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New York Power Authority

Robert J. Barrett
Plant Manager

January 28, 1997
IPN-97-012

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

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Response to NRC Generic Letter 96-06: Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions

Reference: NRC Generic Letter 96-06, T. T. Martin, NRC to Operating Licensees, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996.

Dear Sir:

This letter provides a written summary report stating the conclusions of the evaluation, and corrective actions taken or planned for Indian Point 3 Nuclear Power Plant as requested in reference 1.

The summary report of the evaluations for the Fan Cooler Units is presented in Attachment I and II. As stated in the summary report, the evaluation concludes that containment air cooler cooling water systems are susceptible to waterhammer and two-phase flow conditions during postulated design basis accident conditions. The waterhammer loads, however, have been preliminarily evaluated to be within operable limits. The two phase flow condition is predicted to occur downstream of the FCUs, outside of containment, at the manual isolation valves on the service water return line. The predicted reduction in service water flow will not result in the overall long term heat removal capability of the FCU below their design basis accident heat removal requirement. Therefore, for the Generic Letter 96-06 scenario, there is no challenge to either the service water system or FCU operability.

The summary report for the thermal overpressurization evaluation is presented in Attachment III. For those lines/valves determined to be susceptible to thermally induced overpressurization, operability was confirmed in accordance with Generic Letter 91-18. As stated in the summary report, the evaluation concludes that one line will require administrative controls to have its fluid voided from the piping during normal plant operations. This will resolve concerns about the

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integrity of the containment isolation valves in this line and eliminate the need to credit ASME III, Appendix F for the piping. Further, ten lines have been determined to meet the requirements of ASME III Appendix F, but will require further analysis to establish the actions necessary, if any, for compliance to the criteria stated in the Updated Final Safety Analysis Report.

On January 17, 1997 the Authority identified and informed the NRC of a new scenario, described in Attachment I, that postulated a new single failure which challenges the containment fan coolers and the service water system piping, and may affect the evaluations developed in response to this generic letter. An operability assessment in accordance with Generic Letter 91-18 was performed and concluded that service water temperature at the outlet of the Containment Fan Coolers must be maintained less than or equal to 60° F and Containment temperature less than or equal to 85° F. The results of a more detailed evaluation for this scenario will be submitted by March 3, 1997 as an update to this response or it will be submitted earlier as a Licensee Event Report .

The commitments made by the Authority in this letter are contained in Attachment IV. If you have any questions, please contact Mr. K. Peters at (914) 736-8029.

Very truly yours,

Nathan A. Healey

FOR Robert J. Barrett
Plant Manager
Indian Point 3 Nuclear Power Plant

Attachments I, II, III & V

cc: Mr. Hubert J. Miller
Regional Administrator
Region I
U. S. Nuclear Regulatory Commission
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King of Prussia, Pennsylvania 19406

Mr. George Wunder, Project Manager
Project Directorate I-1
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U.S. Nuclear Regulatory Commission
Resident Inspectors' Office
Indian Point 3 Nuclear Power Plant

State of New York
County of Westchester

Nathan A. Heuberger, being duly sworn, deposes and says:

I am the acting Plant Manager of the Indian Point 3 Nuclear Power Plant of which the Power Authority of the State of New York is the owner and operator under Facility Operating License DPR-64. I have read the foregoing "Response to the NRC Generic Letter 96-06" and know the contents thereof; and that the statements and matters set forth therein are true and correct to the best of my knowledge, information and belief.

Nathan A. Heuberger

Nathan A. Heuberger

Subscribed and sworn to before me
this 28TH day of January 1997.

Barbara Ann Taggart

Notary Public

BARBARA ANN TAGGART
NOTARY PUBLIC, State of New York
No. 4851437
Qualified in Putnam County
Commission Expires Jan. 27, 19 98

Summary Report - Generic Letter 96-06 Evaluation
Potential SW Flashing in FCU Coils During LOCA with Coincident LOOP

SUMMARY / CONCLUSIONS

NYPA contracted with consultants to evaluate this potential transient with respect to the Service Water System and Containment Fan Cooler Unit (FCU) configuration. The Authority's initial operability assessment is described below. Subsequent to the initial operability determination, the consultants and the Authority's staff have done an additional evaluation to bound the waterhammer analysis and to demonstrate through theoretical analysis that the bounding analysis is conservative.

