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NLS2003027
June 9, 2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Subject: Response to NRC Generic Letter 96-06
Cooper Nuclear Station, NRC Docket 50-298, DPR-46

References:
1. U.S. Nuclear Regulatory Commission (NRC) letter to Nebraska Public Power District (NPPD) dated April 24, 1998, "Request for Additional Information Related to NRC Generic Letter 96-06"
2. NPPD letter NLS980093 to NRC dated June 30, 1998, "Response to NRC Generic Letter 96-06"

The purpose of this letter is to respond to a U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) on Generic Letter (GL) 96-06 for the Cooper Nuclear Station (CNS) transmitted electronically on April 30, 2002. This letter will also provide an updated response to a previous RAI (Reference 1), and updated information regarding how the three issues from GL 96-06 are addressed for CNS.

Attachment 1 is the response to the RAI transmitted electronically by the NRC Project Manager (PM) on April 30, 2002. Nebraska Public Power District (NPPD) responded electronically on May 28, 2002. On June 5, 2002, the PM requested that one part of the response be clarified and that the response be docketed.

Attachment 2 is a revised, complete response to the NRC RAI dated April 24, 1998 (Reference 1). The initial CNS response, submitted by NPPD letter dated June 30, 1998 (Reference 2), stated that boiling in the containment air coolers would not occur. The response further stated that condensation-induced waterhammer and two-phase flow in the containment air coolers were not expected to occur. Therefore, RAI items 1 through 4 were not applicable to CNS since an analysis for waterhammer and two-phase flow was not required. NPPD has subsequently determined that boiling and waterhammer in the air coolers is now a possibility. CNS has completed the required analysis and is providing the information related to the analysis that was requested by Reference 1.

Attachment 3 is a revised response of how the specific issues stated in GL 96-06 have been addressed at CNS. This response supercedes information related to GL 96-06 Issue No. 1 and Issue No. 2 submitted by previous NPPD correspondence identified in Attachment 4. An

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expanded discussion of additional analysis and modifications to finalize the response to GL 96-06 Issue No. 3 is included in Attachment 3. Additional analyses and modifications were required to ensure thermally-induced over-pressurization of isolated water-filled piping sections does not jeopardize the ability of accident-mitigating systems.

Based on the information provided by Attachments 1, 2, and 3, NPPD considers the issues associated with GL 96-06 to be adequately addressed for CNS.

No commitments are made in this submittal. A copy of this letter is being provided to the NRC Region IV office and to the CNS Resident Inspector in accordance with 10 CFR 50.4(b)(1).

If you have any questions concerning this matter, please contact Mr. Paul Fleming at (402) 825-2774.

Sincerely,

Clay C. Warren

Clay C. Warren
Vice President - Nuclear and
Chief Nuclear Officer

/rar

Attachments:

cc: Regional Administrator w/ attachments
USNRC - Region IV

Senior Project Manager w/ attachments
USNRC - NRR Project Directorate IV-1

Senior Resident Inspector w/ attachments
USNRC

NPG Distribution w/o attachments

Records w/ attachments

Affidavit

STATE OF NEBRASKA)
)
NEMAHA COUNTY)

Clay C. Warren, being first duly sworn, deposes and says that he is an authorized representative of the Nebraska Public Power District, a public corporation and political subdivision of the State of Nebraska; that he is duly authorized to submit this correspondence on behalf of Nebraska Public Power District; and that the statements contained herein are true to the best of his knowledge and belief.

Clay C. Warren

Clay C. Warren

Subscribed in my presence and sworn to before me this 9 day of June, 2003.

Wilma M Werner

NOTARY PUBLIC

GENERAL NOTARY - State of Nebraska
WILMA M. WERNER
My Comm. Exp. Oct. 26, 2006

Attachment 1

Response to NRC Electronic Request for Additional Information
Dated April 30, 2002

Cooper Nuclear Station
Nebraska Public Power District

Background Information on System Configuration and Operation

Cooling water to the drywell coolers, or fan coil units (FCUs), at Cooper Nuclear Station (CNS) is supplied by the Reactor Equipment Cooling (REC) System. Drywell cooling by the use of REC flow to the drywell FCUs is a non-essential function at CNS that is not credited for design basis accident mitigation. REC has two divisions with two pumps in each, for a total of four pumps. During normal plant operation one pump in each division is operating in normal mode and the other is in standby mode. Valve REC-MOV-702 is the isolation valve in the REC supply line from the heat exchanger to the drywell, and REC-MOV-709 is the valve in the REC return from the drywell back to the pump suction. The REC pumps de-energize on Loss of Offsite Power (LOOP). Only those pumps in standby will sequentially load onto their respective emergency diesel generator bus after a 20-second delay.

1. NRC Request

   Do the cooling water lines into the drywell for supply and discharge to the drywell coolers auto isolate following a loss-of-coolant accident?

   NPPD Response

   Supply isolation valve REC-MOV-702 automatically closes on low REC header pressure following a 40-second delay. Return isolation valve REC-MOV-709 is controlled from a switch in the main control room and does not automatically close.

   NPPD considers this system configuration to be acceptable with respect to voiding in the FCUs and minimizing waterhammer for a Loss of Coolant Accident (LOCA) concurrent with a LOOP. An analysis for Drywell Unit Cooler Voiding During Small Break LOCA shows very little steam voiding would occur in the first minute before REC flow would be restored. The condition of minimal voiding and restored REC flow at reduced flow would result in negligible waterhammer and no resulting damage to the REC piping.

