CATEGORY 1

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SUBJECT: Submits response to GL 96-06, "Assurance of Equipment Operability & Containment Integrity During DBA Conditions."

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January 28, 1997

U.S. Nuclear Regulatory Commission
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Subject: Catawba Nuclear Station Units 1 & 2
Docket Nos. 50 -413, 414
McGuire Nuclear Station Units 1 & 2
Docket Nos. 50 -369, 370
Oconee Nuclear Station Units 1, 2, & 3
Docket Nos. 50 -269, 270, 287
Response to Generic letter 96-06: Assurance of Equipment Operability and Containment Integrity During Design-Basis Conditions

Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Conditions," dated September 30, 1996, requested licensees to determine if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions and to determine if piping systems that penetrate containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

Additionally, the GL required that two written responses be submitted. First, within 30 days, licensees were to submit a written response indicating whether or not the requested actions will be completed and submitted within the requested time period and include a discussion of any alternative actions. Second, within 120 days, licensees were to submit a written summary report stating actions taken, conclusions reached relative to the susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurization of piping that penetrates containment, the basis for continued operability of the affected systems and components as applicable, and corrective

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actions that were implemented or are planned to be implemented. If systems were found to be susceptible to the conditions that are discussed in this GL, identify the systems affected and describe the specific circumstances involved.

A written response was submitted by Duke on October 29, 1996 indicating that we would complete the requested actions and submit the requested information within the requested time period.

Duke has completed its evaluation of the issues identified in the GL and determined that the McGuire and Catawba Nuclear Stations containment air cooler cooling water systems are not susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions. The piping systems that penetrate the containment are not susceptible to thermal expansion of fluid so that overpressurization of piping could occur. However, Duke has determined that certain aspects of GL 96-06 do apply to Oconee Nuclear Station. Duke's response describes the results of the analyses for Oconee Nuclear Station and the basis for operability of the affected systems. In addition, commitments to perform additional analyses to determine long-term resolutions for the applicable GL 96-06 issues at Oconee Nuclear Station are described.

The summary reports of the evaluations are provided in Attachment 1 for McGuire Nuclear Station, Attachment 2 for Catawba Nuclear Station and Attachment 3 for Oconee Nuclear Station.

I declare under penalty of perjury that these statements are true and correct to the best of my knowledge.

Should you have questions or need additional information, please contact Allison Jones-Young at (704) 382-3154.

Very truly yours,

M.S. Tuckman

M.S. Tuckman
Senior Vice President
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ATTACHMENTS

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ATTACHMENT 1

McGUIRE NUCLEAR STATION REPORT

McGUIRE

Requested Action 1

Determine if containment air cooler cooling water systems are susceptible to either waterhammer or two phase flow conditions during postulated accident conditions.

Response:

Background

McGuire utilizes four (4) independent ventilation systems to maintain containment temperature within the specified limits detailed in Technical Specifications under normal operating conditions. The affected systems consist of VL (Lower Containment Ventilation System), VR (Control Rod Drive Mechanism Ventilation System), VT (Incore Instrument Room Ventilation System), and VU (Upper Containment Ventilation System). For removal of containment waste heat loads, each containment ventilation system is supplied with raw water cooling from two (2) independent service water systems. The specific systems consist of RN (Nuclear Service Water System) and RV (Containment Ventilation Cooling Water System). Although not a distinct ventilation system, the RN System also supplies raw water cooling to the Reactor Coolant Pump Motor Stator Coolers. These coolers are closed loop operation in which conditioned air is recirculated within the motors only.

None of the affected ventilation systems or service water systems perform a safety-related function. They are designed to maintain containment (or Motor) temperatures within their respective limits during normal operating conditions only. They are not utilized to mitigate the consequences of a design basis accident. McGuire has no safety-related containment cooling or ventilation systems. Peak containment pressure and temperature conditions following a design basis accident are controlled primarily through the passive operation of the Ice Condenser (NF) System in conjunction with active operation of the Residual Heat Removal (ND) System and Containment Spray (NS) System.

All containment ventilation systems at McGuire are non-nuclear safety-related. During a design basis accident (LOCA or MSLB) with loss of offsite power, the associated fans for the containment ventilation systems are tripped which ends all normal operating condition containment cooling. Upon receiving a high-high containment pressure signal, the containment penetrations for the service water systems supply and return headers to the containment ventilation systems (or Motor Stator Coolers) are double isolated.

Under these postulated conditions, the raw water within the service water system piping and cooling coils inside containment is subject to increased temperature due to direct contact with the harsh containment environment under design basis accident conditions. Per UFSAR Figure 6-10 (Lower Compartment Temperature), the maximum projected temperature is approximately 235°F occurring very early in a design basis accident (large break LOCA). At this containment temperature, it is marginally possible to form steam voids within the water solid, isolated service water piping if the pressure within the piping remains at the assumed low value of the service water discharge header ($\cong$ 10 psig) versus the supply header ($\cong$80 psig). This pressure scenario is not likely as the pressure within the stagnant service water piping will increase as a function of containment temperature such that the required saturation temperature at the actual piping internal pressure is greater than the projected containment temperature.

Conservatively assuming conditions are favorable for steam formation and pipe voiding, the potential for low energy waterhammer exists if the service water system is placed back into operation at elevated containment temperatures. To place the raw water cooling systems back into service following a design basis accident, the engineered safety features (ESF) actuation signals which isolated the service water system containment penetration isolation valves must be manually reset. This action to reset ESF signals can be performed by Operations personnel at any time following the initial stages of a design basis accident. However, containment environment conditions and adherence with approved emergency/abnormal operating procedures following a large break LOCA or MSLB, would preclude the necessity for service water system operation in the near term following an initiating event.

Per UFSAR Figure 6-8 (Containment Pressure Transient—Large Break LOCA), projected containment pressure remains above 6 psig for a period exceeding 10^5 seconds (greater than 24 hours) following the initiation of the design basis accident. Again, referring to UFSAR Figure 6-10 (Lower Compartment Temperature—Large Break LOCA), projected containment temperature at 10^5 seconds is approximately 180°F. Therefore, when containment pressure conditions are more favorable for permitting renewed operation of the service water systems, containment temperatures no longer support conditions favorable for steam formation and pipe voiding.

In the event of a small break LOCA or small line rupture, conditions within containment may allow the continued operation or restart of ventilation/cooling water systems. In fact, it may be beneficial to operate these non-nuclear safety related ventilation /cooling water systems as an aid to help reduce containment pressure and temperature. Under the relatively mild containment environmental conditions during these postulated events, heat loads are not sufficient to promote steam void formation. Waterhammer or two-phase flow will not be a legitimate concern should the ventilation/cooling systems be placed back into operation.

In summary, with regards to the "waterhammer and two-phase flow" concern, McGuire's containment ventilation systems and their associated service water cooling systems are not utilized for mitigation of design basis accidents. Since the systems are not in operation (and would not be placed in operation due to strict adherence with emergency/abnormal procedures) during conditions favorable for the formation of steam voids, the waterhammer problem and its potential resultant damage is not applicable to McGuire. No additional actions are required to resolve this problem.