Based on analyses to date, the Authority concludes that IP3 SWS and FCUs will withstand the postulated transients described in Generic Letter 96-06 and continue to be operable, as the initial operability assessment had determined. The reports will be available on site for review.

In addition, a new single failure has been postulated which has the potential of changing the conclusion described above. This information was communicated to the NRC on January 17, 1997. The new scenario is the postulation of a FCU circuit breaker failing to "OPEN", as designed, during a large break loss of coolant accident (LOCA) with a coincident loss of offsite power (LOOP). Therefore, the repowering from the emergency diesel generator (EDG) within 10 seconds with the FCU fan circuit breaker in the "CLOSED" position would start the FCU motor, forcing hot, moist containment atmosphere across the FCU heat exchanger (coils) while the liquid side of the heat exchanger (service water) is stagnant. Thermal energy will be transferred from the containment air to the stagnant liquid inventory of the FCU. If this occurs from time zero to SWS flow, the FCU liquid inventory may be heated to saturated conditions, steam generated, and voids in the service water system formed. Voids could present a potential for waterhammer when the service water system is repowered.

The probability of occurrence for this scenario has been estimated to be 1.67 10^{-9} per year based on the data taken from the IP3 Individual Plant Examination, IP3-RPT-MULT-01539, Rev. 0, June 1994.

An operability determination evaluated the effects of this new scenario. The following operational temperature limits were implemented as compensatory measures based on an operability evaluation:

> FCU service water outlet temperature less than of equal to 60° F and;
> Containment temperature less than or equal to 85° F.

Summary Report - Generic Letter 96-06 Evaluation
Potential SW Flashing in FCU Coils During LOCA with Coincident LOOP

The SWS was designed and is operated to preclude waterhammer for normal operation, start-up and shutdown. Precluding waterhammer for possible accident scenarios was based upon the analyses and events postulated within the industry at the time the plant was designed.

REVIEW & ANALYSIS

For IP3, the postulated transient condition follows a design basis LOCA with a coincident LOOP. The analysis considers the LOCA vapor containment heat profile and bounds the MSLB profile for the duration of the interrupted service water flow. Due to service water pump coastdown being much shorter than FCU fan coastdown, hot, steam-laden containment air would continue to be drawn over the FCU coils, heating the stagnant service water contained in the cooler coils. If service water flow is not restored, the water in the coils will flash to steam. If service water is restored beyond this point in time, the collapse of steam voids in the cooler coils could result in a waterhammer effect, creating hydrodynamic loads potentially challenging the integrity of the FCUs, service water piping and supports, the design basis heat removal capability of the FCUs, and containment integrity.

The evaluation performed assumed as the single failure, the failure of the first service water pump in the engineered safeguards sequence to start, resulting in an additional delay (approximately 1 second) in re-establishing service water cooling to the FCUs.

The evaluation of FCU performance under postulated LOCA plus LOOP was done for the following conditions:

Time (Sec)	Event	Comments
0	Normal Operation	Steady State Conditions
0+	LOCA + LOOP	Design Basis Assumptions
2.0	SW coastdown to zero	SW pump coastdown
30.41	SW (full flow to FCUs)	SW pump startup @ 25.41 sec, 5 sec. to attain full flow
35.0 - 50.0	End of simulation (as applicable)	

The analysis assumed service water is fully re-established at 30.41 seconds into the event (including delays related to Emergency Diesel Generator startup, second SW pump startup and to reach full speed).

Summary Report - Generic Letter 96-06 Evaluation
Potential SW Flashing in FCU Coils During LOCA with Coincident LOOP

The fan coastdown, and the amount of fouling assumed are the two parameters which drive the heat transfer process. With fan coastdown time from inplant testing and the design clean value of 0.00135 fouling factor, the heat transfer for this new scenario is sufficiently reduced at the operational temperature limits such that no meaningful void fraction is predicted to occur in the tube.