   During a LOCA without LOOP the REC pumps would continue to operate. In this case there would be no pressure reduction that would result in REC-MOV-702 closure and no potential for waterhammer.

2. <u>NRC Request</u>

Do the drywell coolers trip following a LOCA?

<u>NPPD Response</u>

Yes. The fans for the drywell air coolers shut off on a signal of either high drywell pressure or low reactor vessel level. These signals originate from the initiation logic for the Core Spray System. In addition, as discussed above in the response to NRC Request number 1, a LOCA concurrent with LOOP will result in a decreasing REC pressure and potential closure of REC supply isolation valve REC-MOV-702.

NPPD considers this configuration to be acceptable because the drywell coolers are not required for LOCA mitigation and do not perform a safety function. Shutting the fans off reduces the heatup of the FCU tubes, thereby reducing the likelihood of voiding in the tubes.

3. <u>NRC Request</u>

Will relief valves in the drywell cooling water lines inside the drywell open following a LOCA?

<u>NPPD Response</u>

There is one relief valve (REC-RV-10RV) on the REC piping inside the drywell. The setpoint of this valve is approximately 150 psig. The temperature in the REC piping will ultimately reach equilibrium with the drywell atmosphere temperature. The highest temperature expected (from a small steam line break) is approximately 332°F. The drywell cooling water line return valve does not auto isolate. Therefore, any thermal expansion of the cooling water is relieved to the REC pump suction. The corresponding saturation pressure is approximately 91 psig (106 psia). With a setpoint of 150 psig this relief valve is not expected to open following a LOCA.

4. <u>NRC Request</u>

Original request: Are there provisions for manually restarting the drywell cooling system following a LOCA that take into account possible void formation in the system?

Requested clarification: As appropriate, indicate whether written instructions for preventing waterhammer are (or will be) included in emergency operating procedures (EOPs).

<u>NPPD Response</u>

Voiding in the FCU tubing has been determined by calculation to occur, but that insignificant or no waterhammer will result if restored REC flow to each drywell cooler is maintained less than 390 gpm. Existing data from the latest REC flow-balancing test shows flow less than

390 gpm. In addition, the opening stroke time of approximately 32 seconds for valve REC-MOV-702 will result in an initial slow flow that will slowly sweep away the voids. This relatively slow opening time reduces the likelihood of waterhammer when compared with a sudden valve opening and rapid steam void collapse.

The EOPs specify that drywell cooling is to be operated when average drywell temperature cannot be maintained below 150°F, and that drywell spray is required before average drywell temperature reaches 280°F. Analysis has determined that voiding in the FCUs will occur if drywell temperature reaches 276°F. As a result a caution has been added to the REC System Operating Procedure that restoring REC flow to drywell FCUs with drywell temperature greater than 260°F could result in a breach of FCU tubing.

Attachment 2

Response to NRC Request for Additional Information
NRC Letter dated April 24, 1998

Cooper Nuclear Station
Nebraska Public Power District

Nebraska Public Power District (NPPD) submitted the initial response to the April 24, 1998 Response for Additional Information (RAI) by letter NLS980093 dated June 30, 1998. The letter stated that condensation-induced waterhammer and two-phase flow are not expected to occur, and because an analysis was determined to not be required, the additional information requested in items 1 through 4 is not applicable to Cooper Nuclear Station (CNS). Although that was correct at the time, CNS has since identified a more limiting drywell temperature profile. Subsequent analysis determined that voiding will occur in the Fan Coil Units (FCU) during the worst case of a 1.0 square foot Small Steam Line Break (SSLB) which results in superheated conditions in the drywell. The analysis further determined that voiding will occur in the FCUs during any break that results in drywell temperature greater than 276°F. Because voiding in the drywell coolers is now considered possible, NPPD is submitting a response to items 1 through 4, along with an improved figure requested by item 5.

1. NRC Request

   If a methodology other than that discussed in NUREG/CR-5520, "Diagnosis of Condensation-Induced Waterhammer", was used in evaluating the effects of waterhammer, describe this methodology in detail. Also, explain why this methodology is applicable and gives conservative results for the Cooper Nuclear Station (typically accomplished through rigorous plant-specific modeling, testing, and analysis).

   NPPD Response

   The methodology discussed in NUREG/CR-5220 was used to perform the analysis of the effects of potential waterhammer events at CNS. Section 5.2.3 of the NUREG/CR was determined to be the case most similar to the containment air cooler situation. Complete voiding of the containment fan cooler tubes was assumed. Equipment locations, pump and valve operation and pump and cooler flow rates (air and water) used as input to the analysis were obtained from controlled plant documentation. The method used in the analysis was determined to be conservative since it was assumed that the entire cooling coil was voided prior to the Reactor Equipment Cooling (REC) pump being restarted and that the REC flow was at the design flow rates. Thus, maximum impact loads as determined in Figure C.8 of NUREG/CR-5220 would be felt at the cooling coil tube bends.

2. NRC Request

For both the 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 bench mark the codes for the specific loading conditions involved. (see Standard Review Plan Section 3.9.1)*

NPPD Response

a. The GOTHIC computer code was used by a qualified contractor to determine the extent of voiding in the containment fan cooler tubes. GOTHIC is a fully qualified, industry wide computer code that is used to determine temperature and pressure responses in containment during a Loss of Coolant Accident (LOCA). The containment fan cooler model used in the CNS analysis was similar to that provided in EPRI TR-109612, "GOTHIC Analysis of Containment Fan Cooler Unit (CFCU) Cooling Water Response Following a LOCA with Loss of Offsite Power."

