Lewis Sumner
Vice President
Hatch Project Support

Southern Nuclear Operating Company, Inc.
40 Inverness Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.7279
Fax 205.992.0341

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

Edwin I. Hatch Nuclear Plant
Completion of Remaining Actions Addressing GL 96-06 Concerns

Ladies and Gentlemen:

By letter dated May 2, 2002, the Nuclear Regulatory Commission (NRC) requested Southern Nuclear Operating Company (SNC) to complete actions addressing GL 96-06(1) concerns for the Edwin I. Hatch Nuclear Plant. In GL 96-06, the NRC requested that "licensees evaluate (among other things) waterhammer and two-phase flow concerns associated with containment air cooler cooling water systems." SNC's remaining actions to fully address these concerns were previously deferred pending NRC review and approval of EPRI Report TR-1003098.(2) By letter dated April 3, 2002, the NRC provided acceptance of EPRI Report TR-113594 (later changed to TR-1003098) for performing evaluations addressing GL 96-06 waterhammer concerns to the extent specified and within the limitations delineated in the subject EPRI report and the associated NRC Safety Evaluation Report (SER). The purpose of this letter is to complete the remaining actions relative to GL 96-06 issues and submit the information referred to in Section 3.3 of the NRC SER for EPRI Report TR-113594 (later changed to TR-1003098).

The Enclosure provides summaries of SNC's previous responses to GL 96-06, as well as SNC's response to the NRC's letter dated May 2, 2002. Attachment 1 to the Enclosure provides the certification that the EPRI methodology, including clarifications, was properly applied, and plant-specific risk considerations are consistent with the risk perspective provided in the EPRI letter to the NRC dated February 1, 2002.(3) Attachment 2 provides SNC's response to the NRC's April 14, 1998, Request for Additional Information(4), as requested by the NRC in the May 2, 2002 letter. Attachment 3 provides a summary of the results and conclusions of the Unit 1 water hammer analysis.

Should you have any questions in this regard, please contact this office.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

DMC/sp

A072

References:

1. NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996.

2. EPRI Technical Basis Report and User's Manual, Final Report No. 1003098, "Generic Letter 96-06 Water Hammer Issues Resolution," dated April 12, 2002.

3. EPRI letter to NRC, "Response to ACRS Comments (Letter dated 10/23/01) on the EPRI Report on Resolution of NRC GL 96-06 Waterhammer Issues," dated February 1, 2002.

4. NRC letter to SNC, "Request for Additional Information Related to the Generic Letter (GL) 96-06 Response for Edwin I. Hatch Nuclear Plant Units 1 and 2 (TAC Nos. M96819 and M96820), dated April 14, 1998.

Enclosure: Resolution of Remaining Actions Relative to Generic Letter 96-06
Attachment 1: Certification of Applicability of EPRI Methodology
Attachment 2: Response to Request for Additional Information (NRC Letter Dated April 14, 1998)
Attachment 3: Unit 1 Generic Letter 96-06 Waterhammer Issue Summary and Conclusions

cc: Southern Nuclear Operating Company (w/o enclosures)
Mr. P. H. Wells, Nuclear Plant General Manager
SNC Document Management (R-Type A02.001)

U. S. Nuclear Regulatory Commission, Washington, D. C.
Mr. L. N. Olshan, Licensing Project Manager - Hatch

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector - Hatch

State of Georgia (w/o enclosures)
Mr. L.C. Barrett, Commissioner - Department of Natural Resources

Enclosure

Edwin I. Hatch Nuclear Plant
Resolution of Remaining Actions Relative to Generic Letter 96-06

Enclosure

Edwin I. Hatch Nuclear Plant
Resolution of Remaining Actions Relative to Generic Letter 96-06

BACKGROUND

In Generic Letter (GL) 96-06[1][2], the NRC identified the following safety-significant issues, related to the primary containment integrity and equipment operability, that could have generic implications and thus, required further evaluation by all licensees:

(1) *Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of waterhammer during either a loss-of-coolant accident (LOCA) or a main steam line break (MSLB). These cooling water systems were not designed to withstand the hydrodynamic effects of waterhammer and corrective actions may be needed to satisfy system design and operability requirements.*

(2) *Cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios. The heat removal assumptions for design-basis accident scenarios were based on single-phase flow conditions. Corrective actions may be needed to satisfy system design and operability requirements.*

(3) *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 could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system design and operability requirements.*

The 120-day response to the GL 96-06 required addressees to submit a written summary report providing the following information:

1. A determination of whether containment air cooler cooling systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions.