BACKGROUND

Westinghouse Nuclear Safety Advisory Letter NSAL 96-003, "Containment Fan Cooler Operation During a Design Basis Accident" identifies the potential for steam flashing in the Containment FCUs during a design basis accident. This issue was reported by a Westinghouse plant whose FCUs were supplied by a closed loop Component Cooling Water (CCW) system. The transient condition occurs under a postulated LOCA with LOOP when the component cooling water pumps and the FCUs trip due to the LOOP. By virtue of the longer coastdown time of the FCU fan than that of the CCW pumps, and FCU fan restart prior to CCW pump restart, the hot, steam laden containment air continues to be drawn through the FCU coils, transferring the accident containment heat to the component cooling water captured in the cooling coils.

This transient scenario was subsequently identified in NRC Information Notice 96-45 and NRC Generic Letter 96-06. The Generic Letter identified the need to consider this potential transient for the more limiting of a design basis loss of coolant accident (LOCA) or main steamline break (MSLB).

IP3 BACKGROUND

The FSAR describes the capability of Containment FCUs to perform their function without raising the exit temperature of the service water to the boiling point and does not describe the event postulated by GL 96-06 which could result in boiling. Therefore, it appears, the potential effects of service water boiling in the FCU coils under the current postulated transient conditions were not considered during the original design and licensing of the Indian Point 3 plant.

The Containment Air Recirculation Cooling and Filtration System was designed to recirculate and cool the containment atmosphere in the event of a LOCA and thereby ensure that the containment pressure will not exceed its design value of 47 psig at 271° F (100% humidity). The design basis accident minimum heat removal capacity is 49 x 10^6 Btu/hr per FCU.

A review of the historical documentation for IP3 has not identified a specific water hammer analysis for the SWS. Additionally, the Atomic Energy Commission (AEC) and Nuclear Regulatory Commission (NRC) did not raise any water hammer concerns for the SWS during the initial licensing of IP3.

**Summary Report - Generic Letter 96-06 Evaluation
Potential SW Flashing in FCU Coils During LOCA with Coincident LOOP**

Coastdown was recognized as a key input parameter for the analysis of the transient condition. Therefore, a test was performed to record measured fan coastdown data. With the plant at power operating conditions, a single fan was tripped while the four remaining fans stayed in operation. The collected data requires subsequent correction to simulate the accident conditions and the simultaneous coastdown of all 5 fans.

Three independent transient analyses were performed. While two of the analyses conclude that no or no significant voiding in the FCU coils will occur, one analysis predicts some voiding. This result is attributed to the longer FCU fan coastdown time used in the analysis. The rate at which the coastdown occurs determines the heat input to the stagnant service water in the FCU coils. Therefore, a longer coastdown time is more conservative for predicting a boiling condition in the coil.

Using the predicted void volume in the FCU coil, a preliminary calculation using LIQT, a code for evaluation of hydraulic transients, resulted in an unbalanced load on the piping system. The structural integrity of selected sections of service water return piping was preliminarily evaluated using conservative bounding maximum loads and the ADLPIPE computer code.

Based on a preliminary assessment of the pipe stresses for this condition using the ANSI B31.1 (1967) code methods stresses at some locations are above design allowables, but the system remains operable.

CONCLUSION

It is concluded, based on evaluations, that the SWS and containment FCUs will remain operable and perform their design accident functions with the single failures considered during the original design and licensing following a LOCA with a coincident LOOP because the service water flow will be fully restored in time to limit the amount of steaming and void formation in the FCU coils such that the resulting waterhammer loads are within acceptable limits.

The most conservative evaluation demonstrated that in the event of a design basis loss-of-coolant accident (LOCA) or main steamline break (MSLB) with a concurrent loss of offsite power (LOOP), service water contained in the Containment FCU coils will result in steam void formation. Re-establishment of service water cooling to the FCUs will occur in time to limit the size of the void and resulting hydrodynamic (waterhammer) loads to operable limits.

Preliminary results indicate that the water hammer loads will not result in the service water piping stresses exceeding its operable limits. A most highly loaded support was evaluated based on the waterhammer loads and determined to be over its nominal rating, but would remain operable based on a lowering of the factor of safety.