   Benchmarking of the voiding analysis required as part of the design verification process was performed. This benchmarking involved determining that the GOTHIC code output was reasonable with respect to the code input. The code output was determined to be reasonable based on GOTHIC software model test case results and by comparison to previous voiding analyses performed at other utilities, in consideration of:

   (1) The small mass of fluid in the fan cooler tubes that is heating up,
   (2) The amount of time the containment temperature exceeds the saturation temperature of the fluid, and
   (3) The fan cooler configuration (large heat transfer surfaces) as well as general experience in two-phase flow and fluid voiding.

   As part of the contractor's Quality Assurance Program, the GOTHIC code test cases were run and compared against the results provided in the GOTHIC Users Manual to ensure the code was installed and operating properly on the computer platform. The GOTHIC computer code has been benchmarked against numerous test data. The benchmarking and qualification of the GOTHIC code is described in NAI-8907-09, Rev. 6, "GOTHIC Containment Analysis Package Qualification Report," Version 7.0, July 2001. The GOTHIC computer code has been previously used by other licensees in the nuclear industry for safety related applications for various two-phase and single-phase fluid applications.

   Analytical methods provided in NUREG-CR-5220 and ASME Boiler and Pressure Vessel (B&PV) Code Section III, Subsection NC-3600, were used for the waterhammer impact analysis. Because these criteria and methods are comparable to existing CNS analysis guidelines, and because this analysis did not involve a computer program, no benchmarking was required.

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).*

NPPD Response

Assumptions were used in the analysis of voiding in the drywell air cooler FCU units that tended to give results of greater voiding. These assumptions were:

1) A homogeneous temperature profile was assumed for the water contained within the cooling coil tubes. The assumed temperature was the maximum REC temperature of 112°F. This made the fluid in the tube closer to saturation conditions and thus promotes voiding. Actually, the temperature at the cooling water inlet of the tube would be lower. (CNS Technical Specifications specify a limit on the REC supply water temperature of less than or equal to 100°F.)
2) Stagnant fluid conditions in the cooler; no REC pump coast down considered. This allows the fluid to heat up quicker, with no sweeping of any warmer water or voids by the flow.
3) The air temperature throughout the cooler was set at the drywell temperature. This is conservative since there is a time lag for the high containment air temperature to reach all of the cooling coils.
4) All tubes in the fan cooler were assumed to experience the same amount of voiding as the single tube located at the highest point in the fan cooler.
5) A fan coast down time of 60 seconds, based upon industry experience, was assumed since no CNS-specific data was available. Since the results of the voiding showed that the tube is almost completely voided, this coast down time only affects the rate of voiding. The rate of voiding was not used in this analysis.
6) The analysis begins with the REC temperature in the fan cooler already at the limiting (maximum) temperature of 112 degrees. This temperature is conservative in that it is closer to saturation conditions at the start of the analysis and aids in the void formation.
7) Saturated steam conditions in the drywell with no non-condensable gases present are assumed for the duration of the event. This assumption is conservative since any non-condensable gases present will significantly reduce the heat transfer rate.
8) The interior of the drywell air coolers is assumed to be instantly bathed in saturated steam. This assumption is conservative as a finite amount of time will be required for the steam to enter the duct and to travel down to the drywell air coolers, as well as to completely purge all non-condensable gases.
9) The level in the REC surge tank is assumed to be at the low-level alarm setpoint elevation of 982-ft. 9.5-inch. This assumption is conservative since the normal water level is maintained above the low-level alarm setpoint. An alarm sounds in the control room if the level drops below this setpoint.

The following assumptions were used in the waterhammer analysis.

1) REC flow is assumed to be at the maximum steady state value. In reality there will be a gradual buildup of flow due to the flow characteristics of the REC inlet valve during the opening stroke.
2) No cushioning of REC flow by uncondensed steam. In reality some uncondensed steam will remain in the drywell air coolers. This steam will be compressed during the final moments of impact.
3) The front edge of the water slug is assumed to be solid. In reality the front edge of the water slug will be ragged or wavy, and contain entrained vapor. This results in an oblique ("sloppy") impact that is spread out over a period of time. Since the water is eventually brought to rest, large loads may still occur, but they may be attenuated, especially if the impact time exceeds the time for propagation of the pressure waves to and from the liquid.
4) REC design flow rates through the cooler, with no friction or other energy dissipating phenomena acting on the water slug as it travels through the pipe, were assumed in determining the impact force on the cooler tubes. In reality friction or other energy dissipating phenomena will reduce the velocity of the water slug before impact.

Since the cooling coil tubes are relatively small (in both diameter and length) and the waterhammer will occur very soon after a REC pump starts, with no prolonged two-phase condition occurring, no flow induced vibration or erosion different than normal system operation is expected. This is reasonable since the void will collapse very quickly causing the fluid in the coolers and the REC piping to return to a single-phase condition after the waterhammer occurs.

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

Licensees may find NUREG/CR-6031, "Cavitation Guide for Control Valves", helpful in addressing some aspects of the two-phase flow analyses.