Requested Action 2

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

Response:

The issue of containment penetration thermal expansion over-pressure protection was initially examined for McGuire based on Operating Experience (NRC Daily Reports dated 7/3/96, 8/1/96, 8/2/96 (Beaver Valley), 7/25/96, 7/26/96 (Vermont Yankee) and Information Notice 96-49. Additionally, Section 3.2.8 "Over-pressure Protection" of McGuire's Plant Design Basis Specification for Containment Penetrations (MCS-1465.00-00-0003) was reviewed including references for that section, a 9/9/91 letter (MMSE-91-229), and 3/21/92 follow-up letter (MMSF-02-008) which documented closure of several open items from the previous letter.

Evaluation of Information Notice 96-49

MCS-1465.00-00-0003, Section 3.2.8 states that "Although not required by specific regulatory or licensing commitments, where the potential exists to overpressurize containment penetration piping due to thermal expansion of the trapped fluid in the penetration piping, overpressure protection shall be provided." This criteria was applied to all containment fluid penetrations with relief protection considered necessary for penetrations with normally closed, hard seated valves or with automatic closure gate and/or globe valves both inside and outside containment unless excluded on the following basis. Overpressure protection was considered unnecessary for penetrations which:

a) utilize at least one check valve as an inside (containment) isolation valve through which the pressure increase could flow unimpeded to a relief device located elsewhere along the system, or through soft seated valves per d).
b) contain fluid at a higher temperature than peak containment accident temperature
c) contain air, steam or other compressible gas mixture

d) utilize at least one soft seated valve capable of slight leakage or displacement (diaphragm, butterfly, or plug valves)
e) utilize a single isolation device either inside or outside containment

The Containment Penetrations section of each system's Design Basis Specification or Test Acceptance Criteria (TAC) sheets for relief or check valves providing overpressure protection document necessary overpressure protection requirements for penetrations and associated overpressure protection relief or check valves.

Evaluation of GL 96-06

Penetration Review

To prepare the 120 day response for GL 96-06, a more detailed review of the issue was conducted. A spreadsheet was prepared listing all containment mechanical penetrations. (Spare penetrations were not included on the basis that their closed volume fluid is compressible.) UFSAR Tables 6-111,-112,-113 were used to corroborate the existing design basis position and previous McGuire review performed in response to Catawba Nuclear Station Problem Investigation Report (PIR) 0-C90-0273. Also reviewed were McGuire piping penetration lists on drawings MC-1677-1.1, -1.2,-1.3 as well as procedures OP/1,2/A/6100/23, "Controlling Procedure for Containment Closure," Enclosure 4.6 to ensure the spreadsheet was complete. Each flow diagram was reviewed to verify information contained in the UFSAR tables.

The review challenged the position for over-pressure protection previously concluded for several types of penetrations as follows:

1) Some closed system penetrations were previously exempted from overpressure protection concern on the basis of a single containment isolation valve. This type of penetration was specifically emphasized in the generic letter where the containment isolation valves for two penetrations and the piping system inside (or outside) containment connect them to form a closed volume.
Some penetrations were previously exempted from over-pressure protection concern on the basis that butterfly

(and other resilient seat) valves would relieve any pressure increase above normal because the soft seat material would yield under the higher loading. The generic letter cited specific examples where butterfly valves (assumed soft seated) would not open because of trapped fluid at an increased temperature.

2) Many penetrations appear to have been previously exempted from overpressure protection concern on the basis that a check valve toward containment would allow penetration overpressure relief without documentation of a relief path downstream of the check valve.

The spreadsheet was prepared listing every fluid containment penetration with all considered potentially susceptible to overpressurization. For each penetration, a review was conducted of flow diagrams and operating procedures to develop a basis for overpressure protection for each penetration. The following summary was developed from the spreadsheet:

a) 49 penetrations utilize at least one check valve as an inside (containment) isolation valve through which the pressure increase could flow unimpeded to a relief device located elsewhere along the system. Sixteen of those penetrations utilize a check valve which was installed as a bypass to an existing containment isolation valve specifically for provision of an overpressure relief path.
b) 4 penetrations contain fluid at a higher temperature than peak containment accident temperature
c) 55 penetrations contain air, steam or other compressible gas mixture
d) 5 penetrations utilize at least one soft seated valve capable of slight leakage or displacement (diaphragm valves)
e) 5 penetrations utilize only a single isolation barrier either inside or outside containment
f) 8 penetrations contain relief valves which provide overpressure protection.

Conclusion: The evaluation performed in response to GL 96-06 concludes that sufficient overpressure protection exists for all McGuire penetrations.

Additional Closed Piping Systems Review

Also evaluated were closed systems immediately connected to valves inside containment that are required to open to mitigate the consequences of design basis accidents. This included the PORV Block Valves (1,2NC0031, 0033, and 0035) and Reactor Vessel Head Vent Valves (1,2NC0272, 0273, 0274, and 0275).

Three PORVs exist per unit with a block valve on the NC side and a drain line between the PORV and block valve. Each PORV block valve is normally open to allow the PORV to be available for pressure control when desired. The drain lines are also normally open to remove condensate between the two main valves. In the event a PORV develops a leak, the associated block valve and drain valve are closed which would form a closed volume. The PORVs are not designed for water service, and as such are operated to prevent water in the line. It is possible for the PORV line to become charged with water, however this could only happen with the NC system water solid for a long period of time with the block valve open. When the PORV block and drain valve are closed, the fluid would be primarily air with the possibility of some water due to condensation of leakage. It is therefore not credible that this closed volume would be subjected to over-pressurization due to thermal expansion of water.

Two parallel head vent lines exist on the top of the reactor vessel heads. Each line has two series normally closed solenoid valves with no other valves or branches between them. Since the vessel is normally completely filled with water, it is conceivable that the closed volume between the two solenoid valves would be water solid with leakage through the inboard valve. Because of the valve design (double piloted solenoid operated valves), if the pressure on the downstream side of the inboard valve increased to that of the upstream side, the valve would tend to relieve to the NC system. (Downstream pressure would reduce the seating force on the main seat. Very high downstream pressure would also open the bleed path through the pilots.) The NC system safety valves therefore provide over-pressure protection of this closed volume between the head vent solenoid valves.

Conclusion: The evaluation performed in response to GL 96-06 concludes that sufficient overpressure protection exists for all isolation valves on closed systems inside containment which have a safety related function to open at McGuire.

ATTACHMENT 2
CATAWBA NUCLEAR STATION REPORT

CATAWBA

Requested Action 1

Determine if containment air cooler cooling water systems are susceptible to either waterhammer or two phase flow conditions during postulated accident conditions.

INTRODUCTION

Catawba and McGuire Nuclear Stations are of the Westinghouse Ice Condenser containment design, which results in some major design bases differences when compared to other pressurized water reactor containments. While other, large dry containment pressure suppression systems depend on safety-related fan cooler units, which are subject to loss of cooling flow during the initial stages of a design basis Loss of Coolant Accident (LOCA) and assumed Loss of non-emergency Offsite Power (LOOP), the McGuire and Catawba containments have a passive heat sink at approximately 15 °F, through which the LOCA blowdown energy is quenched. The Ice Condenser design results in an efficient heat removal mechanism that limits containment pressure during the blowdown phase to approximately 7 psig. Peak containment pressure is less than 14.7 psig, and occurs after the ice bed is melted out and pressure is limited by the Containment Spray System.