Summary Report - Generic Letter 96-06 Evaluation
Potential Degraded FCU Heat Removal Capacity Due to Two-Phase Flow

SUMMARY / CONCLUSIONS

The susceptibility of the SWS to two phase flow conditions in the FCU and FCU discharge piping has been evaluated. Analyses have concluded that two phase flow conditions will occur in the FCU service water return lines, on the downstream side of the manual isolation valves (SWN-44 series) outside containment. This two phase flow condition will result in reduced service water flow to the FCUs. The amount of flow reduction is dependent on FCU coil cleanliness. Fouling factors ranging from perfectly clean to design dirty coils were considered. These analyses conclude that two phase flow conditions would not reduce the capability of the FCUs to meet the design basis heat removal requirement.

IP3 BACKGROUND

The FSAR describes the capability of Containment FCUs to perform their function without raising the exit temperature of the service water to the boiling point. Therefore, it appears, the potential effects of service water boiling in the Fan Cooler unit coils was not a postulated condition during the original design and licensing of the Indian Point 3 plant. Likewise, the potential for reduced FCU service water flow due to a two phase flow condition in the downstream piping was not postulated.

REVIEW & ANALYSIS

Based on NYPA evaluations, the IP3 service water system is susceptible to two-phase flow at the FCU discharge piping at steady state flow conditions during the accident. The following conclusions are derived:

- Service water flow flashes and chokes at the downstream of the SWN-44 throttle valves due to back pressure being lower than saturation pressure

- No flashing occurs upstream of the SWN-44 throttle valves, regardless of the cleanliness of the FCU coils

- The predicted two-phase flow does not result in any adverse loads on the service water piping or pipe supports

- The FCU design basis heat removal capacity (49×10^6 Btu/hr per FCU) is met even with reduced service water flow due to flashing and choked flow

It is concluded that the two phase flow condition in the FCU service water return lines, downstream of the manual isolation valves outside containment, does not result in FCU heat removal capability below the design basis post accident requirement of 49×10^6 Btu/hr. The service water system and the FCUs will continue to be operable within their design basis.

Summary Report - Generic Letter 96-06 Evaluation
Potential Degraded FCU Heat Removal Capacity Due to Two-Phase Flow

CONCLUSIONS

It is concluded, based on evaluations, that the service water system and containment FCUs will remain operable and perform their design accident functions with the single failures considered during the original design and licensing with two phase flow occurring at the manual isolation valves in the service water piping downstream of the FCUs, outside of containment. This two phase flow condition will result in reduced service water flow to the FCUs. The predicted reduction in service water flow will not result in reduced FCU heat removal capability below the design basis accident heat removal requirement. Therefore, there is no challenge to either the service water system or FCU operability.

The evaluation demonstrated that the occurrence of two-phase flow in the FCU service water return lines will not result in degradation of the heat removal capability of the FCUs below that required for design basis accident heat removal.

Summary Report - Generic Letter 96-06 Evaluation
Evaluation of Thermal Overpressurization of Isolated Piping Sections

SUMMARY / CONCLUSIONS

The susceptibility and potential for thermally induced overpressurization of isolated piping sections at Indian Point 3 has been evaluated in response to NRC Generic Letter 96-06 "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions", dated September 30, 1996.

Isolated piping sections requiring more detailed evaluations for thermally induced overpressurization were identified using the following screening criteria:

(1) Thermal pressurization affects only isolated piping sections of water-filled lines penetrating containment. Therefore lines containing gas (N2, air, etc.) were eliminated from the thermal pressurization and containment integrity evaluation.

(2) Water-filled lines with operating temperatures exceeding the maximum ambient temperature as a result of a postulated accident (i.e., **LOCA, HELB, or MSLB)** were eliminated from further evaluation, since those lines are not subjected to trapped fluid expansion during design-basis accident conditions.

(3) Lines penetrating containment and required to be open and to remain open during post-accident conditions were eliminated from further thermal pressurization evaluation, since those lines are not isolated, and therefore, not subjected to overpressurization.

Of the 137 piping penetrations and valving configurations screened, 50 lines were identified as not meeting the above criteria, requiring additional engineering analysis.