NPPD Response

The worst-case containment temperature scenario, a 1.00-square foot small break LOCA, was used to maximize the heat transfer into the cooling coil tubes. This scenario produced the highest temperatures in containment. A Loss of Offsite Power (LOOP) was assumed since this causes the REC pumps and the fans to stop. Also, due to system design, the system outlet valve remains open (i.e., no cooling coil isolation such that solid conditions occur) so that maximum voiding would be able to occur in the system. Also, the highest tube location in the containment air cooler was analyzed for voiding since this presented the lowest static pressure available (lowest saturation temperature) and the most likely location for voiding. A stagnant water condition in the tubes was also assumed since the REC pumps are not running. Fan coastdown was assumed to be 60 seconds to maximize steam entry and the heat transfer to the coils. Based upon the voiding results from the one cooling coil tube case, all tubes in the fan coolers were assumed to be completely voided. Thus, the fluid impact analysis discussed in NUREG/CR-5220, using the design flow rates through the fan coolers, was then readily performed using the conservative peak overpressure formula. Specific conservatisms in this analysis were previously discussed.

The waterhammer scenarios presented in NUREG/CR-5220 were considered with the scenario presented in Section 5.2.3 of the NUREG/CR most closely matching the situation at CNS. Load combinations for the cooling coil stress determinations were based on comparable CNS load case considerations and comparable ASME B&PV Code criteria.

d. *Confirm that the analyses included a complete failure modes and effects analysis (FMEA) for all components (including electrical and pneumatic failures) that could affect the severity of the waterhammer and confirm that the FMEA is documented and available for review, or explain why a complete and fully documented FMEA was not performed.*

NPPD Response

An FMEA was not performed. The analysis showed that no damage will occur due to waterhammer impact, based on the total cooling coil stresses being less than allowable values.

e. *Explain and justify all uses of "engineering judgement."*

NPPD Response

The analyses were performed using industry-accepted analytical techniques and methodologies. Assumptions used in the analyses were documented in the calculation. The only "engineering judgements" used in the analysis are the time of 60 seconds for fan coastdown, and a conclusion that drywell cooling isolation valves would remain operable.

This coastdown time is based upon industry experience with the subject at hand. A parametric study to demonstrate acceptability of assumptions determined that the coastdown time must be extended by more than 50 percent to significantly impact the analysis model predictions.

The conclusion that the drywell cooling isolation valves would remain operable is based on the results of the waterhammer analysis that demonstrates the cooling coil tubes are not expected to be damaged. The carbon steel valve wall thickness is approximately six times greater than the copper cooling coil tube wall thickness. Therefore CNS concludes that the isolation valves will remain operable.

3. NRC Request

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 for the Cooper Nuclear Station.

NPPD Response

Uncertainty analysis was not required in this analysis since it was determined that close to 100% voiding (i.e., worst possible consequences) will occur, and the waterhammer impact analysis was performed using the peak overpressure formula from NUREG/CR-5220. Thus, based on a conservative, worst-case scenario that bounds all others, CNS considered that an uncertainty analysis was not required.

The only areas of uncertainty that could be considered are 1) that of the flow uncertainty when the REC pumps are started, and 2) the maximum REC temperature of 112°F. Using maximum uncertainty values, 1) the REC flow could be greater than the design flow to the coolers and 2) the REC temperature could be greater than 112 degrees.

The first item is bounded by the fact that flow to the coolers would have to be at least 390 gpm during this waterhammer scenario before any tube damage would occur. This is considerably greater than the design flow of 265 gpm. Additionally, the voiding analysis was performed assuming the REC flow to the coolers was zero. Because zero is the lowest possible flow rate no uncertainty in the flow rate is required.

The second item is irrelevant since voiding was determined to be at a maximum. If the temperature were greater than 112 degrees, the maximum voiding would occur sooner. However, since a maximum value of voiding was used as the input parameter in the waterhammer analysis, the temperature uncertainty is not a concern.

4. NRC Request

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 equipment vendors; and confirm that the system will maintain its integrity for all event scenarios and that the system isolation valves will remain operable.*

NPPD Response

The waterhammer analysis performed showed that CNS acceptance criteria for the load combination resulted in material stress allowable values of the fan cooler tubes not being exceeded. The scenario analyzed is considered to be the worst-case and bounding for other scenarios through the assumptions of no flow in the FCU, maximum REC operating temperatures (based on current limits in CNS Technical Specifications), and the highest containment temperature profiles. The analysis also determined the maximum REC flow allowable to the FCUs without causing damage. This analysis showed that damage to the tubes in the FCUs from waterhammer is very unlikely provided that REC flow is less than 390 gpm. Since the cooling coils are not expected to be damaged in this scenario, the system will maintain its integrity. Since the system drywell return valve remains open during this scenario (which allows the voiding to occur), the REC system isolation valves will remain operable once REC flow is reinitiated. With the REC system drywell supply valve closed, and subsequently re-opened, and the consequent collapse of the void in the drywell air cooler fan coil units, the valve will remain operable during the entire scenario. This is based on engineering judgement that if the waterhammer does not damage the cooler tube, it is less likely to damage the valve.

There is an interim period when the emergency diesel generator starts, at least one REC pump starts, and the REC drywell supply valve is energized and begins to close. This results in some slight void collapse, but voiding returns when the valve fully shuts and there is no further REC flow.