2. A determination of whether piping systems that penetrate containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

3. Conclusions that were reached relative to susceptibility of water hammer and two-phase flow in the containment air cooler cooling water system and overpressurization of piping that penetrates containment.

4. The basis for continued operability of affected systems and components as applicable.

5. Corrective actions that either were implemented or are planned to be implemented.

6. Identification of systems found to be susceptible to the conditions discussed in GL 96-06 and a description of the specific circumstances involved.

SUMMARY OF PREVIOUS SNC CORRESPONDENCE TO THE NRC

Southern Nuclear Operating Company (SNC) has provided to the NRC several responses to GL 96-06 issues.[3, 4, 5, 6, 7, 8] A summary of these responses is provided below.

Summary of SNC's Response to GL 96-06 Issue No. (1)

1. Due to the system design for Plant Hatch Unit 1, waterhammer is possible in the drywell cooling units/piping system whether or not the fan cooling units are operated following a LOCA with a loss of offsite power (LOSP). Thus, SNC elected to perform a detailed analysis to determine the potential for waterhammer and its effects on the piping system and support structure. Also, SNC elected to participate in the EPRI demonstration program, which was to provide more realistic assumptions to use in the analysis. Further response to this issue was deferred pending completion and review of the EPRI Technical Basis Report and User's Manual.[8]

2. Cooling water to the Unit 2 drywell coolers is provided by a closed-loop drywell chilled water system. This system, which includes the chillers, pumps, and drywell cooling units, receives a LOCA signal that will automatically shut down the system. Prior to restarting the system, the operator must manually override the signal using a LOCA override procedure. SNC considered the various failure modes of the system and concluded that waterhammer could occur even after the containment has begun to cool. Thus, Emergency Operating Procedure 31EO-EOP-100-2S, Miscellaneous Emergency Overrides, was revised to prohibit operation of the drywell coolers, in conjunction with a LOCA, when boiling due to containment high temperature may have occurred in any of the drywell cooling units/piping system.[8, 9, 10, 11, 12]

Summary of SNC's Response to GL 96-06 Issue No. (2)

The Plant Hatch accident analyses do not take credit for operation of the drywell coolers following a LOCA. Hence, availability of the drywell coolers for post accident heat removal is not a safety concern. However, the cooling water piping to the coolers [plant service water (PSW) system for Unit 1 and the chilled water system for Unit 2] performs the safety related function of containment isolation following a LOCA (since credit is taken for the closed loop inside the primary containment). The events described in GL 96-06 have the potential for a breach in the primary containment boundary following a LOCA in conjunction with an LOSP [see Summary of SNC's Response to GL 96-06 Issue No. (1) above]. The concern of breach of the primary containment due to overpressurization of penetrations is addressed below in Summary of SNC's Response to GL 96-06 Issue No. (3).

Summary of SNC's Response to GL 96-06 Issue No. (3)

The following penetrations were identified to be potentially susceptible to overpressurization and required further evaluation:[4, 5, 6]

- Units 1 and 2 Residual Heat Removal (RHR) Shutdown Cooling Suction Line (Penetration No. 12).
- Units 1 and 2 Drywell Floor Drain Sump Discharge Line (Penetration No. 19).

- Units 1 and 2 Drywell Equipment Drain Sump Discharge Line (Penetration No. 18).
- Units 1 and 2 Demineralized Water Lines to the Drywell (Penetration No. 46).
- Unit 2 Drywell Chemical Drain Sump Line (Penetration No. 55).

Based upon the evaluation results, plant modifications under Design Change Requests (DCRs) 97-005 and 97-006 were made to ensure the integrity of Penetration Nos. 12, 18, and 19. The piping sections subject to thermally induced pressure buildup were provided with a pressure relief path to ensure the piping will not exceed the design pressure limits.