As each of these 50 isolated piping sections pass through the containment wall into the PAB pipe penetration area, they were evaluated for the effects of bounding temperature conditions in the PAB pipe penetration area and the bounding post accident containment temperature conditions. This resulted in the consideration of two different accident scenarios. The bounding post accident containment temperature condition is realized following a Loss of Coolant Accident (LOCA). The bounding temperature condition in the PAB pipe penetration area results from the High Energy Line Break (HELB) scenario of a Steam Generator Blowdown line outside of containment. It was conservatively assumed that the HELB scenario generated a Phase A containment isolation signal, trapping fluid between containment isolation valves that close on a Phase A signal. Conversely it was conservatively assumed that no SI signal resulted from the HELB in the PAB for those Containment Isolation valves that are normally closed and open on SI.

Summary Report - Generic Letter 96-06 Evaluation
Evaluation of Thermal Overpressurization of Isolated Piping Sections

Of the 50 lines identified as requiring additional engineering analysis, 34 lines were evaluated as not susceptible to overpressurization during either a LOCA or a HELB, based on the design characteristics of the evaluated systems (e.g. installed relief valves).

The remaining 16 line/valve configurations required further analysis for either one or both of the accident scenarios. Refer to Table 1 for description of each of these line/valve configurations. A summary of the results of this analysis is as follows:

- Five line/valve configurations contain valves that would lift off their seat, relieving pressure, prior to exceeding design basis transient stress loading criteria.

- Ten line/valve configurations do not have means to relieve pressure resulting from either or both LOCA or PAB HELB temperature effects. The temperature profiles for the DBA conditions were used to quantify the thermally induced pressurization of the trapped fluid. This resultant pressurization was then evaluated and it was determined that the criteria of ASME section III, Appendix F, would be met ensuring operability of the affected piping/valving systems. These lines/valves require further engineering analysis to determine the actions necessary, if any, for compliance with the criteria stated in the Updated Final Safety Analysis Report.

- The remaining line/valve configuration (recirculation pump discharge sample) was also evaluated to quantify the thermally induced pressurization resulting from the HELB temperature profile as this line does not have a means to relieve pressure. The resultant pressurization was then evaluated and it was determined that this line would meet the criteria of ASME section III, Appendix F. The containment isolation valves associated with this piping do not have assurance of bonnet to body leak tightness, and, are a potential vulnerability to containment integrity. The current IST measured leakage, well within the allowable leakage, would provide for depressurization of the affected piping, ensuring operability. The configuration of these valves can be administratively controlled to ensure the fluid is voided during normal plant operation to preclude future concerns. It should be noted that this line would not be required for use during the PAB HELB accident for which these conditions were postulated.

- In determining the susceptibility to thermal overpressurization for the 16 lines discussed above, extremely conservative assumptions have been utilized. For instance, as indicated previously, it has been assumed that normally open containment isolation valves that receive a Phase A signal would close as a result of a postulated HELB in the piping penetration area. Further evaluation is expected to show that this is not the case. In addition, no credit was taken for the cushioning effect of nitrogen applied to certain lines/valves during post-accident operation of the Isolation Valve Seal Water System.

Summary Report - Generic Letter 96-06 Evaluation
Evaluation of Thermal Overpressurization of Isolated Piping Sections

CORRECTIVE ACTIONS

1. Administrative controls will be revised to ensure fluid is voided from the recirculation pump discharge sample line during normal plant operation to preclude potential overpressurization during design basis conditions.

2. Engineering analysis will be performed for the 10 lines/valves to determine the actions necessary, if any, for compliance to the criteria stated in the Updated Final Safety Analysis Report.

