CP&L
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SERIAL: HNP-02-142

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SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/LICENSE NO. NPF-63

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING GENERIC LETTER 96-06, "ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS"

Ladies and Gentlemen:

By letter dated May 16, 2002, the NRC provided Carolina Power & Light Company (CP&L) a copy of the evaluation titled, "Evaluation of Electric Power Research Institute Report TR-113594, 'Resolution of Generic Letter 96-06 Waterhammer Issues,' Volumes 1 and 2 dated December 2000." The NRC letter also requested CP&L to complete actions to address Generic Letter (GL) 96-06 and to submit the information referred to in Section 3.3 of the NRC evaluation.

The actions to address GL 96-06 have been completed. Attachment 1 provides the response to the request for additional information (RAI) referred to in Section 3.3 of the NRC evaluation. Attachment 2 provides the response to the previous request for additional information regarding two-phase flow. Attachment 3 provides the simplified diagrams of affected systems.

Please refer any questions regarding this submittal to Mr. John Caves at (919) 362-3137.

Sincerely,

John S. Keenan

John S. Keenan

A072

Harris Nuclear Plant
5413 Shearon Harris Road
New Hill, NC 27562

JSK/jpy

Attachments:
1. Response to the Request for Additional Information Referred to in Section 3.3 of the NRC Evaluation
2. Response to the Previous Request for Additional Information Regarding Two-Phase Flow
3. Simplified Diagrams of Affected Systems

John S. Keenan, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge and belief, and the sources of his information are employees, contractors, and agents of Carolina Power & Light Company.

Carolyn A Rice

My Commission Expires
June 21, 2004

Notary (Seal)

My commission expires:

c: Mr. J. B. Brady, NRC Sr. Resident Inspector
Mr. R. Subbaratnam, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

ATTACHMENT 1 TO SERIAL: HNP-02-142

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1

RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION
REFERRED TO IN SECTION 3.3 OF THE NRC EVALUATION

NRC Request for Information:

Certify 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 uncushioned velocity and pressure are more than 40 percent greater than the cushioned values, also certify that the pipe failure probability assumption remains bounding.

Response:

The recommended waterhammer evaluation methodology is contained in the Electric Power Research Institute (EPRI) final report 1006456 (formerly EPRI TR-113594), and consists of the following analysis steps, summarized from Section 2.2.

1) Evaluate System:
   a) Gather Data
   b) Determine Limiting Plant Configuration
2) Model System Hydraulics:
   a) Determine Fan Cooler Unit Performance
   b) Determine System Voiding
   c) Determine System Refill
3) Determine Condensation Induced Waterhammer (CIWH) Magnitude
4) Determine Potential Closure Locations
5) Determine Column Closure Waterhammer (CCWH) Magnitude and Pulse Characteristics:
   a) Determine Column Closure Velocity Limited by Inertia
   b) Determine Released Non-Condensables
   c) Determine Refilling Velocity Limited by Cushioning
   d) Determine CCWH Magnitude and Pulse Shape
6) Determine Pressure Pulse Propagation and Pipe Loading:
   a) Determine Loading Functions (Force-Time Histories)
   b) Determine Pulse Amplification and/or Attenuation
7) Determine Pipe Stress and Support System Loads

Analysis Steps 1 and 2, including subtasks, were performed in a manner consistent with or more conservative than the EPRI methodology, as documented in vendor calculation 1133948-C-001 from ABS Consulting. Since the original analysis was performed before the final approval of the EPRI methodology, the inputs used in the analysis were subsequently evaluated in the vendor calculation and were found to produce conservative results.

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Response (continued):

An additional plant specific task described in EPRI final report 1006456, Analysis Step 2, Subtasks b and c (Determine System Voiding and Refilling), requires that plants review the piping for outliers to the normal analysis, including voided vertical risers, dead legs, orifices in the voided region, and refilling Froude number. These outliers were specifically reviewed in vendor calculation 1133948-C-001 from ABS Consulting, and no outliers were found that would impact the analysis in vendor report 96191-TR-02 from Altran Corporation.