General Operating Procedures 34GO-OPS-028-1 and 34GO-OPS-028-2S, Drywell Closeout, were revised to ensure piping between the isolation valves at Penetration Nos. 46 (Units 1 and 2) and 55 (Unit 2) is drained prior to plant startup.(4)(5)

By letter dated January 21, 1999, the NRC accepted SNC's corrective actions in response to GL 96-06 Issue No. (3).(12)

SNC's CLOSEOUT OF GL 96-06 ISSUES

By letter dated May 2, 2002,(16) the NRC requested SNC to complete actions to address GL 96-06 and submit the information referred to in Section 3.3 of the NRC Safety Evaluation Report (SER) for EPRI Report TR-113594 (later corrected to 1003098).(13) Licensees who choose to use the methodology in the EPRI Report for addressing the GL 96-06 waterhammer issue may do so by supplementing their response to include:

- *Certification that the EPRI methodology, including clarifications, was properly applied, and that plant-specific risk considerations are consistent with the risk perspective that was provided in the EPRI letter dated February 1, 2002. If the uncushioned velocity and pressure are more than 40 percent greater than the cushioned values, also certify that the pipe failure probability assumption remains bounding. Any questions that were asked previously by the staff with respect to the GL 96-06 waterhammer issue should be disregarded.*
- *The additional information that was requested in RAIs that were issued by the NRC staff with respect to the GL 96-06 two-phase flow issue (as applicable).*
- *A brief summary of the results and conclusions that were reached with respect to the waterhammer and two-phase flow issues, including problems that were identified along with corrective actions that were taken. If corrective actions are planned but have not been completed, confirm that the affected systems remain operable and provide the schedule for completing any remaining corrective actions.*

The following discussion provides the information requested in Section 3.3 of the NRC SER.[13]

The Edwin I. Hatch Nuclear Plant Unit 1 waterhammer analysis for the drywell coolers is complete. The analysis was performed using the EPRI methodology and the clarifications provided in the EPRI report. The plant-specific risk considerations are consistent with the risk perspective provided in the EPRI letter dated February 1, 2002.[14] (See Attachment 1.) No credit for the cushioning effects of gas and steam in the void just prior to column closure is taken.

Based upon the analysis results, the logic for the PSW inlet valves to the drywell coolers required modification to ensure the valves stay open to mitigate/lessen the waterhammer severity and ensure the integrity of the piping following a LOCA in conjunction with an LOSP. Plant modifications under DCR 02-020 to provide the necessary corrective action are complete. Attachment 3 provides the brief summary of the results and conclusions of the Unit 1 water hammer analysis.

The Unit 1 PSW system analyses for GL 96-06 included an evaluation of the possibility of two-phase flow through and downstream of the drywell cooler tubes. From a functional point of view, occurrence of two-phase flow is not relevant because these coolers are not credited for post accident heat removal in the plant accident analyses. From the waterhammer point of view, one is concerned with large-scale steam bubble formation in the two-phase flow regions and their possible collapse when transported to colder water that could lead to a potential waterhammer situation. Analysis demonstrated that the degree of subcooling in water leaving the PSW drywell coolers tubes, under maximum heat removal conditions during a LOCA, is so large that two-phase flow is not expected in the tubes or downstream piping. Furthermore, there are no major flow restrictions in the piping where cavitation can occur. Therefore, SNC concluded that the two-phase flow issue does not exist in the Unit 1 PSW system.

By letter dated April 14, 1998, the NRC issued a Request for Additional Information (RAI).[11] SNC provided the requested responses by letters dated June 30, 1998; July 8, 1998; and November 20, 1998.[6, 7, 8] A summary of SNC's responses to the RAI is provided in Attachment 2.

REFERENCES:

1. NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996.

2. NRC Generic Letter 96-06, Supplement 1: "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated November 13, 1997.

3. Georgia Power Company (GPC) Letter No. HL-5253 to NRC, "30 Day Response to GL 96-06," dated October 21, 1996.

4. GPC Letter No. HL-5302 to NRC, "120 Day Response to GL 96-06," dated January 27, 1997.

5. SNC letter HL-5492 to NRC, "Generic Letter 96-06 - Thermally Induced Pressurization," dated October 20, 1997.

6. SNC letter HL-5645 to NRC, "Generic Letter 96-06 - Water Hammer in Containment Coolers," dated June 30, 1998.

7. SNC letter HL-5652 to NRC, "Generic Letter 96-06 - Water Hammer in Containment Coolers, Revised Response to Question No. 2," dated July 8, 1998.

8. SNC letter HL-5708 to NRC, "Updated Response to Request for Additional Information, Generic Letter 96-06 - Water Hammer in Containment Coolers," dated November 20, 1998.

9. NRC letter to SNC, "Request for Additional Information Regarding Response to Generic Letter (GL) 96-06, Edwin I. Hatch Nuclear Plant Units 1 and 2 (TAC Nos. M97478 and M97479)," dated September 23, 1997.

10. NRC letter to SNC, "Information Pertaining to Edwin I. Hatch Nuclear Plant Unit 1 Implementation of Modifications Associated with GL 96-06, 'Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions' (TAC No. M96819)," dated November 3, 1997.