CONTAINMENT PIPING AND VALVING SCREENING ANALYSIS SUMMARY
TABLE 1

ITEM NO.	PEN/LINE DESCRIPTION	EVALUATION SUMMARY (1)		ACCEPTANCE CRITERIA/ REMARKS
		PAB TEMP EFFECT	VC TEMP EFFECT	
1.	PEN U, LINE # 22-3"-AC-152N: EXCESS LETDOWN HEAT HX CCWs SUPPLY	YES STRUCTURAL EVALUATION OF PIPING BETWEEN CIVs	NO LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVE.	CIVs ARE AIR OPERATED DIAPHRAGM VALVES (SELF RELIEVING) INTERCONNECTING PIPING MEETS DESIGN BASIS REQUIREMENTS
2.	PEN Y LINE # 33-3"-RC-151R: PRIMARY MAKE-UP WATER SUPPLY TO PRT AND RCP SEAL STANDPIPES	YES STRUCTURAL EVALUATION OF PIPING BETWEEN CIVs	YES STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	LINE CONTAINS AIR OPERATED DIAPHRAGM VALVES (SELF RELIEVING). PIPING MEETS DESIGN BASIS REQUIREMENTS
3.	PEN Y, LINE DW-2"-DW-151: DEMINERALIZED WATER INTO CONTAINMENT	YES THERMAL/STRUCTURAL EVALUATION FOR CIVs AND INTERCONNECTING PIPING	NO LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVE	MEETS ASME SECTION III APPENDIX F CRITERIA
4.	PEN TT, LINE # 711-3/8"-SL-2505R: RECIRCULATION PUMP DISCHARGE SAMPLE LINE	YES THERMAL/STRUCTURAL EVALUATION OF CIVs AND INTERCONNECTING PIPING	NO LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE UPSTREAM OF CIVs IS OPEN TO THE RECIRCULATION SYSTEM	PIPING MEETS ASME SECTION III APPENDIX F CRITERIA. FOR CIVs, BODY TO BONNET JOINT VULNERABLE TO LEAKAGE (SEE NOTE 2)
5.	PEN K, LINE # 10-14"-AC-601R: RESIDUAL HEAT REMOVAL LOOP OUT	YES THERMAL/STRUCTURAL EVALUATION OF THE CIV (DOUBLE DISC GATE VALVE). LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVE.	YES THERMAL/STRUCTURAL EVALUATION OF RHR SUCTION ISOLATION VALVE. LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVE.	VALVES MEET ASME SECTION III, APPENDIX F CRITERIA
6.	PEN W, LINE # 25-3/8"-SL-2505R: PRESSURIZER STEAM SPACE SAMPLE LINE	YES THERMAL/STRUCTURAL EVALUATION TO ACCOUNT FOR THE POSSIBILITY OF INLEAKAGE BETWEEN CIVs	YES THERMAL/STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	MEETS ASME SECTION III, APPENDIX F CRITERIA
7.	PEN W, LINE # 26-3/8"-SL-2505R: PRESSURIZER LIQUID SPACE SAMPLE LINE	YES THERMAL/STRUCTURAL EVALUATION TO ACCOUNT FOR THE POSSIBILITY OF INLEAKAGE BETWEEN CIVs	YES THERMAL/STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	MEETS ASME SECTION III, APPENDIX F CRITERIA