The Harris Nuclear Plant (HNP) meets the EPRI criteria permitting Analysis Step 3 (Determine Condensation Induced Waterhammer (CIWH) Magnitude) to not be explicitly analyzed. Although not required, CIWH magnitudes were explicitly analyzed for HNP in vendor report 96191-TR-02 from Altran Corporation. Comparison of the system conditions with the EPRI CIWH waterhammer criteria is shown in vendor calculation 1133948-C-001 from ABS Consulting. Specifically, the system steam pressure at the time of the postulated CIWH is less than 20 psig and the piping has been shown by test and analysis to be capable of withstanding a Column Closure Waterhammer (CCWH) following a Loss of Offsite Power (LOOP).

Analysis Steps 4 and 5, including subtasks, were performed in a manner consistent with or more conservative than the EPRI methodology, as documented in vendor calculation 1133948-C-001 from ABS Consulting. Since the original analysis was performed before the final approval of the EPRI methodology, the inputs used in the analysis were subsequently evaluated in the vendor calculation and were found to produce conservative results.

As a conservative measure, Analysis Step 5 did not take credit for cushioning of the closure velocity. Therefore, the uncushioned velocity and pressure are equal to the cushioned values, and the pipe failure probability assumption remains bounded.

Analysis Steps 6 (Determine Pressure Pulse Propagation and Pipe Loading) and 7 (Determine Pipe Stress and Support System Loads) were satisfied using previous test data and stress analysis to show system acceptability. Specifically, tested (LOOP only) waterhammers did not result in observed piping or support damage, and the developed pipe stress and support loads were found to be acceptable. Since LOOP only waterhammer magnitudes bound the waterhammers predicted for a Loss of Coolant Accident (LOCA)/LOOP, the system is qualified for LOCA/LOOP.

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REFERRED TO IN SECTION 3.3 OF THE NRC EVALUATION

Response (continued):

Plant specific risk of GL 96-06 induced pipe failure is much lower than the EPRI risk perspective provided in an EPRI letter dated February 1, 2002, as described in the NRC Safety Evaluation Report (SER). Specifically, the plant risk of a combined LOCA/LOOP is 4.82×10^{-6}. The SER value for the likelihood of subsequent pipe failure is 1×10^{-4}, based on analyzed pipe stress levels meeting the design code. These values produce a combined risk of 4.82×10^{-10}, which is significantly less than the EPRI risk level of 1×10^{-7}. Therefore, the plant risk is bounded by the EPRI risk perspective.

NRC Request for Information:

Provide 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).

Response:

A request for additional information (RAI) with respect to the GL 96-06 two-phase flow issue was provided in NRC letter dated June 24, 1998. The RAI items from the June 24, 1998 letter that pertain to two-phase flow are addressed in Attachment 3 below.

NRC Request for Information:

Provide 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.

Response:

The analyses performed for Shearon Harris, Unit 1, provided conservative evaluations of waterhammer and two-phase flow conditions in the Service Water (SW) system, during the draining, refill, and post-refill periods of the transient.

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REFERRED TO IN SECTION 3.3 OF THE NRC EVALUATION

Response (continued):

During the initial, draining portion of the transient, voids will form in elevated portions of the supply and discharge piping adjacent to the Containment Fan Coolers (CFCs). Heat transferred from the LOCA environment will boil water in the CFC tubes and steam will pass through the CFC headers to pressurize the voids in the supply and discharge side piping. Condensation Induced Waterhammers (CIWHs) are expected to occur; however, the system conditions meet the EPRI criteria, showing that the bounding system loads will be due to Column Closure Waterhammer (CCWH). Therefore, CIWHs do not require specific evaluation.

Upon pump restart and steam void re-closure, CCWH is expected to occur. The LOOP with LOCA waterhammer magnitude is bounded by LOOP without LOCA waterhammer magnitudes due to the LOCA conditions causing the release of non-condensable gas and pressurization of the void. Both of these effects will help to moderate the column closure impact and resulting waterhammer pressure. The limiting LOOP alone waterhammer pressure magnitude is conservatively predicted to be 400 psig. The pipe stress and support loads produced by tested and predicted waterhammers were found to be acceptable. Steps were also taken to prevent the LOOP waterhammers from occurring in the future by modifying operation procedures to keep the SW system from draining during simulated LOOP tests.

After void re-closure, hot water generated in the fan cooler will travel to restricting orifices and outlet control valves (1SW-119 and 1SW-120). Two-phase flow analysis shows that the water passing through the orifices will not cause flashing, but a brief period of flashing can occur at the outlet control valves. This flashing duration will last approximately five seconds and will be completed within 100.5 seconds from the start of the transient.