11. NRC letter to SNC, "Request for Additional Information Related to the Generic Letter (GL) 96-06 Response for Edwin I. Hatch Nuclear Plant Units 1 and 2 (TAC Nos. M96819 and M96820)," dated April 14, 1998.

12. NRC letter to SNC, "Generic Letter (GL) 96-06 Responses for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (TAC Nos. M96819 and M96820)," dated January 21, 1999.

13. NRC letter to Mr. Vaughn Wagoner, Chairman EPRI Waterhammer Project Utility Advisory Group - NRC Acceptance of EPRI Report TR-113594 (later corrected to 1003098), "Resolution of Generic Letter 96-06 Waterhammer Issues, Volumes 1 and 2," dated April 3, 2002. Enclosure: Evaluation of Electric Power Research Institute Report TR-113594, "Resolution of Generic Letter 9606 Waterhammer Issues," Volumes 1 and 2, dated December 2000 (NRC SER).

14. EPRI letter to NRC, "Response to ACRS Comments (Letter dated 10/23/01) on the EPRI Report on Resolution of NRC GL 96-06 Waterhammer Issues," dated February 1, 2002.

15. EPRI Technical Basis Report and User's Manual, Final Report No. 1003098, "Generic Letter 96-06 Water Hammer Issues Resolution," dated April 12, 2002.

16. NRC letter to Lewis Sumner (SNC), "Edwin I. Hatch Nuclear Plant Units 1 and 2 - RE: Electric Power Research Institute (EPRI) Report TR-113594, 'Resolution Generic Letter 96-06 Water Hammer Issues,' Volumes 1 and 2 (TAC Nos. M96819 and M96820)," dated May 2, 2002.

Attachment 1

Edwin I. Hatch Nuclear Plant
Certification of Applicability of EPRI Methodology

The EPRI methodology, including clarifications, was properly applied in a manner suited to the Plant Hatch Probabilistic Safety Assessment (PSA) model, and plant-specific risk considerations are consistent with the risk perspective provided in the EPRI letter to the NRC dated February 1, 2002.[1]

The EPRI methodology suggests treating the loss-of-coolant accident/loss of offsite power (LOCA/LOSP) frequency as a contributor to the conditional core damage probability and conditional large early release probability. The conditional model is based on core damage and large early release due to loss of the drywell coolers.

The drywell coolers are not considered a major source of containment heat removal for the Plant Hatch PSA. The PSA contributions are:

1. A special initiating event based on high drywell pressure.

2. A contributor to a LOCA signal, which triggers various model actions.

3. A contributor to the need to depressurize the reactor based on drywell temperature with a failure to vent.

The predominate PSA failure mechanisms for loss of the Unit 1 drywell cooler system are plant service water (PSW) inlet and outlet valve failures, and operator error associated with restarting the cooling system following a LOCA condition. In addition, the model worth of the PSA special initiating event for loss of drywell cooling is very low.

Using the EPRI methodology for the Generic Letter (GL) 96-06 response, a proposed LOCA/LOSP initiating event frequency is as follows.

The Plant Hatch PSA model uses the frequencies listed below in calculating LOCA and LOSP. The frequencies are used to calculate a value for the initiating event, LOCA/LOSP.

- Large Break LOCA (LLOCA) = 3.0E-05 events/year
- Medium Break LOCA (MLOCA) = 3.98E-05 events/year
- LOSP = 1.89E-02 events/year

1. LLOCA x LOSP = 5.67E-07 events/year
2. MLOCA x LOSP = 7.5E-07 events/year
3. Combined Total = 1.32E-06 events/year

If the special initiating event for the loss of drywell cooling frequency is used as a conditional probability, the result is 6.0E-09 based on the Unit 1, Revision 1a, PSA model (1E-10 cutoff frequency). This result is for core damage frequency (CDF), because the loss of drywell cooling special initiating event has negligible effect on the large early release probability. Multiplication of Item 1, 2, or 3 with this number produces a value significantly lower than value given in Regulatory Guide 1.74.

The end result, including the frequency for item 1 above, is bounded by EPRI Report TR-1003098, Section 3.[2]

REFERENCES:

1. EPRI letter to NRC, "Response to ACRS Comments (Letter dated 10/23/01) on the EPRI Report on Resolution of NRC GL 96-06 Waterhammer Issues," dated February 1, 2002.