The ice condenser's primary function is the absorption of thermal energy released abruptly in the event of a loss- of-coolant accident, for the purpose of limiting the initial peak pressure in the containment. A secondary function of the ice condenser is the further absorption of energy after the initial incident, causing the containment pressure to be reduced to and held at a lower level for a period of time. The sodium tetraborate solution produced by a partial melt-down of the ice absorbs and retains iodine released during the accident and serves as a heat transfer medium and neutron poison for cooling the reactor core following the postulated incident. The main part of the ice condenser is a mass of approximately 2,000,000 pounds of sodium tetraborate ice stored in an annular chamber inside the containment shell. The chamber is designed to provide a flow passage between the lower compartment holding the Reactor Coolant System and the upper portion of the containment during accident conditions, and to act as a static, insulated cold storage compartment at all other times. Ten minutes after the initial blowdown, the Containment Air Return Fans are actuated to return the cooled air from the Upper Compartment to the Lower Compartment and continue the circulation of heated air up through the ice bed. There is no cooling water associated with the operation of these fans.

Background

Catawba utilizes the Containment Ventilation (VV) System and the Containment Chilled Water (YV) System during normal plant operations to maintain the upper and lower containment temperatures within Technical Specification Limits. The Containment Chilled Water (YV) System also provides cooling for the Reactor Coolant Pump Motor Air Coolers.

The VV/YV System is composed of the following subsystems:

1. Containment Lower Compartment Ventilation System

2. Containment Upper Compartment Ventilation System

3. Incore Instrumentation Room Ventilation System

4. Control Rod Drive Mechanism Ventilation System

5. Containment Auxiliary Charcoal Filter System

6. Containment Chilled Water System

During normal plant operations, the cooling medium for the lower compartment, upper containment, and incore instrumentation room fan-coil air handling units as well as the reactor coolant pump motor coolers, is the Containment Chilled Water (YV) System. No cooling water is supplied (no heat exchangers exist) for the upper containment return air, containment pipe tunnel booster, and control rod drive mechanism vane-axial fans or the charcoal filter system.

Upon Loss of Offsite Power (LOOP) the cooling medium for the containment heat exchangers swaps from the Containment Chilled Water (YV) System to the Nuclear Service Water (RN) System after a five minute time delay. The Containment Chilled Water and Containment Auxiliary Charcoal Systems are not available on the Blackout Power System. The remainder of the VV System, as described above, is energized upon a LOOP from the Blackout Power System. Throttle valves to miscellaneous coolers fail open to maximize flow to these components and assist in maintaining containment pressure and temperature to provide equipment protection and minimize the likelihood of generating an "artificial" ESF actuation due to a containment pressure increase.

The VV/YV System is Non-Nuclear Safety Related and is not relied upon to mitigate any postulated accidents. No credit is taken for the operation of the VV/YV System in analyzing the consequences of any design basis accident.

Peak containment temperature and pressure conditions following a design basis accident are controlled primarily through the passive operation of the Ice Condenser (NF) System in conjunction with the active operation of the Containment Spray (NS) System, Containment Air Return Fan (VX) System, and the Residual Heat Removal (ND) System.

Response to Requested Action #1 ***(Waterhammer or Two Phase Flow)***

Review of normal operation and licensee postulated accidents and the system response:

Normal Operation

- Containment Chilled Water (YV) System in service.
- If the flow of chilled water to containment is interrupted (due to chiller trip or similar event) the Nuclear Service Water (RN) System aligns to supply cooling water to upper and lower containment headers after a 5 minute time delay.

Evaluation

Normal containment cooling is supplied by the Containment Chilled Water (YV) System. This is a closed loop system which supplies the chilled water pumps with pressurized water at approximately 33 psig (at elevation 596') by the use of a surge/pressure tank. After passing through the chillers, the water is supplied to the upper containment header and lower containment header at a temperature of approximately 45 °F. If the flow of chilled water to containment is interrupted (due to chiller trip or similar event), the RN system will supply cooling water after a five minute time delay.

Catawba has previously evaluated and determined that if Nuclear Service Water (RN) System flow is interrupted to the Upper Containment Ventilation Units (UCVUs), the potential for water column separation on the return header from the UCVUs exists, and the potential for waterhammer exists (Ref. C). The interruption of cooling water flow to the UCVUs happens periodically during inservice stroke testing of the containment isolation valves. A waterhammer analysis program of this transient is documented in Reference D, and the results were evaluated by the stress analysis and support/restraint groups. It was determined that the piping and support system is capable of withstanding the predicted waterhammer loads without failure (Ref C). Also, an inspection was performed on the piping, supports and restraints. No damage or evidence of unusual movement was found (Ref C).

Loss of Offsite Power (LOOP)

- UCVUs, Lower Containment Ventilation Units (LCVUs), and the Incore Instrumentation Room Ventilation Units (IIRVUs) lose power, are restarted after approximately 18 seconds per the Emergency Diesel Generator sequencer.
- Containment Chilled Water (YV) System chillers and pumps lose power.
- after 5 minutes, Nuclear Service Water (RN) System aligns to supply cooling water to upper and lower containment headers.

Evaluation

For the lower containment header the pressure is approximately 38 psig at its highest elevation (considering static head only, Ref. A), with a supply temperature of approximately 45 °F and a saturation temperature of approximately 285 °F (Ref. F). From Reference B, the temperature in lower containment during the Catawba Unit 2 LOOP in February 1996, rose less than five degrees above the normal operating temperature (approximately 110 °F, Tech Spec limit of 120 °F) within the first five minutes of the event. Therefore the stagnant chilled water will not heat up and flash to steam prior to the alignment of the RN System to the lower cooling units, or create a waterhammer event upon the alignment of the RN System to the lower cooling units. After the five minute time delay has elapsed, the RN System is aligned to supply cooling water to the upper and lower containment headers.

For the upper containment header the pressure and temperature is approximately 6 psig at its highest elevation (considering static head only, Ref. A)and 45 °F, with a saturation temperature of approximately 230 °F (Ref. F). From Reference B, the temperature in the upper containment during the Catawba Unit 2 LOOP in February 1996, increased less than five degrees above the normal operating temperature (approximately 90 °F, Technical Specification limit is 100 °F) within the first five minutes of the event, so the stagnant chilled water will not heat up and flash to steam prior to the alignment of the RN System to the UCVUs. When the five minute time delay has elapsed, RN is aligned to supply cooling water to the upper and lower containment headers.

As previously described, Catawba has previously evaluated and determined that if flow is interrupted to the UCVUs, a transient may occur when the system is restarted as described in the "Normal Operation" section of this response.

Loss of Coolant Accident (LOCA), containment pressure $\leq$ 3.0 psig

- Safety Injection signal, (Ss), but no high-high containment pressure signal (Sp).

- UCVUs, LCVUs, and the IIRVUs remain in service.
- The containment penetrations for cooling water to the upper and lower containment headers remain in service, and Containment Chilled Water (YV) System is still in service supplying the upper and lower containment headers.

Evaluation

The containment penetrations for cooling water to the upper and lower containment headers remain in service and the cooling water (YV System) supply to the upper and lower containment headers remains in service. From Reference E, during a LOCA, upper containment temperature remains below 180 °F during the entire accident and lower containment temperature remains below 240 °F during the entire accident. From References A and F , the chilled water supply and return header pressure to the upper and lower containment header is above the saturation pressure of water at 180 °F or 240 °F, therefore, the potential for waterhammer or two phase flow does not exist.

Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), or Feed Line Break with containment pressure ≥ 3.0 psig

- Safety Injection signal, (Ss), and high-high containment pressure signal (Sp)
- UCVUs, LCVUs, and the IIRVUs remain in service.
- The containment isolation valves for the upper and lower containment chilled water headers isolate.

Evaluation

The containment penetrations for cooling water to the UCVUs LCVUs, and IIRVUs isolate.

As previously described, Catawba has previously evaluated and determined that if flow is interrupted to the UCVUs, a transient may occur when the system is restarted as described in the "Normal Operation" section of this response. Review of the Emergency Operating Procedures verified that cooling water to the upper and lower containment headers is not realigned or placed back into service until the plant is ready to return back to normal operation.

Loss of Coolant Accident (LOCA), containment pressure ≤ 3.0 psig, concurrent with LOOP

- Safety Injection signal, (Ss), but no high-high containment pressure signal (Sp)
- UCVUs, LCVUs, and IIRVUs lose power since they are non nuclear safety related loads.
- The containment penetrations for cooling water to the upper and lower containment headers remain in service.
- Containment Chilled Water (YV) System chillers and pumps lose power.
- after a five minute time delay, the Nuclear Service Water (RN) System aligns to supply the upper and lower containment headers.

Evaluation

The containment penetrations for cooling water to the upper and lower containment headers remain in service, but the containment chilled water supply to the upper and lower containment headers is interrupted. After five minutes, the RN System aligns to supply cooling water to the upper and lower containment headers. From Reference E, during a LOCA, upper containment temperature remains below 105 °F during the first fifty minutes of the accident and lower containment temperature remains below 240 °F during the entire accident. During the first five minutes when the chilled water supply is stagnant and RN System is not aligned, the upper and lower containment header pressure (considering static head only) is above the saturation pressure of water at 105 °F or 240 °F (Ref. A and F), therefore, the stagnant chilled water will not heat up and flash to steam prior to the alignment of the RN System.

As previously described, Catawba has previously evaluated and determined that if flow is interrupted to the UCVUs, a transient may occur when the system is restarted as described in the "Normal Operation" section of this response.

Loss of Coolant Accident (LOCA), Main Steam Line Break (MSLB), or Feed Line Break with containment pressure ≥ 3.0 psig, concurrent with LOOP

- Safety Injection signal, (Ss), and high-high containment pressure signal (Sp)
- UCVUs, LCVUs, and IIRVUs lose power since they are non nuclear safety related loads.
- The containment penetrations for cooling water to the upper and lower containment headers isolate.
- Containment Chilled Water (YV) System chillers and pumps lose power.

Evaluation

The containment penetrations for cooling water to the upper and lower containment headers isolate.

As previously described, Catawba has previously evaluated and determined that if flow is interrupted to the UCVUs, a transient may occur when the system is restarted as described in the "Normal Operation" section of this response. Review of the Emergency Operating Procedures verified that cooling water to the upper and lower containment headers is not realigned or placed back into service until the plant is ready to return back to normal operation.

CONCLUSIONS

As a result of the Ice Condenser containment, there is no active, safety related cooling water system function inside containment used in the mitigation of design basis events which input heat to the containment. Also, as a result of the ice condenser containment, the location of the non-safety UCVUs, while above the level of the assured, open service water system discharge to the environment, assures that only cool air from the ice bed is in contact with these cooling coils during the initial stages of design basis accidents which input heat to the containment. Locations of the non-safety coolers in the lower containment (LCVUs, Incore Instrument Room Coolers, reactor coolant pump motor coolers), which are subject to the initial energy blowdown of a LOCA event, are equal to or lower than the environmental discharge elevation, which minimizes the possibility of flashing in these coolers during both Large Break LOCAs and design basis events too small to generate a containment isolation signal. Overpressure protection of the cooling water headers inside containment is discussed in the response to Requested Action 2.

The heat load values as defined in the Catawba UFSAR are not based on single-phase flow assumptions for the containment air cooler cooling water systems. Rather, the heat loads for steam and feedwater line breaks, peak clad temperature analysis, and containment peak pressure analysis are unique to Westinghouse Ice Condenser containments, which do not require safety related, water cooled, containment fan coolers. Accordingly, Catawba has performed evaluations and testing to ensure that the Safety Related System (RN System) connected to the non-safety containment air coolers (and their associated cooling water systems that may be affected by waterhammer or by two-phase flow) are capable of performing their required safety functions and that containment integrity will be maintained.

BASIS FOR CONTINUED OPERABILITY

Operability of the safety related Nuclear Service Water (RN) System has been previously evaluated and determined for the scenario similar to those described above. No new Operability issues were identified.

CORRECTIVE ACTIONS

No corrective actions are required.

REFERENCES

A. CNC-1223.24-00-0046, Nuclear Service Water Equipment Active Isolation Valves.
B. Operator Aided Computer (OAC) data for containment temperature during the CNS LOOP, February 1996.
C. PIP 0-C92-0154, Waterhammer Evaluation for UCVUs
D. CNC-1223.24-00-0039, Waterhammer Analysis for RN Piping for Upper Containment Ventilation Units.
E. CNS UFSAR, Chapter 6 and 15.
F. ASME Steam Tables.

Requested Action 2

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

BACKGROUND

Catawba's original design included the concept of protection of containment penetration piping from gross overpressurization through the use of inside-containment check valves, where possible (incoming lines), and small check valves installed to bypass inside containment valves that automatically close (other lines penetrating containment). This originated from a design criteria, originally specified in the Westinghouse Systems Standard Design Criteria, Nuclear Steam Supply System, Section 1.14, "Containment Isolation", Rev. 2, dated 1/16/73, and was implemented throughout similar containment penetrations on Catawba balance-of-plant systems. This criteria included acceptability of allowing pressurization of penetrations containing soft seat valves or globe valves, which were credited with lifting or leaking slightly to relieve the overpressure condition. Additionally, although the systems standards manual did not so specify, numerous valves downstream of these containment penetrations were specified by engineering to be normally open, normally throttled, or locked open to provide an open relief flow path into the associated system's overpressure protection device inside containment. Operations Procedure valve alignment checklists throughout the systems use the flow diagram locking designations to specify that the required valves are locked by procedure.

It is important to note that even though non-safety systems penetrate containment and have isolation valves that are normally closed, or close on ESF signals, the section of piping and the isolation valves are classified and analyzed as Duke Class B (ANS Safety Class 2). In the case of these non-safety system penetrations, overpressure protection or soft seat valves are provided to allow the expansion of fluid from water solid penetrations and from the non-safety headers inside containment. Further discussion is provided in the following paragraphs.

ACTIONS TAKEN IN RESPONSE TO THE REQUESTED ACTIONS

(2) Containment Isolation Overpressure Protection Issues

Catawba Design Specification CNS-1465.00-00-0003, "Plant Design Basis Specification For Containment Process Penetrations", (Ref 2) contains criteria that represent the Catawba design and licensing position on containment isolation piping. NRC and industry standards are referenced as listed below.

> References "Regulatory Guide 1.141 Containment Isolation Provisions for Fluid Systems, Revision 0, April 1978," "Regulatory Guide 1.11 Instrument Lines Penetrating Primary Reactor Containment, Revision 0, March 1971" and "ANSI N271-1976 Containment Isolation Provision for Fluid Systems" document industry guidelines issued to aid utilities and NSSS suppliers in developing criteria for complying with the quality standards required by 10CFR50. The Duke Power/Catawba philosophy is developed using philosophies presented in these documents.