CONTAINMENT PIPING AND VALVING SCREENING ANALYSIS SUMMARY
TABLE 1

ITEM NO.	PEN/LINE DESCRIPTION	EVALUATION SUMMARY (1)		ACCEPTANCE CRITERIA/ REMARKS
		PAB TEMP EFFECT	VC TEMP EFFECT	
8.	PEN W, LINE # 59-3/8"-SL-2505R: REACTOR COOLANT SYSTEM SAMPLE	YES THERMAL/STRUCTURAL EVALUATION TO ACCOUNT FOR THE POSSIBILITY OF INLEAKAGE BETWEEN CIVs	YES THERMAL/STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	MEETS ASME SECTION III, APPENDIX F CRITERIA
9.	PEN Y, LINE # 31-3/4"-SL-1501R: SAFETY INJECTION TEST LINE FROM ACCUMULATOR TANKS # 31 THRU 34 TO RWST	YES THERMAL/STRUCTURAL EVALUATION OF CIVs AND INTERCONNECTING PIPING	NO LINE AND ITS ASSOCIATED CIVs IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE INSIDE CONTAINMENT UP TO THE FIRST CIV IS PROTECTED BY RELIEF VALVE	MEETS ASME SECTION III APPENDIX F CRITERIA
10.	PEN RR, LINE # 69-3/8"-SL-2505R: ACCUMULATORS SAMPLE LINE	YES THERMAL/STRUCTURAL EVALUATION TO ACCOUNT FOR THE POSSIBILITY OF INLEAKAGE BETWEEN CIVs.	YES THERMAL/STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	MEETS ASME SECTION III, APPENDIX F CRITERIA
11.	PEN R, LINE # 18-3"-AC-152N: EXCESS LETDOWN HEAT HX CCWS RETURN LINE	YES STRUCTURAL EVALUATION OF PIPING BETWEEN CIVs.	NO LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVE.	ONE CIV IS AIR OPERATED DIAPHRAGM VALVE (SELF RELIEVING). INTERCONNECTING PIPING MEETS DESIGN REQUIREMENTS
12.	PEN Y, LINE #338-2"-WD-151R: CONTAINMENT SUMP PUMP DISCHARGE LINE	YES STRUCTURAL EVALUATION OF PIPING BETWEEN CIVs.	YES, STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	LINE CONTAINS AIR OPERATED DIAPHRAGM VALVES (SELF RELIEVING) PIPING MEETS DESIGN BASIS REQUIREMENTS
13.	PEN Z, LINE #40-3"-WD-151R: RCDT PUMP DISCHARGE LINE	YES STRUCTURAL EVALUATION OF PIPING BETWEEN CIVs.	YES, STRUCTURAL EVALUATION OF LINE INSIDE CONTAINMENT UP TO THE FIRST CIV	LINE CONTAINS AIR OPERATED DIAPHRAGM VALVES (SELF RELIEVING) PIPING MEETS DESIGN BASIS REQUIREMENTS
14.	PEN QQ, LINE #60-8"-SI-601R: RESIDUAL HEAT REMOVAL LOOP TO SI PUMPS LINE	YES THERMAL/STRUCTURAL EVALUATION OF THE CIVs (DOUBLE DISC GATE VALVE) LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVES.	NO LINE INSIDE CONTAINMENT IS PROTECTED BY RELIEF VALVES	MEETS ASME SECTION III, APPENDIX F CRITERIA

CONTAINMENT PIPING AND VALVING SCREENING ANALYSIS SUMMARY
TABLE 1

ITEM NO.	PEN/LINE DESCRIPTION	EVALUATION SUMMARY (1)		REMARKS
		PAB TEMP EFFECT	VC TEMP EFFECT	
15.	PEN Q, LINE #16-4"-SI-1501R: SAFETY INJECTION HEADERS	YES THERMAL/STRUCTURAL EVALUATION OF THE CIVs (DOUBLE DISC GATE VALVE) LINE IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVE.	NO LINE INSIDE CONTAINMENT IS PROTECTED BY RELIEF VALVE	MEETS ASME SECTION III, APPENDIX F CRITERIA
16	PEN QQ, LINE #294-3/8"-SI-2505R: RESIDUAL HEAT REMOVAL SAMPLING LINE	YES THERMAL/STRUCTURAL EVALUATION OF CIVs AND INTERCONNECTING PIPING.	NO LINE INSIDE CONTAINMENT UPSTREAM OF THE CIVs IS NOT SUSCEPTIBLE TO THERMALLY INDUCED OVERPRESSURIZATION. LINE IS PROTECTED BY RELIEF VALVES	MEETS ASME SECTION III APPENDIX F CRITERIA

NOTES:

1. Containment temperature effect on CIVs and their interconnecting piping located outside containment following a LOCA is negligible, since the temperature along the pipe will decay sufficiently prior to reaching the first containment isolation valve.

2. Configuration of CIVs associated with line # 711, recirculation pump discharge sample line will be administratively controlled. Draining of the sampling line between CIVs is an easily obtainable procedural corrective action.

COMMITMENT LIST

Number	Commitment	Due
IPN-97-	The results of the evaluation for the postulated new single failure (fan supply breaker fails to open during LOCA and a coincident LOOP) will be submitted by March 3, 1997 as an update to this response or it will be submitted earlier as a Licensee Event Report .	March 3, 1997
IPN-97-	Administrative controls will be revised to ensure fluid is voided from the recirculation pump discharge sample line during normal plant operation to preclude potential overpressurization during design basis conditions.	Prior to exceeding Cold Shutdown from the present Forced Outage
IPN-97-	Perform engineering analysis for the 10 lines/valves to determine the actions necessary, if any, for compliance to the criteria stated in the Updated Final Safety Analysis Report.	Prior to R09 start-up