The potential for the flashing to degrade containment heat removal does not affect the design basis heat removal of the CFCs, since no credit is taken for heat removal until 110 seconds into the transient. Any potential two-phase flow generated in the SW system will stop before the system's heat removal capability is required. Therefore, the effects of the two-phase flow are not significant to the design basis operation of the SW system.

The corrective actions are complete.

ATTACHMENT 2 TO SERIAL: HNP-02-142

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RESPONSE TO THE PREVIOUS REQUEST FOR ADDITIONAL INFORMATION REGARDING TWO-PHASE FLOW

NRC Request for Information:

1. 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. Also, explain why this methodology is applicable and gives conservative results (typically accomplished through rigorous plant-specific modeling, testing, and analysis).

2. For both waterhammer and two-phase flow analyses, provide the following information:

   a. Identify any computer codes that were used in the waterhammer and two-phase flow analyses and describe the methods used to benchmark the codes for the specific loading conditions involved (see Standard Review Plan Section 3.9.1).

   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. 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 waterhammer types and water slug scenarios should be considered, as well as temperatures, pressures, flow rates, load combinations, and potential component failures. Additional considerations for two-phase flow include:

      - the consequences of steam formation, transport, and accumulation;
      - cavitation, resonance, and fatigue effects; and
      - erosion considerations.

      It is important for licensees to realize that in addition to heat transfer considerations, two-phase flow also involves structural and system integrity concerns that must be addressed. Licenses may find NUREG/CR-6031, "Cavitation Guide for Control Valves" helpful in addressing some aspects of the two-phase flow analyses.

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INFORMATION REGARDING TWO-PHASE FLOW

NRC Request for Information (continued):

d. Confirm that the 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 system and confirm that the FMEA is documented and available for review, or explain why a complete and fully documented FMEA was not performed.

e. Explain and justify all uses of "engineering judgment".

3. Determine the uncertainty 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.

4. Confirm that the waterhammer and two-phase flow loading conditions do not exceed any design specifications or recommended service conditions for the piping system and components, including those stated by vendors; and confirm that the system will continue to perform its design-basis functions as assumed in the safety analysis report for the facility and that containment isolation valves will remain operable.

5. Discuss specific system operating parameters and other operating restrictions that must be maintained to assure that the waterhammer and two-phase flow analyses remain valid, 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 reliance on any non-safety-related instrumentation and controls in this regard.

6. 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.

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

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RESPONSE TO THE PREVIOUS REQUEST FOR ADDITIONAL
INFORMATION REGARDING TWO-PHASE FLOW

Response:

1. This request is not applicable to the two-phase flow issue, so no response is required for this item. However, NUREG/CR-5220 was not specifically used. The methodology evaluating the effects of Condensation Induced Waterhammers (CIWHs) is discussed in Attachment 1 above.

2.a. No computer codes were used in the development of the two-phase flow analysis. Analysis methods employed classical text-book solutions, using spreadsheets for repeated calculations. The spreadsheet formulas and results were individually checked in accordance with the vendor Quality Assurance (QA) program.

2.b. Modeling assumptions for the two-phase flow analysis provide a conservative model of Containment Fan Cooler (CFC) heating, bounding possible input parameters. Specifically, the entire volume of the CFCs was assumed to reach containment temperature by the time of pump restart. Heated water that could potentially provide small amounts of flashing (<2%) at the outlet control valves was conservatively assumed to choke the flow, providing maximum flow restriction. No relevant effects were omitted from the two-phase flow analysis. Paragraph 2.c below provides additional information.

2.c. The goals of the two-phase flow analysis were to determine if flashing could occur at the outlet orifices and outlet control valves, and to determine if the heated water would pass from the system within the 110-second window before heat transfer is credited in the containment analysis. The worst-case scenario for this two-phase flow analysis maximizes the temperature of the fluid, minimizes the pressure at the outlet header, and minimizes the flow. Various system lineups and component failures were considered, and the limiting system alignment was determined to be the result of the following initial condition and assumed component failures:

- The initial condition was for Normal Service Water (NSW) feeding the "A" and "B" trains of containment cooling lined up to return flow to the cooling tower.
- Assumed failure of bus MCC 1B35-SB, which supplies power to valves 1SW-274 and 1SW-275 (cooling tower isolation), 1SW-271 (auxiliary reservoir isolation), and 1SW-40 (NSW isolation).
- Assumed failure of valve 1SW-50 (NSW supply check valve).