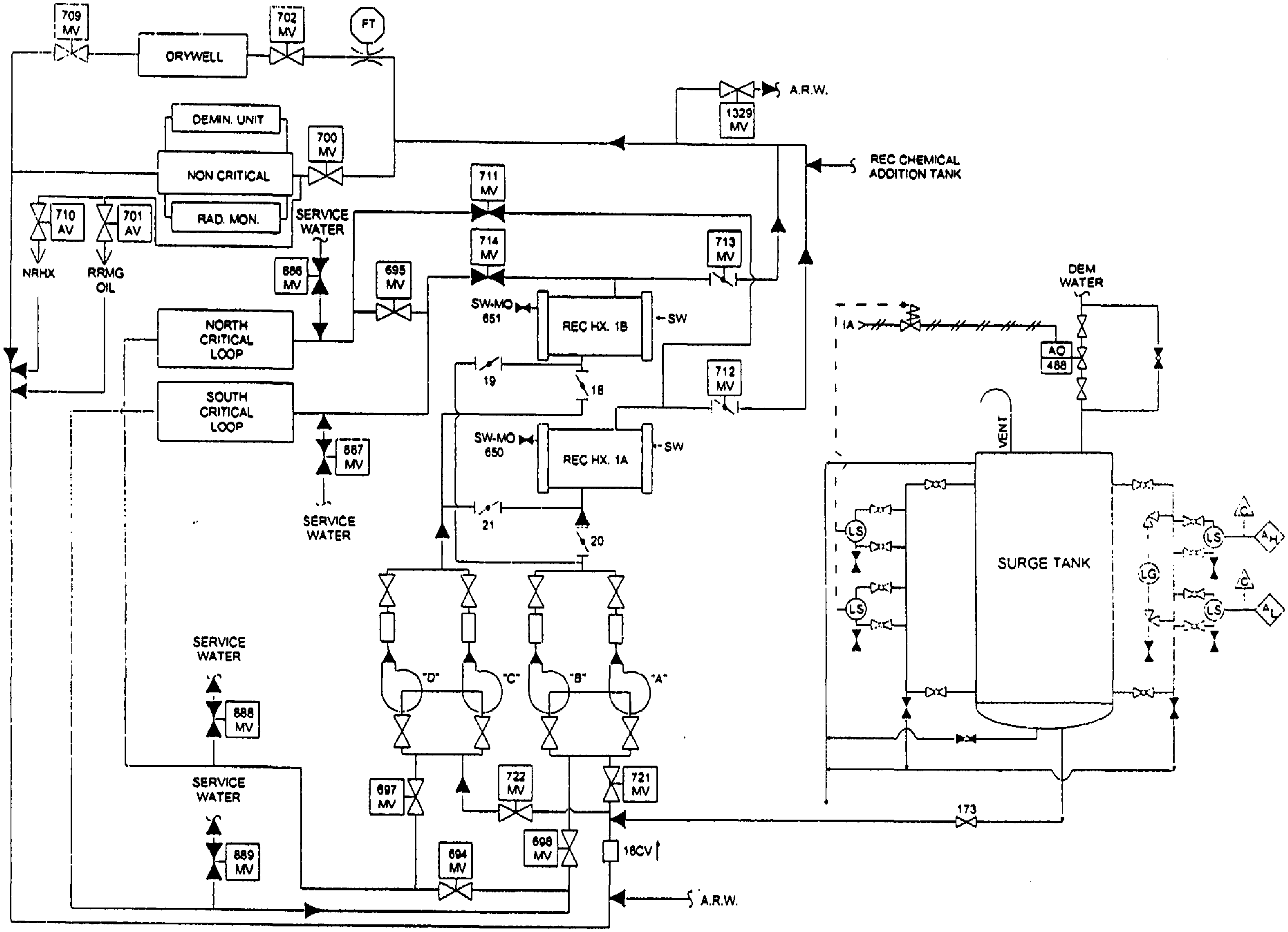
5. NRC Request

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

NPPD Response

A simplified flow diagram of the REC System showing the major and active components is provided as Figure 1. A simplified diagram showing the elevations of the major components is provided as Figure 2. (The box in Figure 2 entitled "Other DW loads" represents loads that are located both above and below the coolers. For simplicity of the figure these elevations are not shown.) The lengths of piping runs are not shown in Figure 2 since CNS considers the lengths of piping runs in the REC System as having little or no impact on the applicable analyses.