While some of the statements and figures presented in these standards imply the presence of check valves and bypass check valves to relieve the safety related portions of this piping, none provide this implication for non-safety portions of piping inside containment. The following excerpt from Catawba UFSAR Section 6.2.4.2.2, shows the design and licensing basis criteria for overpressure protection of safety related containment isolation piping.

1. Penetrations requiring relief protection
    a. Penetrations consisting of normally closed, metal to metal seat gate or globe valves both inside and outside containment, unless excluded on some other basis. (See Item 2 below).

b. Same as a. except one of the valves could be a soft seat globe valve.

c. Penetrations consisting of automatic closure gate and/or globe valves (S, T, P, or other safety signal) both inside and outside containment unless excluded on some other basis (See Item 2 below).

d. Penetrations consisting of automatic closure gate valves served by the Containment Valve Injection Water (NW) System (and actuated by S, T, P, or other safety signal) both inside and outside containment.

2. Penetrations for which the overpressure protection feature is considered unnecessary, and thus is not provided include the following:

a. Penetrations including a check valve for the inside isolation valve, through which the pressure increase could flow unimpeded to a relief device located elsewhere along the system, or through soft seated valve(s) per Category 2.d.

b. Penetrations which, during normal operation, contain fluid at a temperature higher than containment peak accident temperature, and thus whose isolation would not result in thermal expansion following the accident. (No penetrations are currently excluded from relief protection on this basis.)

c. Penetrations which, during normal operation, contain air, steam, or other compressible gas mixtures, and thus whose isolation would not result in gross overpressurization due to thermal expansion following the accident. This feature can also be accomplished by draining water or oil filled lines during normal power operation.

d. Penetrations which utilize at least one soft seated valve capable of slight leakage or displacement to accommodate thermal expansion following the accident. For instance, this feature is attributed to diaphragm, butterfly, and soft seat plug valves. Diaphragm and butterfly valves are prone to leak-by at high differential pressure, while plug valve leakage may be a combination of leak-by, stem leakage, and plug displacement.

e. Penetrations which utilize a single isolation device either inside or outside containment and thus are incapable of being overpressurized.

f. Equipment Decontamination (WE) piping judged to be adequate to withstand heatup due to 8015 psia design pressure.

The Containment Penetrations sections of each system's Design Basis Specification list and categorize all penetrations associated with that system, providing adequate justification, as required.

In order to respond to the Requested Action item 2 of this GL, a new evaluation has been conducted to ensure that all containment isolation piping has been properly classified using earlier criteria and evaluations, and that all liquid filled containment isolation piping has been protected per the intent of ASME Code Section III requirements. This new G.L. 96-06 review has considered the additional failure modes that have been identified in the examples provided in the G.L. One area which was further evaluated is the issue of those penetrations that connect to closed piping systems inside containment, in order to ensure that the penetration piping and valves will not be adversely affected by expansion of fluid throughout these systems. The following lists identify the pressure relief path for each penetration and connected piping inside containment.

CONCLUSIONS

All safety related systems penetrating containment have been found to be adequately protected against overpressure transients resulting from a postulated LOCA or Steam Line

Break. The following paragraphs summarize the results of the G.L. 96-06 review.

The following penetrations, designated as Overpressure protection criteria 2.a (Ref 1 & 3), contain incompressible fluid, but do not require additional overpressure protection because they have an inside containment isolation check valve, through which the pressure increase could flow unimpeded to a relief valve inside containment:

Item No.	Pen. No.	Description	Design (psig)	Relief valve (psig)
5	M273	PZR Aux Spray Transient Line	2485	2485
6	M330	NV Charging Line	2685	2485
8	M343	NC Pump Seal Inj Wtr A Supply	2735	2485
9	M339	NC Pump Seal Inj Wtr B Supply	2735	2485
10	M344	NC Pump Seal Inj Wtr C Supply	2735	2485
11	M350	NC Pump Seal Inj Wtr D Supply	2735	2485
12	M259	Reactor Makeup Wtr Flush Hdr	150	150
15	M373	Ice Cond Glycol Pmps Disch	150	150
21	M351	NV Pmp Inj Line to Cold Legs	2735	2485
24	M207	ND Xover Disch to Hot Legs	2485	2485
25	M320	NI Pmp B Disch to Hot Legs	2485	2485
26	M317	NI Pmp A Disch to Hot Legs	2485	2485
27	M336	ND HX A Disch to Cold Legs	2485	2485
28	M307	ND HX B Disch to Cold Legs	2485	2485
29	M352	NI Pmps A&B Disch to C.L.	2485	2485
35	M362	Containment Spray Line	200	(open to Cont.)
36	M370	Containment Spray Line	200	(open to Cont.)
37	M380	Containment Spray Line	200	(open to Cont.)
38	M387	Containment Spray Line	200	(open to Cont.)
39	M369	ND Containment Spray Line A	175	(open to Cont.)
40	M381	ND Containment Spray Line B	175	(open to Cont.)
57	M376	Comp Cool to NCDT HX	135	135
59	M328	Comp Cool to RV & NCP Coolers	135	135
64	M240	NSW to NC Pmp & Lower CVUs	150	150
66	M385	NSW to Upper CVUs	150	150
108	M253	Cont Vlv Inj Wtr A Train	150	150
109	M243	Cont Vlv Inj Wtr B Train	150	150
110	M228	Standby Makeup Pmp Disch	2735	2485

(Trapped volume in pump/closed valves has capability to expand against Nitrogen precharge in suction and discharge pulsation dampers)

Total number of penetrations = 28 x 2 Units = 56

The following penetrations, designated as Overpressure protection criteria 1.a, 1.b, 1.c, or 1.d, (Ref 1 & 3), contain incompressible fluid, and include either a relief valve or a bypass check valve around the inside containment isolation valve for overpressure protection. The former relieves thermal expansion directly, the latter provides a check valve, through which the pressure increase could flow unimpeded to a relief valve inside containment, which is also identified in the following list:

Item No.	Pen. No.	Description	Design (psig)	Relief (psig)	valve No.
4	M347	NV Letdown Line	600	600	NV14(RV)
7	M256	NC Pump Seal Wtr Return	150	150	NV87(RV)
16	M372	Ice Con Glycol Pmp Suct	150	(open to Cont.)	
				and	NF235(CV)
23	M322	NI Test Line	2735	2735	NI481(RV)
				and	NI471(CV)
42	M345	NCDT HX Discharge	150	25	WL462(RV)
				and	WL806(CV)
43	M221	Vent Unit Cond. Drn Hdr	150	(3 psig Loop Seal/ then open to Cont.)	
				and	WL868(CV)
44	M374	Cont. Fl & Eq Sump Pmp	150	35	WL826(RV)
				and	WL321(CV)
45	M359	SG Drain Pmp Discharge	150	150	WLA33(RV)
				and	WLA22(CV)
50	M235	PZR Sample	2735	2485	(PZR SRVs)
				and	NM424(CV)
51	M310	NC Hot Leg Sample	2735	2485	(PZR SRVs)
				and	NM425(CV)
52	M236	NI CL Accumulator Sample	700	700	NM69(RV)
53	M335	SG A Sample	1435	1175	(SG SRVs)
				and	NM426(CV)
54	M338	SG B Sample	1435	1175	(SG SRVs)
				and	NM427(CV)
55	M340	SG C Sample	1435	1175	(SG SRVs)
				and	NM428(CV)
56	M341	SG D Sample	1435	1175	(SG SRVs)
				and	NM429(CV)
58	M355	Comp Cool from NCDT HX	135	135	KC330(RV)
				and	KC814(RV)
				and	KC280(CV)