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INFORMATION REGARDING TWO-PHASE FLOW

Response (continued):

This scenario resulted in the lowest outlet header pressure. Maximum fluid temperature and maximum flow restriction were achieved by the conservative CFC temperature and choking assumptions described previously. The limiting containment event was determined to be the large-break Loss of Coolant Accident (LOCA) event, which bounds the Main Steam Line Break (MSLB) event, due to the greater moisture content and higher heat transfer to the CFCs.

Given this worst-case model, the two-phase flow analysis determined that flashing only had the potential to occur at the outlet control valves. Any flow limitation due to flashing lasts approximately five seconds, and the heated water passes from the system within 100.5 seconds from the start of the transient. Potential secondary waterhammers due to this flashing were evaluated and determined to be negligible. Potential feedback from the choking limited flow to create another volume of heated water in the CFCs and a secondary flow restriction was also evaluated and determined not to be possible.

After 100.5 seconds from the start of the transient, no other flow restrictions would inhibit normal system operation. Given the brief period of these events, the damaging potential of cavitation erosion or vibration to produce resonance or fatigue is negligible.

2.d. A formal Failure Mode and Effects Analysis (FMEA) was not performed, however, the intent was met in the conservatively-modeled two-phase flow analysis by considering the potential failure of components and the system alignments that would affect the two-phase flow conditions. The assumed component failures and initial condition are as follows:

- The initial condition was for Normal Service Water (NSW) feeding the "A" and "B" trains of containment cooling lined up to return flow to the cooling tower.
- Assumed failure of bus MCC 1B35-SB, which supplies power to valves 1SW-274 and 1SW-275 (cooling tower isolation), 1SW-271 (auxiliary reservoir isolation), and 1SW-40 (NSW isolation).
- Assumed failure of valve 1SW-50 (NSW supply check valve).

This limiting system alignment separates the parallel SW trains and produces the lowest outlet header pressure, increasing the potential for flashing in the outlet control valves.

2.e. Unsupported engineering judgment without reference or justification was not used.

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RESPONSE TO THE PREVIOUS REQUEST FOR ADDITIONAL
INFORMATION REGARDING TWO-PHASE FLOW

Response (continued):

3. Uncertainty was not specifically determined. Instead, conservative assumptions for system alignment, heat transfer, and flow restriction were used in the two-phase flow analysis to bound uncertainty. By assuming the CFCs reach containment temperature during the event, no potential exists for greater heat transfer. By assuming that potential flashing produces choked flow conditions, no potential exists for flow to be more restricted. Therefore, the two-phase flow analysis bounds uncertainty.

4. The two-phase flow analysis confirms that the piping system and components will continue to perform their design-basis functions. The two-phase flow analysis demonstrates that the CFCs satisfy their design-basis heat removal capability regardless of the possibility of flashing at the outlet control valves. Since piping integrity is also shown through the waterhammer analysis, it is confirmed that the system will continue to perform its design-basis function. The CFC containment isolation valves are rugged butterfly valves that will remain operable after a two-phase flow transient.

5. The two-phase flow analysis is based mainly on physical configuration and characteristics that would require a plant modification to alter. The closure times for the NSW supply to Emergency Service Water (ESW) header valves (1SW-39 & 1SW-40) are important to assure isolation of the NSW flow path when ESW is started. Limits have been placed in the quarterly test procedures for these valves to assure that these times are not extended beyond analysis values. Based on these administrative controls, Technical Specification changes would not be appropriate. Non-safety related instrumentation and controls are not relied upon for this analysis.