FIGURE 1



REC SYSTEM Normal Power Operation

Figure 1, Rev. 11

COR002-19

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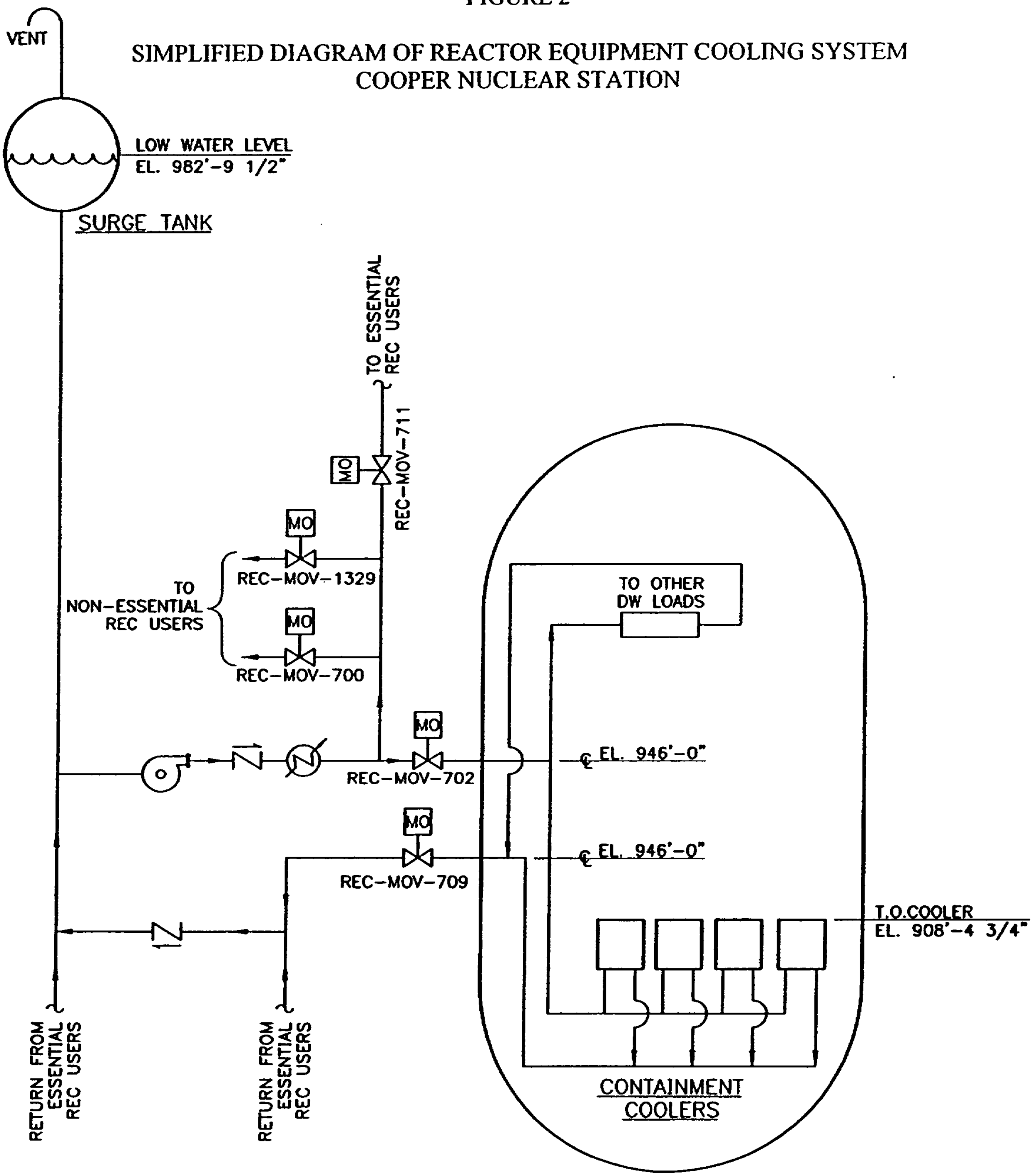
FIGURE 2

SIMPLIFIED DIAGRAM OF REACTOR EQUIPMENT COOLING SYSTEM
COOPER NUCLEAR STATION



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Attachment 3
Response to Generic Letter 96-06

Cooper Nuclear Station
Nebraska Public Power District

The following is an updated summary of how the three issues of concern identified in Generic Letter (GL) 96-06 are addressed at Cooper Nuclear Station (CNS). This summary is provided to ensure that the information available to the NRC staff for their review of how CNS has addressed GL 96-06 is current.

GL 96-06 Issue No. 1

Cooling water systems serving the containment coolers may be exposed to the hydrodynamic effects of waterhammer during either a loss-of-coolant accident (LOCA) or a main steamline 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.

NPPD Response

During normal operation, cooling of the containment (drywell) is provided by four drywell air coolers. Cooling water to the drywell air coolers is provided by the Reactor Equipment Cooling (REC) System. The piping in the REC System is made up of critical and non-critical supply loops:

1. The critical supply loops that serve the safety-related heat loads; and
2. The non-critical supply loops that serve the non-safety-related heat loads.

The drywell air coolers are not required to mitigate the consequences of either a LOCA or a MSLB. Accordingly, the coolers are non-safety-related heat loads and are supplied by the non-critical REC supply loops.

While no credit is taken for the drywell air coolers in mitigating the consequences of a LOCA or MSLB, the susceptibility of the coolers to void formation in such an event was evaluated. The drywell air cooler evaluation assumed the worst case scenario of a 1.00 square foot Small Steam Line Break (SSLB) concurrent with a Loss of Offsite Power (LOOP). (This SSLB, concurrent with a LOOP, is considered to be worst case because it results in the highest temperature in the drywell, and because it is expected that REC flow to the drywell coolers would isolate under these conditions, resulting in voiding. If offsite power remains available REC flow to the drywell air coolers is expected to be maintained with minimal resulting voiding expected.)

The REC System surge tank maintains a static head of pressure on the system. The cooling coil tubes at the top of the drywell air coolers will experience the lowest pressure of the tubes in the cooler. With the surge tank water level at the elevation of the low level alarm, the pressure in the cooling coil tubes at the top of the cooler due to static head will be 46.4 psia. The corresponding saturation temperature is 276°F. Therefore, voiding in the drywell air coolers will occur if the local temperature near the coolers exceeds 276°F.

The drywell air cooler evaluation demonstrated that significant voiding would occur with REC flow to the coolers stopped. This evaluation was performed assuming the REC supply was at the current Technical Specification temperature limit of 100°F. The resulting cooler outlet temperature of 112°F was used in the analysis.

The drywell air cooler evaluation determined that tolerable waterhammer loads would occur if normal REC flow to the coolers was reinitiated with the voiding present. The evaluation further determined that the stresses on the tubes from the waterhammer would be less than allowable stresses. Based on that determination, the evaluation concluded that the waterhammer impact load is acceptable and that no tube failure is expected.