60	M321	Comp Cool RV Supp. Cool and NCP Coolers Vent Units	135	135	KC281(RV) and KC279(CV)
63	M323	Comp Cool Drn Sump	135	135	KC814(RV) and KC47(CV)
65	M230	NSW frm NC pmp & LCVU	150	150	RN499 (RV), RN807 (RV), RN815 (RV), RN823 (RV), and RN485(CV)
77	M455	SG A Blowdown	1185	1175	(SG SRVs) and BB52(CV)
78	M142	SG B Blowdown	1185	1175	(SG SRVs) and BB53(CV)
79	M3105	SG C Blowdown	1185	1175	(SG SRVs) and BB54(CV)
80	M277	SG D Blowdown	1185	1175	(SG SRVs) and BB55(CV)

Total number of penetrations = 23 x 2 Units = 46

The following penetrations, designated as Overpressure protection criteria 2.e, (Ref 1 & 3), contain incompressible fluid, but do not require additional overpressure protection because they utilize a single isolation device either inside or outside containment and thus are incapable of being overpressurized:

Item No.	Pen. No.	Description	Design (psig)	Relief valve (psig)
19	M276	ND Pump Suct A from Loop (note overpressure between double valves is relieved via ND116(CV) back to Hot Leg)	525	450 ND3(RV)
20	M315	ND Pump Suct B from Loop (note overpressure between double valves is relieved via ND117(CV) back to Hot Leg)	525	450 ND38(RV)
30	M303	Cont. Sump recirc line A	50	(open to Cont.)
31	M210	Cont. Sump recirc line B	50	(open to Cont.)
47	C354	Fuel Transfer Tube	50	(Fuel transfer valve KF122 is normally open to fuel pool as a supply to the Standby Makeup pump, and Flange on containment side has double O-rings for Containment Integrity.)
61	M218	Comp. Cool to Excess LD HX	135	135 KC313(RV) and KC814(RV)
62	M217	Comp. Cool from Excess LD HX	135	135 KC313(RV) and KC814(RV)
83	M110	Feedwater A	1185	1175 (SG SRVs)

84	M262	Feedwater B	1185	1175 (SG SRVs)
85	M309	Feedwater C	1185	1175 (SG SRVs)
86	M422	Feedwater D	1185	1175 (SG SRVs)
87	M143	Aux. Feedwater A	1385	1175 (SG SRVs)
88	M278	Aux. Feedwater B	1385	1175 (SG SRVs)
89	M3106	Aux. Feedwater C	1385	1175 (SG SRVs)
90	M457	Aux. Feedwater D	1385	1175 (SG SRVs)
91	M113	Main Steam A	1185	1175 (SG SRVs)
92	M261	Main Steam B	1185	1175 (SG SRVs)
93	M393	Main Steam C	1185	1175 (SG SRVs)
94	M423	Main Steam D	1185	1175 (SG SRVs)

Total number of penetrations = 19 x 2 Units = 38

The following penetrations, designated as Overpressure protection criteria 2.c, contain air, steam, or gas during normal operation, and therefore do not require additional overpressure protection because they are not subject to thermal expansion of an incompressible fluid. These penetrations are listed below for completeness.

Item No.	Pen. No.	Description	Design (psig)	Contents (fluid)
2	M212	Nitrogen to PRT	100	Nitrogen
3	M327	NC pump Motor Drn Tk Pmp Dis.	50	Air
13	M394	Ice Cond. Ice Blowing Air	15 (flanged)	Air
14	M395, M371	Ice Cond. Blowing Air	15 (flanged)	Air
17	M322	Cont. H2 Purge Inlet Blr Dis	15	Air
18	M346	Cont. H2 Purge Outlet Line	15	Air
22	M331	Nitrogen to CL Accumulators	1085	Nitrogen
32	M301	Spare	15 (flanged)	Air
33	M141	Spare	15 (flanged)	Air
34	M234	Spare	15 (flanged)	Air
41	M348	NCDT Gas Space to WDT	150	Air
46	M356	Equipment Decon Line	8000	Air
49	M377	Refueling Cavity Fill Line	50	Air
67	M308	Spare	15 (flanged)	Air
68	M213	Incore Inst. Room Purge In	15	Air
69	M140	Incore Inst Room Purge Out	15	Air
70	M456	Upper Compart. Purge Inlet	15	Air
71	M432	Upper Compart. Purge Inlet	15	Air
72	M357	Lower Compart. Purge Inlet	15	Air
73	M434	Lower Compart. Purge Inlet	15	Air

74	M368	Containment Purge Exhaust	15	Air
75	M433	Containment Purge Exhaust	15	Air
76	M119	Containment Purge Exhaust	15	Air
81	M386	Containment Air Release	15	Air
82	M204	Containment Air Addition	15	Air
95	M316	Interior Fire Protection	150	Air
97	M220	Instrument Air	115	Air
98	M219	Station Air	115	Air
99	M215	Breathing Air	115	Air
100, 101, 102, 103		Cont. Pressure Sensing Lines	15	Air
103, 105		Equipment & Personnel Hatches	15	Air
106	M329	NC pump Motor Drn Tk Pmp Disch.	50	Air
107	M361	Reactor Building Sprinklers	150	Air
111, 112		Cont. Radiation Monitor		Air
113,114,115		ILRT Press. Line		Air
116, 117, 118, 119		Cont. Atmos. H2 Conc. level Xmitter A, B, (in & out)		Air
120		Lower Personnel Air Lock	15	Air
121		Upper Personnel Air Lock	15	Air
122, 122a, 123		Lower PAL Air Supply/Test Lines		Air
124, 124a, 125		Upper PAL Air Supply/Test Lines		Air

Total number of penetrations = 54 x 2 Units = 108

Three containment isolation situations have been identified where the system design pressure (but not piping maximum allowable pressure), may be exceeded, all located on non-safety systems penetrating containment. These cases involve normally closed soft seat plug valves, where no overpressure protection device was required because original criteria 2.a, (Ref 1 & 3), took credit for a combination of seat deformation, seat leakage, or stem seal leakage. All three applications are non-safety related, and the associated system / valves are not required to be used / reopened in any Emergency Procedure. Since these valves are not widely used in the industry and leakage data was not available, tests have been performed to demonstrate that valve leakage occurs prior to reaching Maximum Code allowable pressure.

Item No.	Pen. No.	Description	Design (psig)	Basis/Code Max. Allow Press (psig)
1	M216	PZR Relief Tank Makeup	150	Soft Seat Vlv Lkg/1817

48 M358 Refueling Water Pmp Suct 50 Soft Seat Vlv Lkg/1550
96 M337 Demin. Water 100 Soft Seat Vlv Lkg/1914
(Credit taken for relief of non-safety Demin Water header through Decon. Sink faucets and / or soft seat plug valves)

Total number of penetrations = 3 x 2 Units = 6

<u>Additional Closed Piping Systems Review</u>

Also evaluated were closed systems immediately connected to valves inside containment that are required to open to mitigate the consequences of design basis accidents. This included the PORV Block Valves (1,2NC31B, 33A, and 35B) and Reactor Vessel Head Vent Valves (1,2NC250A, 251B, 252B, and 253A).