6. A simplified diagram of affected systems has been provided in Attachment 3.

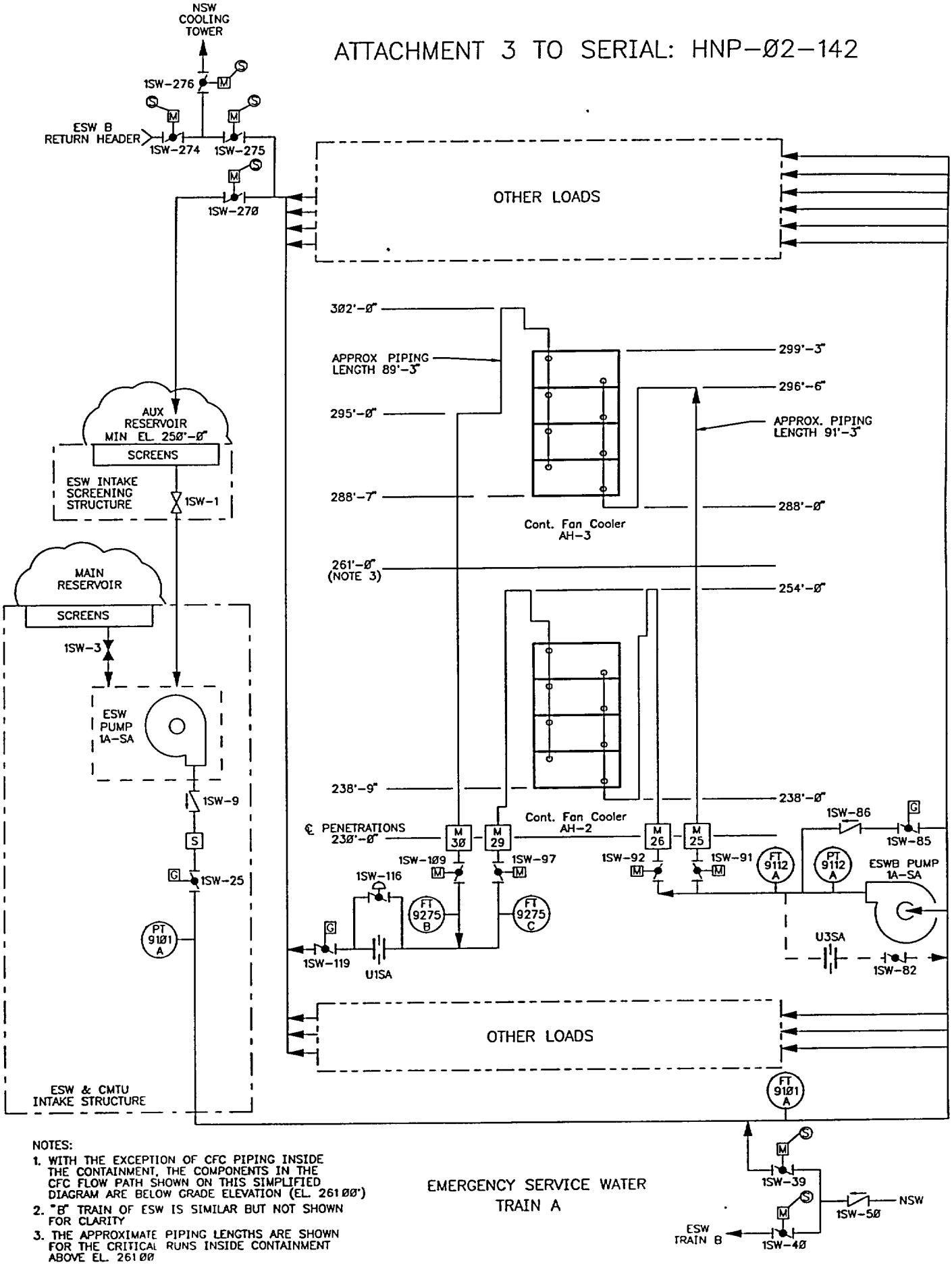
7. Operating procedures have been modified to prevent the system from draining during a simulated Loss of Off-Site Power (LOOP) test. Normal Service Water is used to pressurize the system to ensure that voiding does not occur before the Emergency Service Water (ESW) pumps are started, and to ensure that waterhammers are prevented. No procedure change is required for the two-phase flow issue.

ATTACHMENT 3 TO SERIAL: HNP-02-142

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1

SIMPLIFIED DIAGRAM OF AFFECTED SYSTEMS

ATTACHMENT 3 TO SERIAL: HNP-Ø2-142



NOTES:

1. WITH THE EXCEPTION OF CFC PIPING INSIDE THE CONTAINMENT, THE COMPONENTS IN THE CFC FLOW PATH SHOWN ON THIS SIMPLIFIED DIAGRAM ARE BELOW GRADE ELEVATION (EL. 261 ØØ')
2. "B" TRAIN OF ESW IS SIMILAR BUT NOT SHOWN FOR CLARITY
3. THE APPROXIMATE PIPING LENGTHS ARE SHOWN FOR THE CRITICAL RUNS INSIDE CONTAINMENT ABOVE EL. 261 ØØ

EMERGENCY SERVICE WATER
TRAIN A