However, to minimize the potential for waterhammer damage, the REC system operating procedure has been revised by adding a caution that restoring REC flow to the drywell air coolers with drywell temperature greater than 260°F could result in a breach of Fan Coil Unit (FCU) tubing.

<u>GL 96-06 Issue No. 2</u>

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.

<u>NPPD Response</u>

No credit is taken for the drywell FCUs in design basis accident (DBA) scenarios. Therefore, CNS concludes that there is no impact on heat removal assumptions or the system design and operability.

As discussed in the above response to GL 96-06 Issue No. 1, analyses has shown that voiding in the drywell air coolers will occur if (1) REC flow to the drywell air coolers is interrupted, as would occur in the event offsite power is lost, and (2) the local drywell temperature in the vicinity of the drywell air coolers reaches or exceeds 276°F. However, this is not considered to be a problem for containment cooling following a DBA since the safety-related containment spray mode of the Residual Heat Removal System, and not the drywell coolers, performs the containment cooling function following a DBA.

CNS Emergency Operating Procedure (EOP) 3A (Emergency Procedure 5.8) directs all available drywell cooling be operated in the event that average drywell temperature cannot be maintained below 150°F, and further directs drywell spray be initiated before average drywell temperature reaches 280°F. This action is intended to avoid the drywell liner temperature reaching or exceeding the design value of 281°F. In the event the EOP was entered, drywell spray would be initiated well before the drywell reaches 280°F. As discussed in the above response to GL 96-06 Issue No. 1, voiding in the drywell coolers will occur only if drywell temperature exceeds 276°F, which is very close to 280°F. Thus, it is expected that drywell spray will have been initiated as directed by the EOP to provide containment cooling. The fans for the drywell air coolers and the reactor recirculation pumps are stopped before drywell spray is initiated. Therefore, any voiding that does occur in the coolers will not interfere with the ability of the CNS safety related systems (i.e. containment spray) to provide sufficient containment cooling.

Furthermore, the evaluation of voiding in the drywell air coolers in response to the worst case conditions of a SSLB concurrent with a LOOP concluded that only minimal voiding might occur in the REC piping connecting to the coolers concurrent with voiding in the coolers. This conclusion is based on:

(1) Boiling will occur first in the drywell air coolers due to thin wall tubing in the fan coil units with enhanced heat transfer capabilities and larger surface area-to-mass ratio. The 3-inch and 8-inch REC piping in the drywell has lower heat transfer qualities than the cooling coils; and
(2) The short time period that the drywell temperature is above the REC saturation temperature.

Additionally, the drywell air coolers and the associated section of REC piping are not assumed to function for mitigation of design basis accidents at CNS and are not safety related.

In summary, voiding in the drywell coolers has been shown to occur if REC flow to the coolers is stopped and the drywell air temperature near the drywell air coolers exceeds 276°F. Because REC flow to the drywell coolers is not credited with providing cooling and could be stopped the impact on containment cooling of two-phase flow in the REC System is not a concern.

GL 96-06 Issue No. 3

Thermally-induced over-pressurization 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 operability requirements.

NPPD Response

Containment penetrations were reviewed to identify piping segments that are normally isolated or that isolate in response to a valid LOCA signal. Only those that are completely filled with water were considered, as the compressibility of either gas or steam will naturally limit the potential pressure increase to acceptable limits. The following are the penetrations identified as susceptible to the affects of thermally induced over-pressurization.

Penetration # (1)	Size (inches)	Description
X8	3	Main Steam Drainline
X12	20	Residual Heat Removal Shutdown Cooling Suction
X14	6	Reactor Water Cleanup (RWCU) Suction
X18	3	Drywell Equipment Sump Discharge
X19	3	Drywell Floor Sump Discharge
X20	4	Demineralized Water Header Supply (2)

Notes

(1) A complete loss of instrument air was assumed in developing this listing due to the fact that the station air compressors are non-essential and will be de-energized following a LOOP.

(2) Manually isolated whenever containment integrity is required.

Background

CNS letter NLS970016, dated January 28, 1997, stated that an evaluation of the structural integrity of the bounding configuration had demonstrated that, while minor yielding (on the order of 4% or less) will occur, continued structural integrity is assured for the piping associated with these six penetrations. CNS letter NLS970058, dated March 27, 1997, stated that of the six penetrations, one (penetration X12) was still being analyzed, and three (penetrations X18, X19, and X20) would be modified to relieve any potential over-pressurization. CNS letter NLS970078, dated May 13, 1997, advised the NRC that additional alternatives were being investigated for X12, and that modification of X18, X19, and X20 were complete. Subsequent analyses, reflecting higher drywell temperatures that would occur following a LOCA or SSLB, show that for the unmodified configurations (penetrations X8, X12, and X14) the maximum strain is approximately 5%. Therefore, the structural integrity of the piping associated with these three penetrations is assured.

Additional analyses and modifications have been performed that demonstrate that none of these penetrations will experience an unacceptable over-pressurization condition. The following discussion explains how each of these penetrations was addressed.

Penetration X8

The primary isolation valves in the piping associated with this penetration are normally maintained with the inboard isolation valve open and the outboard isolation valve closed. The outboard isolation valve is closed at power levels above 50% power. At this power level the pipe and contained main steam drain flow temperatures would be above the maximum drywell temperature expected following a LOCA or SSLB, and therefore, thermally induced over-pressurization is not a concern.