Each of three PORVs per unit have a block valve on the NC side, with no loop seal or drain connections between the PORV and block valve. Each PORV block valve is normally open to allow the PORV to be available for pressure control when desired. In the event a PORV develops a leak, the associated block valve is closed which would form a closed volume. The piping is routed off the top of the pressurizer without a loop seal to prevent water in the line. When the PORV block valve is closed, the fluid would be steam off the pressurizer steam space. It is therefore not credible that this closed volume would be subjected to over-pressurization due to thermal expansion of water.

Two parallel head vent lines exist on the top of the reactor vessel heads. Each line has two series normally closed motor operated packless globe valves with a crossover line (not valved) between them. Since the vessel is normally completely filled with water, it is conceivable that the closed volume between the two valves would be water solid with leakage through the inboard valve. Because of the valve design (packless globe valves in which both flow under the seat and a spring assists in opening the valve), these valves are not prone to pressure locking. These valves are opened as part of the LOCA sequence, so it is likely that they will be opened before significant pressurization could occur. Pressure buildup between the valves assists in opening the outboard valve, which would then relieve any trapped pressure between the valves. Redundancy is provided such that failure of any one valve or emergency D/G will not preclude establishing the vent path. Both an A and a B train related valve are provided in each of two parallel paths, with a crosstie between the enclosed volumes such that if one valve sticks closed initially, either train's inboard valve is available to be opened after pressure is relieved by an outboard valve.

Conclusion: The evaluation performed in response to GL 96-06 concludes that sufficient overpressure protection exists for all isolation valves on closed systems inside containment which have a safety related function to open at Catawba.

BASIS FOR CONTINUED OPERABILITY

Containment piping and components have been verified to meet original design criteria. While three penetrations were identified that may be subjected to greater than design pressure as a result of LOCA or MSLB transients, pressure buildup is limited, and is relieved by soft seat valve leak-by, stem leakage, or plug displacement prior to piping exceeding ASME Code maximum allowable pressure. The basis for each penetration's Operability has been adequately demonstrated.

CORRECTIVE ACTIONS

No corrective actions are required as a result of this Generic Letter 96-06 Review.

REFERENCES

1. Calculation CNC-1223.02-00-0016, "Evaluation of Containment Isolation Overpressure Protection Features in Response to PIR 0-C90-0273"
2. CNS-1465.00-00-0003, "Plant Design Basis Specification For Containment Process Penetrations"
3. Catawba UFSAR Section 6.2, "Containment Systems", inclusive of Figures and Tables, e.g., Table 6-77, "Containment Isolation Valve Data" and Figures 6-112 through 6-115, "Containment Piping Penetration Valve Arrangements"
4. "ANSI N271-1976 Containment Isolation Provision for Fluid Systems"
5. Memo to File "Leak Testing of 3" and 4" Plug Valves to Support Catawba's Response to GL 96-06, File No: CN-1205.04

ATTACHMENT 3

OCONEE NUCLEAR STATION REPORT

OCONEE

Requested Action 1

Determine if containment air cooler cooling water systems are susceptible to either waterhammer or two phase flow conditions during postulated accident conditions.

Response:

I. System Description

The Low Pressure Service Water (LPSW) System is an open raw water support system that takes suction from the Condenser Circulating Water (CCW) System crossover line and supplies cooling water to safety and non-safety related loads in the Turbine Building, Auxiliary Building, and Reactor Building. The LPSW (lake) temperature can vary from 40°F to 85°F. Simplified LPSW flow diagrams are shown in Figures 1 and 2. The elevations of the LPSW piping and components of interest are shown in Figures 3 and 4.

Safety related cooling loads include the following:

- Low Pressure Injection (LPI) System coolers (decay heat removal coolers)
- Reactor Building cooling units (RBCUs)
- High Pressure Injection (HPI) System pump motor bearing coolers
- Motor-driven emergency feedwater pump motors
- Turbine-driven emergency feedwater pump bearing jacket cooler

Unit 1 and Unit 2 share three 15,000 gal/min LPSW pumps. The LPSW pumps take suction from the 42 inch crossover line between the condenser inlet headers. Two LPSW pumps are supplied by one suction line and the other pump is supplied by the other suction line. Suction is provided to the LPSW pumps via gravity flow or siphon flow from the CCW System (the emergency or ECCW mode) following a design basis accident if the CCW pumps are not running due to a loss of power.

The LPSW System provides cooling for components in the Turbine Building, the Auxiliary Building, and in the Reactor Building. Two separate 24 inch lines provide LPSW flow to the components in the Auxiliary and Reactor Buildings. These two supply lines are further divided into four separate supply headers, two supplying the components in Unit 1 and two supplying the components in Unit 2. The LPI coolers and the RBCUs are supplied by separate LPSW supply lines. The return lines from the LPI coolers and the RBCUs maintain separation to a point beyond a remote-operated isolation valve.

For Unit 3, each of the two 15,000 gal/min LPSW pumps take suction from the CCW crossover line. These pumps provide cooling water via separate supply lines to engineered safeguards equipment in the Reactor Building and the Auxiliary Building, similar to Units 1 and 2. The return lines from the Unit 3 LPI coolers and RBCUs maintain separation to a point beyond a remote-operated isolation valve.

The Turbine Building requirements for LPSW flow are supplied from other separate headers. The three pumps associated with Units 1 and 2 have a Turbine Building header serving the Turbine Building requirements for Units 1 and 2. The two pumps associated with Unit 3 also have a Turbine Building header to supply the Unit 3 requirements. The LPSW System can provide sufficient flow to the required safety-related loads following a seismic event. Valves LPSW-139 (shared for Units 1 and 2) and 3LPSW-139 are remotely-operated, seismically-qualified valves which can isolate the non-seismic, non-essential header from the safety-related portions of the system.

The three (per unit) RBCUs ("A," "B," and "C") are supplied by individual lines from the separate LPSW supply headers. Each inlet line is provided with a motor operated shutoff valve located outside the Reactor Building (LPSW-16, -19, -22). Similarly, each discharge line from the coolers is provided with a motor operated valve located outside the Reactor Building (LPSW-18, -21, -24). This allows each cooler to be isolated individually.

During normal operation, the "A" and "C" coolers receive throttled flow and provide normal Reactor Building cooling, while flow through the "B" cooler is isolated by valve LPSW-566. Flow through the "B" cooler is diverted through valve LPSW-565 to the four non safety-related Reactor Building auxiliary cooling units (RBACUs) to also provide normal Reactor Building cooling. The RBACUs are not required to operate during any design basis accident. LPSW flow to the RBACUs is automatically isolated (LPSW-565 closes) by a high Reactor Building pressure engineered safeguards (ES) signal, thereby returning full LPSW flow to the RBCUs (LPSW-566 opens). The baseline stroke time for valves 1,2,3LPSW-565,-566 are all greater than 30 seconds. An ES actuation signals the outlet valves (LPSW-18, -21, -24) on the three RBCUs to stroke fully open automatically to assure emergency flow through the RBCUs.