2. EPRI Technical Basis Report and User's Manual, Final Report No. 1003098, "Generic Letter 96-06 Water Hammer Issues Resolution," dated April 12, 2002.

Attachment 2

Edwin I. Hatch Nuclear Plant
Response to Request for Additional Information
(NRC Letter dated April 14, 1998)

NRC Question No. 1

The event descriptions provided in the January 27, 1997 response require some clarification. For example, it is not clear to what extent automatic functions occur such as when the cooling water pumps trip and when the containment/drywell isolation valves for the cooling water systems go closed, and to what extent these functions are automatically restored at some point during the event. Provide a detailed description of the "worst case" scenarios for water hammer and two-phase flow, which is clear on component status and response, taking into consideration the complete range of event possibilities, system configurations, and parameters. For example, all water hammer types and water slug scenarios should be considered as well temperatures, pressures, flow rates, load combinations and potential component failures. Additional examples include:

- *The consequences of column separation and rejoining (not adequately addressed for Hatch Unit 1);*
- *The consequences of steam formation, transport, and accumulation;*
- *Cavitation, resonance, and fatigue effects; and*
- *Erosion considerations.*

The licensee's assessment of the two-phase flow issue only addressed heat transfer considerations, and did not include consideration of the last three items (above). Licensees may find NUREG/CR-6031, "Cavitation Guide for Control Valves," helpful in addressing some aspects of the two-phase flow analyses.

SNC Response

Unit 1

SNC elected to perform a detailed analysis to determine the potential for waterhammer and its effects on the piping system and support structure by use of the EPRI methodology. The analysis is complete and the summary of the results and conclusions are provided in Attachment 3.

Unit 2

SNC provided a final response to NRC Question No. 1 by letters dated June 30, 1998,(1) and November 20, 1998.(2)

Also, both Units 1 and 2 were evaluated for the potential of two-phase flow. As discussed in the Enclosure, SNC concluded that the potential for two-phase flow does not exist at either unit.

NRC Question No. 2

The licensee's response indicates that because EOPs do not provide for use of the Reactor Building Closed Cooling Water (RBCCW) system and the containment equipment coolers, the system is not a concern for waterhammer. However, during an accident, operators may consider the RBCCW system to still be available as an option for accident mitigation. Discuss measures that exist to assure that operators will not use the RBCCW system as an option during periods when waterhammer and two-phase flow could compromise system integrity.

SNC Response

The final response to NRC Question No. 2 was provided by SNC letter dated July 8, 1998.(3)

NRC Question No. 3

The waterhammer discussion for Hatch Unit 2 is confusing. The licensee's response indicates that waterhammer could occur in the chilled water system and, because a detailed waterhammer analysis has not been performed, plant procedures were revised to prohibit operation of the containment area coolers when containment temperatures are above the point where boiling may occur. However, the discussion goes on to say that primary evaluation shows that the containment boundary provided by the system piping and cooling coils is expected to be maintained following a waterhammer event. Clarification is required as to what positive measures have been taken, including procedure changes and changes to EOPs, to assure that containment integrity will not be compromised as a result of a waterhammer in the chilled water system.

SNC Response

The final response to NRC Question No. 3 was provided SNC letters dated June 30, 1998,(1) and November 20, 1998.(2).

NRC Question No. 4

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

SNC Response

The failure effects of components associated with the Units 1 and 2 drywell coolers with a potential to affect the waterhammer analysis were provided to the NRC by SNC letter dated June 30, 1998.(1)

The detailed water hammer analysis performed for Unit 1 considers the effects of failure of these components. No detailed waterhammer analysis was needed for Unit 2. However, the effects of the failures of all these components were considered in reaching this conclusion, as discussed in Reference 1.

Also, both Units 1 and 2 were evaluated for the potential of two-phase flow. As discussed in the Enclosure, SNC concluded that the potential for two-phase flow does not exist at either unit.

NRC Question No. 5

Explain and justify all uses of "engineering judgment" that were used in assessing the waterhammer and two-phase flow issues.

SNC Response

The original Unit 1 waterhammer evaluation was based on "engineering judgment." However, a detailed analysis using the guidance of EPRI Report TR-1003098[4], not engineering judgment, was completed for Unit 1. No further response to NRC Question No. 5 is needed.

The Unit 2 evaluation is not based on engineering judgment.

Also, both Units 1 and 2 were evaluated for the potential of two-phase flow. As discussed in the Enclosure, SNC concluded that the potential for two-phase flow does not exist at either unit.

