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William J. Cahill, Jr.
Chief Nuclear Officer

January 27, 1997
JPN-97-003

U. S. Nuclear Regulatory Commission
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Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
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 the 120 day response for the James A. FitzPatrick Nuclear Power Plant as requested by the referenced Generic Letter 96-06 (GL 96-06). The generic letter requests licensees to perform an evaluation to determine:

(1) if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions; and

(2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

The generic letter further states that if the systems are susceptible to these conditions, an operability assessment of the affected systems shall be performed and corrective actions implemented as appropriate.

The summary report, stating the conclusions of the evaluation, is presented in Attachment 1. The report concludes that a potential two phase flow / water hammer event during the postulated GL 96-06 accident is not a safety concern. For the thermal overpressurization issue the report concludes that systems evaluated for the GL 96-06 scenarios are either not susceptible to these conditions, or if susceptible, retain operability of their safety related function during DBA conditions.

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The conclusions for the thermal overpressurization evaluation are based on an operability determination performed in accordance with the guidelines of Generic Letter 91-18. Detailed analyses are in progress for final disposition of the thermal overpressurization issue. The results of this detailed analyses, including a long term corrective action plan if required, will be submitted by May 27, 1997.

Attachment 2 summarizes the commitments made in this submittal. If you have any questions, please contact Ms. C. D. Faison.

Very truly yours,

William J. Cahill

William J. Cahill, Jr.
Chief Nuclear Officer

STATE OF NEW YORK
COUNTY OF WESTCHESTER

Subscribed and sworn to before me
this 27th day of January, 1997.

Geraldine Strand
Notary Public

GERALDINE STRAND
Notary Public, State of New York
No. 4991272
Qualified in Westchester County
Commission Expires Jan. 27, 1998

Attachments: as stated

cc: Regional Administrator
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James A. Fitzpatrick Nuclear Power Plant
Docket 50-333

ATTACHMENT 1 TO JPN 97-003

Summary Report - Generic Letter 96-06 Evaluation

Water Hammer and Two Phase Flow Evaluation

Generic Letter 96-06 requests licensees to determine if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions. This issue is primarily concerned with safety-related containment air coolers which are credited for heat removal from containment during design basis accident (DBA) conditions. The generic letter does point out, however, that water hammer in cooling water systems associated with non-safety related containment air coolers can also challenge containment integrity by creating a containment bypass flow path for interfacing safety related systems. The effects of water hammer and two-phase flow on the containment air cooler cooling water system was not determined for the FitzPatrick plant since their occurrence, and any damage associated with their occurrence, would not affect a safety-related function as discussed below.

For the FitzPatrick plant, the containment air coolers of concern are the Drywell Coolers (68E-1A to 1D and 68E-3A to 3D). These units are non-safety related and are not credited for heat removal from containment during DBA conditions. Therefore the two phase flow concern of GL 96-06, which deals with degradation of the cooler heat removal capability during DBA scenarios, is not a concern for the FitzPatrick plant.

The cooling water system associated with the Drywell Coolers is the non-safety related Reactor Building Closed Loop Cooling (RBCLC) system. An interface does exist between the safety related Emergency Service Water (ESW) system and the RBCLC. However, the interconnecting piping circuits from the ESW to the RBCLC are isolated by normally closed valves. Manual operator action is required to feed ESW to the RBCLC system. Containment isolation valves, operated from the control room, provide the ability to isolate the RBCLC piping loop to the Drywell coolers should a break have occurred. Therefore, the water hammer concern of GL 96-06, which deals with degradation of containment integrity due to the creation of a "by-pass flow path", is not a concern for the FitzPatrick plant. The Authority will review the plant procedures to determine if additional operator instructions are warranted.

Based on the above analysis, it is concluded that the two phase flow / water hammer event postulated in GL 96-06 is not a safety concern for the FitzPatrick plant.

Additionally, NRC Information Notice 96-60 "Potential Common Mode Post-Accident Failure of RHR Heat Exchangers" dated November 14, 1996 has been reviewed. It was concluded that the concerns of this information notice were not applicable to this plant due to the use of a keep full system.