Penetration X12

This penetration had been identified as one that might experience an over-pressurization condition under post-LOCA conditions. CNS letter NLS970078 dated May 13, 1997, stated that installation of a relief valve was under consideration. CNS letter NLS970231 dated February 27, 1998, stated that the modification had been placed on hold until the NRC had made a determination that use of ASME B&PV Code, Section III, Appendix F criteria was acceptable as a permanent resolution of the over-pressurization issue.

A thermal transient analysis and local stress analysis of Penetration X12 was performed using ANSYS, a general-purpose finite element computer software program. The pipe and fluid final peak temperature was determined based on heat transfer, and the pipe internal pressure was calculated based on the thermal expansion at the peak temperature and plastic deformation of the piping. Local stress was determined based on ANSYS finite element model with plastic-elastic expansion analysis adjacent to the penetration flange.

Based on this analysis of Penetration X12, the resulting pipe stress is less than the allowable of 49 ksi based on ASME B&PV Code, Section III, Appendix F.

Penetration X14

The piping associated with this penetration is normally hot during power operation. This is based on the RWCU system typically being in service with the isolation valves open. The post-LOCA or post-SSLB drywell temperature is less than the normal operating temperature of this piping.

Under normal RWCU system operation, the penetration would not experience over-pressurization post-LOCA or post-SSLB. Because RWCU is normally in service for control of reactor coolant system chemistry, this line is not typically isolated for extended periods of time. However, for the infrequent times when the penetration would be isolated for an extended period of time during normal power operation, the temperature of the water between the isolation valves could cool down sufficiently that the piping could be subject to over-pressurization in the event of a LOCA/SSLB. An evaluation based on the previous design basis LOCA demonstrated that over-pressurization would be a concern only if the penetration had been isolated for greater than 27.2 hours at power operation just prior to the LOCA. To address this concern CNS revised the RWCU System Operating Procedure by adding a requirement to ensure the penetration is not isolated for more than 26 hours.

A recent analysis utilizing the higher drywell temperatures expected with the current design basis LOCA/SSLB event indicates that the pipe and fluid could cool down to a temperature below the maximum expected drywell accident environment temperature of approximately 330°F if the penetration is isolated for more than 15 hours. Occurrence of the LOCA/SSLB event could then result in over-pressurization of the penetration.

To prevent this CNS revised the RWCU System Operating Procedure by reducing the period of time that the penetration may be isolated before taking compensatory measures to 15 hours. The procedure now requires flow through the RWCU suction line to be established at least every 13 to 15 hours if the penetration has to be isolated for greater than 15 hours. If flow cannot be established, the procedure contains guidance to ensure one of the isolation valves is in the open position. CNS has concluded that because of the low frequency of times that this line will be isolated for more than 15 hours, this method of mitigation is considered acceptable and is preferable to a hardware modification.

Penetration X18

NPPD had determined that installation of a relief valve was necessary to alleviate any over-pressurization concerns with this penetration. This relief valve has been installed.

Penetration X19

NPPD had determined that installation of a relief valve was necessary to alleviate any over-pressurization concerns with this penetration. This relief valve has been installed.

Penetration X20

NPPD had determined that installation of a spectacle flange and addition of a procedural provision to drain the containment piping was necessary to alleviate any over-pressurization concerns with this penetration. The modification and procedure revisions have been completed.

Conclusion

Based on the evaluations, modifications, and procedure requirements discussed above, the penetrations at CNS are not considered to be susceptible to an unacceptable over-pressurization condition.

Attachment 4

Generic Letter 96-06 Correspondence Submitted to the U.S. Nuclear Regulatory Commission (NRC)

**Cooper Nuclear Station (CNS)
Nebraska Public Power District (NPPD)**

Correspondence previously submitted by CNS as responses to Generic Letter 96-06, the GL Supplement 1, and RAIs, are identified in the following table.

Date of NPPD Letter	NPPD Letter Number	Letter Subject
October 28, 1996	NLS960201	30-day response to GL 96-06. Letter stated NPPD would determine if CNS was susceptible to the problems described in GL 96-06 and inform the NRC of the conclusions within 120 days of the date of GL 96-06.
January 28, 1997	NLS970016	120-day response to GL 96-06. It concluded that containment air cooler cooling water systems are not susceptible to either waterhammer or two-phase flow conditions during postulated accidents, and identified six penetrations as susceptible to overpressurization.
March 27, 1997	NLS970058	Identified how the overpressurization concern for the six penetrations X8, X12, X14, X18, X19, X20 would be addressed.
May 13, 1997	NLS970078	Informed the NRC that no modifications to penetration X8 were needed, that the modifications on penetrations X18, X19, and X20 had been implemented, and a change to the disposition of penetrations X12 and X14.
February 27, 1998	NLS970231	Informed the NRC that calculation had shown that overpressurization of penetration X8 would not occur, that modification of X12 was placed on hold pending NRC determination on ASME Section III, Appendix F, and that NPPD would maintain procedural restrictions for X14 until the NRC makes the determination on Appendix F.
June 30, 1998	NLS980093	This was the response to NRC Request for Additional Information provided by NRC letter dated April 24, 1998. It was concluded that condensation-induced waterhammer and two-phase flow in containment air coolers are not expected to occur, and therefore a response to the specific questions was not needed.

Correspondence Number: NLS2003027

The following table identifies those actions committed to by Nebraska Public Power District (NPPD) in this document. Any other actions discussed in the submittal represent intended or planned actions by NPPD. They are described for information only and are not regulatory commitments. Please notify the NL&S Manager at Cooper Nuclear Station of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	COMMITTED DATE OR OUTAGE
None	