NRC Question No. 6

If a methodology other than that discussed in NUREG/CR-5220, "Diagnosis of Condensation-induced Waterhammer," was used in evaluating the effects of waterhammer, describe this alternate methodology in detail. Explain why this methodology is applicable and gives conservative results for the Hatch Units (typically accomplished through rigorous plant-specific modeling, testing, and analysis). Also, provide the following information for each of the Hatch units (as applicable):

a) *Identify any computer codes that were used in the waterhammer and two-phase flow analysis and describe the methods used to benchmark the codes for the specific loading conditions involved (see Standard Review Plan Section 9.3.1).*
b) *Describe and justify all assumptions and input parameters (including those used in any computer codes) such as amplifications due to fluid structure interaction, cushioning, speed of sound, force reductions, and mesh sizes, and explain why the values selected give conservative results. Also, provide justification for omitting any effects that may be relevant to the analysis (e.g. fluid structure interaction, flow induced vibration, erosion).*
c) *Determine the uncertainty that exists in the waterhammer and two-phase flow analyses, explain how the uncertainty was determined, and how it was accounted for in the analyses to assure conservative results for the Hatch units.*

SNC Response

The guidance and methodology discussed in EPRI Report TR-1003098[4] was used for the Unit 1 waterhammer analysis. The computer program HSTA (Hydraulic Systems Transient Analysis) was used to complete the waterhammer analysis. The results of the HSTA analysis were used for the structural evaluations by use of the computer program ME-101. The detailed descriptions of both the programs, including approach, capability, validation, and techniques, were provided to the NRC by SNC letter dated November 20, 1998.[2] Further information is provided in Attachment 3.

A detailed waterhammer analysis for Unit 2 was not needed.

Also, both Units 1 and 2 were evaluated for the potential of two-phase flow. As discussed in the Enclosure, SNC concluded that the potential for two-phase flow does not exist at either unit.

NRC Question No. 7

For those scenarios where waterhammer and two-phase flow could occur (e.g. measures have not been taken to prevent occurrence), confirm that the waterhammer and two-phase flow loading conditions will not exceed any design specifications or recommended service conditions for the piping systems and components (e.g. containment/drywell isolation valves), including those stated by equipment vendors; and confirm that the system will continue to perform its design-basis functions as assumed in the safety analysis report for the facility.

SNC Response

For Unit 1, the piping system design and service conditions were considered in performance of the detailed analysis. The analysis results revealed the need for a design change to comply with the original design basis. This design change was implemented under DCR 02-020, Depower Solenoids on PSW Valves (Drywell Cooler Cooling Water Inlet Valves).

For Unit 2, in order to eliminate the potential for waterhammer, Emergency Operating Procedure 31EO-EOP-100-2S was revised to prohibit operating the drywell coolers, in conjunction with a LOCA, when boiling due to containment high temperature may have occurred in any of the drywell cooling units/piping system. Therefore, a detailed waterhammer analysis for Unit 2 was not needed.

Also, both Units 1 and 2 were evaluated for the potential of two-phase flow. As discussed in the Enclosure, SNC concluded that the potential for two-phase flow does not exist at either unit.

NRC Question No. 8

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

SNC Response

Simplified diagrams of the Unit 1 PSW system and the Unit 2 chilled water system were provided to the NRC by SNC letter dated June 30, 1998.[1]

REFERENCES

1. SNC letter HL-5645 to NRC, "Generic Letter 96-06 - Water Hammer in Containment Coolers," dated June 30, 1998.

2. SNC letter HL-5708 to NRC, "Updated Response to Request for Additional Information, Generic Letter 96-06 - Water Hammer in Containment Coolers," dated November 20, 1998.

3. SNC letter HL-5652 to NRC, "Generic Letter 96-06 – Water Hammer in Containment Coolers, Revised Response to Question No. 2," dated July 8, 1998.

4. EPRI Technical Basis Report and User's Manual, Final Report No. 1003098, "Generic Letter 96-06 Water Hammer Issues Resolution," dated April 12, 2002.

Attachment 3

Edwin I. Hatch Nuclear Plant
Unit 1 Generic Letter 96-06 Waterhammer Issue Summary and Conclusions

Introduction

Unit 1 drywell cooling is an open-loop system with cooling water provided by the plant service water (PSW) system. The PSW pumps take suction from the river and discharge to the circulation water flume. A detailed waterhammer analysis was performed for this system to address the GL 96-06 waterhammer issue. During normal plant operation, PSW system provides cooling to various reactor building & turbine building components and the emergency diesel generators (EDGs). The worst-case scenario for which the analysis was performed is a loss of coolant accident (LOCA) with a concurrent loss of offsite power (LOSP).