Thermal Pressurization Evaluation

Generic Letter 96-06 requests licensees to evaluate if thermally induced overpressurization of isolated water filled piping sections in containment could jeopardize the ability of accident mitigating systems to perform their safety functions and/or lead to a breach of containment integrity via bypass leakage. In order to analyze this issue, the following evaluation process was used:

- a review of Drywell and Suppression Pool penetrations was made to identify those which involved liquid systems,
- valve arrangements for the above systems were reviewed to determine if isolated water filled sections exist and if thermal pressurization was possible,
- a heat transfer model was developed to determine the effect of the post accident ambient area temperatures on the isolated water temperature,
- the isolated pipe section internal pressure was then determined,
- containment integrity and safety system function were evaluated.

The containment penetrations (584 total) were reviewed, of which 314 contained water . After excluding penetrations that do not contain isolated water filled pipe sections only fourteen (14) penetrations remained. Four of these fourteen penetrations were eliminated from further thermal pressurization and containment integrity evaluation since the normal operating water temperature would be the same or higher than the DBA condition. Another two penetrations were eliminated from further evaluation since the limiting pressure, based on valve design feature, resulted in stresses below design allowable limits. The remaining eight penetrations, identified as being susceptible to thermal pressurization concerns, are the following:

Penetration No.	System	Inboard Valve	Outboard Valve
X-8	Main Steam Drain	29MOV-74	29MOV-77
X-19	Drywell Equipment Drain Sump Discharge	20MOV-94	20AOV-95
X-210 A&B	RHR to Suppression Pool	10MOV-34 A&B	10MOV-39 A&B
X-211 A&B	RHR - Suppression Spray Header	10MOV-38 A&B	10MOV-39 A&B
X-224	RCIC Pump Suction from Suppression Pool	13MOV-41	13MOV-39

X-226	HPCI Pump Suction from Suppression Pool	23MOV-58	23MOV-57

System Safety Function Review

The valves associated with the eight penetrations identified to be susceptible to thermal pressurization were evaluated as to their ability to perform their safety function.

The systems associated with penetrations X-8, X-19, and X-224 do not perform safety functions. The associated containment isolation valves are closed for DBA conditions and remain closed for the duration of the event.

Penetrations X-210 A&B and X-211 A&B (four penetrations) employ inboard globe valves with isolated fluid under the seats. An increase in line pressure due to thermal conditions would assist valve opening. Therefore, valve operability and consequently system safety function will not be jeopardized.

The valves associated with penetration X-226 (23MOV-57 & 58) are required to open during transfer of HPCI suction water source from the Condensate Storage Tank (CST) to the Suppression Pool. Thermal pressurization of this containment penetration configuration is due to elevated area temperatures resulting from elevated Suppression Pool temperatures during DBA conditions. A review of plant transient and accident analyses indicates that the transfer function is not required except for initiating events which would render the CST unavailable. No accident sequences have been identified which would require opening of these valves coincident with elevated Suppression Pool temperature. Therefore, the system safety function will not be jeopardized.

Containment Integrity Review

The eight penetrations identified to be susceptible to thermal pressurization were evaluated for operability in accordance with NRC Generic Letter 91-18. Existing stress analyses, pipe support, penetration, integral welded attachment details, and integrity of valve internals, were evaluated for all of the eight penetrations. The evaluation determined that the isolated fluid pressures for the thermal overpressurization condition were above operating design pressures; however, the effected penetrations remain operable in that they will not rupture and they will continue to perform their safety function.

Conclusion

The thermal overpressurization evaluation performed in accordance with GL 91-18 demonstrates that containment integrity is maintained and safety functions are not compromised, for the susceptible containment penetrations and their associated valves, during the GL 96-06 postulated accident. Detailed analyses are in progress for final disposition of the thermal overpressurization issue. The results of this analyses, including a long term corrective action plan if required, will be submitted by May 27, 1997.

Attachment 2 to JPN-97-003

James A. FitzPatrick Nuclear Power Plant

List of Commitments

Number	Commitment	Due Date
JPN-97-003-01	Review plant procedures to determine if additional operator instructions are warranted.	May 27, 1997
JPN-97-003-02	Submit the detailed GL 96-06 analysis, with a corrective action plan if required.	May 27, 1997