	FOR PERSONNEL PROTECTION ONLY. NO CREDIT TAKEN FOR INSULATION IN ANALYSIS.

VALVE INFORMATION

MU-V-25

MANUFACTURER	ROCKWELL
MODEL NO.	K3628(F316)JM
TYPE	GLOBE
BODY MATERIAL	ASTM A182 Gr F316

MU-V-26

MANUFACTURER	ALOYCO
MODEL NO.	N6226-ACC-SP
TYPE	GATE
BODY MATERIAL	ASTM A351 CF8M

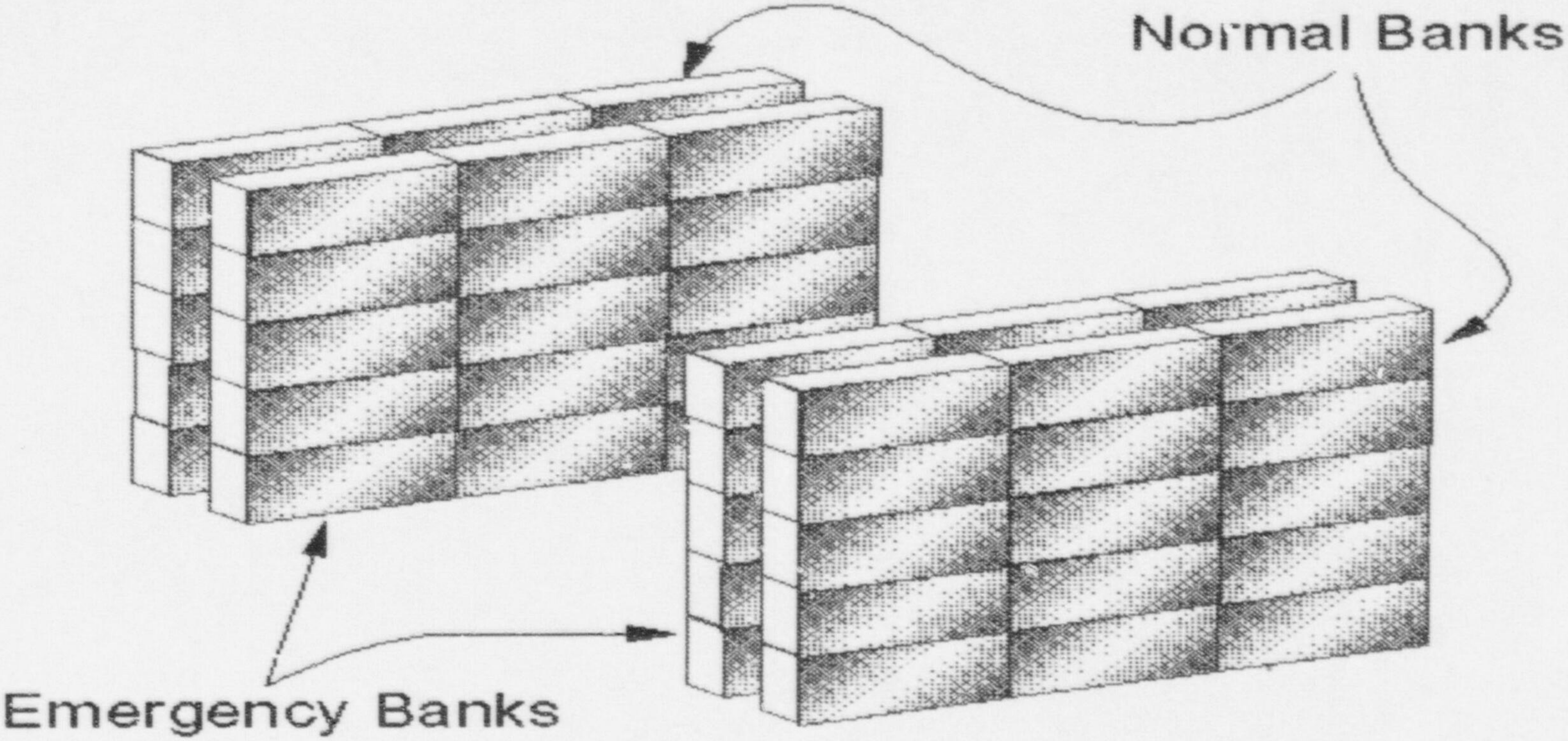
MU-V-233

MANUFACTURER	YARWAY
MODEL NO.	5615B
TYPE	GLOBE
BODY MATERIAL	ASTM A182 F316

CAD FILE NAME: S1A-SKM-1011_S2_R0

Figure 2

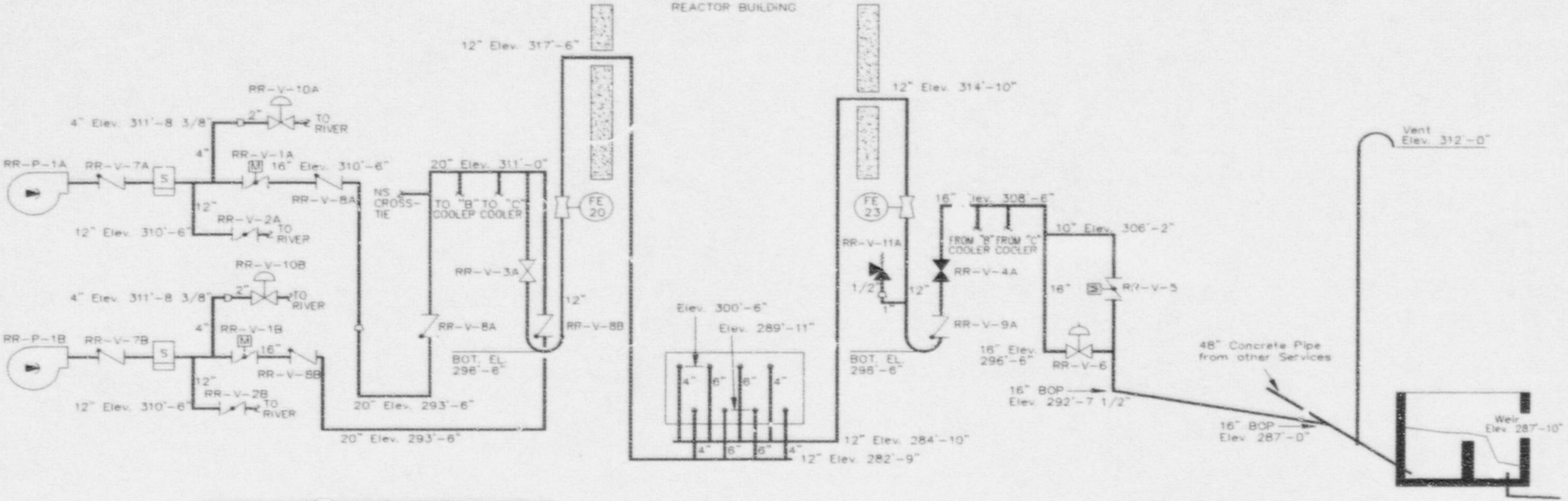
For the GOTHIC model the cooling coils were modeled as a set of five control volumes with a set of corresponding heat structures. The use of five control volumes is a natural division for the modeling since the actual fan cooler is divided into five elevations as is illustrated in the figure below.



As can be seen from this figure the cooler banks are divided into a number of smaller cooling coil units (five high by three wide). For each of the control volumes 6 of these smaller cooling units will define a single control volume within the GOTHIC model.

Figure 3

REACTOR BUILDING EMERGENCY COOLING WATER SYSTEM



	APPROXIMATE PIPE LENGTHS	
1)	PUMP RR-P-1A DISCHARGE TO RR-V-3	1126'
2)	RR-V-3A TO AH-E-1A INLET HEADER	161'
3)	AH-E-1A OUTLET HEADER TO RR-V-4	233'
4)	RR-V-4 TO RR-V-6	34'
5)	RR-V-6 TO MDCT	1212'

CAD FILE NAME: S1A-SKM-1011_S1_R0