Following the LOCA, in conjunction with an LOSP event, the PSW pumps trip, all motor operated valves (MOVs) fail as-is, and the drywell cooler PSW inlet valves (air-operated valves) fail in the open position due to loss of power. The subsequent loss of pump pressure and heat addition from the post LOCA drywell atmosphere will cause voiding (steam generation) in the drywell coolers and adjacent piping. These drywell coolers are located at a higher elevation relative to the circulation water flume. With only an LOSP (i.e., no LOCA), the top coil of the top drywell coolers will void somewhat, but the middle and lower level drywell coolers will remain water solid. Therefore, heat input to the cooler tubes is required to cause any significant voiding in the cooler system.

When power is available from the emergency diesel generators (EDGs) in approximately 12 sec, the turbine building isolates (safety-related single-failure-proof valve arrangement), but the drywell cooler PSW inlet valves remain open because of the modified logic. The containment isolation valves (MOVs) for the drywell cooling system do not receive the containment isolation signal and thus, remain open.

After approximately 30 sec, two of the four PSW pumps (one in each PSW division) start to supply cooling water to the EDGs and the reactor building. The PSW pump aligned to Division I of the PSW system will supply water to the drywell coolers.

Column Closure Waterhammer (CCWH) Versus Condensation-Induced Waterhammer (CIWH)

The determination of the bounding waterhammer between CIWH and CCWH was made using the recommended approach in Section 4.2 of EPRI Report TR-1003098.[1] This approach shows that the CCWH will be bounding for Unit 1. Therefore, an explicit calculation of the CIWH magnitude was not performed for Unit 1, as recommended in Reference 1.

Column Closure Waterhammer

System Voiding Analysis

The voiding caused by heat addition to the drywell cooler tubes was determined in a conservative manner by assuming that, as long as there is boiling in the tubes, the steam pressure in the drywell cooler tubes is the same as the steam pressure in the drywell corresponding to the drywell temperature. This, in effect, considers that the heat transfer from the drywell atmosphere to the PSW in the tubes is infinite. Once the water inventory in the drywell coolers is completely depleted, the steam pressure is reduced as it expands.

The system hydraulics (flow resistances, elevations, etc.) for Unit 1 then gives the total voiding before the PSW pumps restart. The analysis shows that all the drywell coolers tubing and some supply & return piping will void before the pumps restart.

System Refill Analysis

With the void size determined by the system voiding analysis, the computer program HSTA was used to determine the initial closure velocities and forcing functions on affected pipe run segments.

HSTA is a one dimensional finite difference program that is based on the Method of Characteristics (MOC). It allows for large models containing as many as 18,000 nodal points. Since, the Unit 1 PSW system is large and complex with some components at high elevations relative to the supply and discharge reservoir, HSTA provided the needed analytical capability for the waterhammer analysis. The program uses Wylie and Streeter's well-known application of the Method of Characteristics,[2] as referenced in EPRI Report TR-1003098.[1]

The system is nodalized in a fixed time step/nodal distance grid. If either the upstream or downstream node connects to a hydraulic device such as a valve, pump, etc., then this device or "boundary condition" is treated properly in the MOC solution of the governing equations. Several published papers discuss details pertaining to the treatment of several commonly used boundary conditions in the HSTA program. Force-time history computations are performed directly within the HSTA program during the calculation of the flow variables. These forcing functions are directly compatible with the input requirements of the ME101 program, which was used to perform the structural evaluations.

For the analysis of a system such as the cooling water systems having large vapor pockets, an important requirement of the computer program is that it should be able to model large vapor pockets and their movement/collapse accurately. Traditional water column separation and rejoining logic, in which the vapor bubble size is tracked but the bubble itself is always attached to a node, may not be accurate for a bubble size larger than the distance between two adjacent nodes (Discrete-Vapor-Cavity Model, Section 3-8, Reference 2). One of HSTA's unique features is that it contains computational logic that accurately treats vapor pockets as having two bounding vapor/liquid interfaces that can not only move independently of each other but also allow the vapor pocket to translate along the piping system. The pressure in the void, or the "vapor pressure" is an input to the HSTA run. This pressure can be different for different "links".

The HSTA hydraulic model for the piping network is composed of a set of pipes called "links", each of which is subdivided into two or more "'nodes". The distance between each node is known as a "reach length" or "nodal distance". At the beginning and end of each link is an identified boundary condition such as a valve, a reservoir, pump, etc., or a simple continuation. Links can also end at external boundary conditions such as, reservoirs, dead ends, etc. In order to limit the size of the model and the associated computer analysis time, some of the coolers in the PSW system (other than the drywell coolers) with small flows are omitted from the model. This causes some conservatism in the results. Even then, with a nodal distance of 2 ft, a total of 530 "links" were used to model the Unit 1 PSW system. The portions of the piping where vapor pockets formed were modeled in detail to accurately predict the system filling phenomena. Piping and drywell coolers at low elevations were modeled coarsely in order to reduce the model size without sacrificing accuracy. The vapor pressure in the voided regions during the refilling transient is calculated from heat transfer considerations.

Validation of the HSTA program is based on comparison of HSTA predicted results with mainly experimental or test data. This was supplemented by comparisons to independent numerically predicted results available in literature. Details of the program validation are provided in Reference 3.

The maximum calculated initial column closure velocity is approximately 11 ft/sec, giving a maximum unbalanced pipe run segment force of approximately 5.3 kips in the system.

The HSTA calculations did not model any non-condensable in the voided region of the piping and kept the vapor pressure constant. Therefore, the initial closure velocity as calculated by HSTA does not include any cushioning due to air or steam. Furthermore, it was decided not to take any credit for air/steam cushioning that the EPRI methodology allows (using the cushioned to uncushioned closure velocity charts in EPRI Report TR-1003098,[1] adding conservatism to the analysis. It should be noted that this added conservatism not only comes from a larger calculated peak closure velocity/pressure, but also from the smaller rise time for force calculations.

Prior to implementation of DCR 02-020, Depower Solenoids on PSW Valves (Drywell Cooler Cooling Water Inlet Valves), the original design automatically closed the drywell cooler PSW inlet valves (on restoration of power supply) before the PSW pump restart following a LOCA/LOSP scenario. It was analytically found in this case that the steam bubble extended to upstream of this valve. The collapse of this steam bubble trapped by the closed valve upon pump start gave a much larger closure velocity as compared to the final case where the isolation valves do not close. The waterhammer loads for the final case were acceptable. Therefore, the PSW inlet valves are prevented from closing in the corrective action to minimize the GL 96-06 waterhammer potential.

Structural/Stress Evaluations

A force-time history analysis was performed on the supply and return side service water piping of the Unit 1 drywell coolers using the ME101 structural analysis program. These analyses considered loads with the PSW inlet valves open using the force-time history computations and peak pressures from the HSTA program. The time history loads were given in 0.0005-sec increments at each HSTA pipe run segment, which generally occurred at each change of piping direction.

The supply and return side piping models were evaluated with model boundaries running between the drywell penetration, the nozzles of supply/return drywell coolers and the boundary anchors separating models. Stiffnesses for supports, anchors and nozzles, were calculated and used in the ME101 structural model. Where one directional supports were determined to have uplift (such as rod hangers), additional analytical iterations were made, removing the supports from the analysis to determine the effect on adjacent piping and supports due to load and stress redistribution. Additionally, where support capacities were exceeded, additional analytical iterations were made without the affected support(s) from the analysis and determining the effect on adjacent piping and supports due to load and stress redistribution. Final pipe stress and support evaluations were based on an envelope of these computer runs.

The service water piping for the drywell coolers is classified as ANSI B31.7. Piping stresses were well under the $2.4S_h$ stress limit due to waterhammer loads, deadweight and the peak pressure seen during the water hammer transient. Nozzle loads were evaluated by limiting pipe stress to 12,000 psi adjacent to the nozzle. This was the original design basis faulted allowable used in pipe stress calculations. All applied nozzle loads were determined to be well below this limit.

Pipe supports were evaluated in accordance with the ASME Code, Section III, Appendix F. Where pipe support stresses exceeded these limits, they were removed from the analysis. All remaining supports met the acceptance criteria.

REFERENCES

1. EPRI Technical Basis Report and User's Manual, Final Report No. 1003098, "Generic Letter 96-06 Water Hammer Issues Resolution," dated April 12, 2002.

2. Wylie, E. B. and Streeter, V. L. "Fluid Transients in Systems," Prentice Hall, 1993.

3. SNC letter HL-5708 to NRC, "Updated Response to Request for Additional Information, Generic Letter 96-06 - Water Hammer in Containment Coolers," dated November 20, 1998.