During normal operation, the fan motors associated with RBCU "A" and "C" operate at high speed, and the fan motor associated with the "B" RBCU is off. Upon ES actuation, the fan motors associated with the RBCUs operating at high speed ("A" and "C" units) change to low speed, and the idle unit ("B") is energized at low speed. The ES actuation also results in isolation of the LPSW cooling water supply to the reactor coolant pump motor coolers by closing valves LPSW-6 and LPSW-15.

LPSW flow to the LPI coolers is normally throttled using air-operated valves 1-,2-LPSW-251, 1-,2-LPSW-252 and 3LPSW-404, 3LPSW-405. During a design basis accident involving a loss of instrument air, these valves fail open to their travel stops. Motor-operated valves LPSW-4 and LPSW-5 would be used to throttle LPSW flow to the LPI coolers under these conditions.

The LPSW System provides sufficient flow to the LPI coolers and RBCUs to satisfy heat transfer requirements following a design basis accident with a single active failure. Post-accident Reactor Building cooling by the RBCUs is needed in order to reduce containment temperatures and stay within the equipment qualification temperature profile. The RBCUs are not needed to mitigate the short-term peak pressure response following a LOCA. Analysis and testing have been performed, based on the post-accident alignment, to demonstrate the required LPSW System flow to each engineered safeguard system.

During normal operation, the cooling requirements are supplied by operating one LPSW pump per unit. The LPSW requirement following a LOCA and/or a loss of offsite power (LOOP) can also be supplied by one pump per unit. The spare pump is started by the engineered safeguards (ES) actuation signal to provide redundancy for meeting the single failure criteria. The LPSW pumps are connected to the 4160 volt buses which supply power to engineered safeguards equipment. The emergency power supply is adequate to restart the LPSW pumps upon a LOOP and actuate the standby pump if an ES signal is received.

II. Overview of Scenarios

This section provides a brief overview of the LOOP, LOCA/LOOP and MSLB/LOOP scenarios as they relate to the response of the LPSW System, RBCUs and RBACUs.

Description of LOOP Scenario

The LOOP scenario causes a loss of power to the LPSW pumps, which initiates a rapid flow coastdown. The RBCU and RBACU fans also begin to coast down. The water flowing through the RBCUs and RBACUs stops and water column separation occurs as the piping drains due to gravity and the elevation relative to lake level. Water is captured in all loop seals, but otherwise drains to a level approximately 34 feet above the lake level. A vacuum is drawn to the saturation pressure corresponding to the temperature of the service water at the high point of the piping. Most of the dissolved air comes out of solution during the depressurization of the service water.

Based on the existence of loop seals in the design of the LPSW piping, the service water within the RBCUs and RBACUs is at a high enough pressure to remain subcooled for all non-accident Reactor Building temperatures, such as following a LOOP. Therefore, the heat transfer in the RBCUs and RBACUs is limited to sensible heating of the service water and no steam is produced other than that which results from the column separation. The service water can heat up to a temperature no higher than the Reactor Building temperature.

Whether or not the RBCU and RBACU fans remain in operation or coast down has little significance. After power is restored, the LPSW pump restarts and begins to refill the voided piping.

The water column will be accelerated by the pump to a flowrate based on the hydraulic losses in each LPSW flowpath, eventually returning to the flowrates that existed prior to the LOOP. The backpressure in the high points of the LPSW System will initially be very low due to the vacuum condition. The water-steam/air interfaces in the voided regions will be at approximately the same temperature. As the water column fills each voided pipe section, a water column rejoining waterhammer will occur. The magnitude of the waterhammer will be mainly determined by the velocity of the water column based on the pump flowrate.

Description of LOCA/LOOP Scenario

The LOCA/LOOP scenario causes a loss of power to the LPSW pumps, which initiates a rapid flow coastdown. The RBCU and RBACU fans also coast down. The water flowing through the RBCUs and RBACUs stops and water column separation occurs as the piping drains due to gravity and the elevation relative to lake level. The LOCA actuates the engineered safeguards signal which realigns the LPSW System and the RBCU and RBACU fans as described above. The LOCA also causes a rapid increase in containment pressure and temperature. The rate of heat transfer to the service water in the RBCUs and RBACUs increases significantly, even though the fans are coasting down.

The initial decrease in LPSW pressure due to the pump coasting down, and the increase in service water temperature due to RBCU and RBACU heat transfer, causes flashing in the high points and boiling in all RBCUs and RBACUs. After power is restored, the LPSW pump restarts and begins to refill the voided piping. The water column will be accelerated by the pump to a flowrate based on the hydraulic losses in each LPSW flowpath, including the realignment of the LPSW System on the ES signal, and the backpressure resulting from heat transfer to the service water.

The RBCU fans will also restart and maintain the low speed of operation. The piping will eventually be refilled and the post-accident function of cooling the containment atmosphere will be established. During the refill process the voided regions will collapse due to the repressurization of the LPSW System and condensation of the steam. Waterhammers associated with column rejoining will occur. The magnitude of the waterhammer will be mainly determined by the velocity of the water column resulting from the LPSW pump restart, which must be determined by analysis. The presence of any significant condensation events must also be determined by analysis.

Description of Steam Line Break/LOOP Scenario

The steam line break/LOOP (MSLB/LOOP) scenario can result in the same LPSW response as described above for the LOCA/LOOP scenario. The UFSAR MSLB peak containment temperature analysis exceeds the temperature of the LOCA analysis. Similar to the LOCA scenario, the RBCUs are not required for the short-term mitigation of the containment pressure and temperature response following a MSLB. Although the RBCUs are credited in the UFSAR MSLB containment response analysis, the RBCUs are not necessary to stay within the post-accident equipment qualification containment pressure and temperature envelope.

III. Thermal-Hydraulic Analysis Methodology

The LPSW System response to the LOOP and LOCA/LOOP transients is analyzed with a 330 node RELAP5/MOD3.1 (Reference 1) simulation model. RELAP5 MOD 3.1 is a six-equation two-fluid code with the capability to model the thermal-hydraulic response of the LPSW System for the scenarios of interest.

The RELAP5 model includes all the relevant parts of the Oconee Unit 3 LPSW System and the RBCUs and RBACUs. The structural metal in contact with the service water is modeled. Figures 5 and 6 and Table 1 illustrate the nodalization and describe the model. Figure 6 enhances the detail of the part of Figure 5 enclosed by the dashed line. The detailed RELAP5 nodalization enables an accurate prediction of the two-phase fluid flow and heat transfer during the transients of concern. The hydraulic losses in the RELAP5 model are benchmarked by matching LPSW flow test data which includes flowrates in all relevant piping runs to the various LPSW cooling loads. The LPSW flow test data and RELAP5 model results are as follows:

Flow Path	Plant Data	RELAP5
LPI 3A cooler flow	5900 gpm	5897.9 gpm
LPI 3B cooler flow	5950 gpm	5949.6 gpm
RBCU 3A flow	1500 gpm	1500.9 gpm
RBCU 3B flow	0 gpm	0.0 gpm
RBCU 3C flow	1430 gpm	1430.7 gpm
RB Aux Cooler flow	1140 gpm	1140.8 gpm
Total CC flow	1750 gpm	1750.2 gpm
Main Turbine Oil Cooler	5515 gpm	5515.9 gpm
RCP Motor Coolers	1100 gpm	1101.0 gpm

The RBCUs and RBACUs are modeled as two-sided conductors. The RBCU model heat transfer area is adjusted to match or exceed the design heat transfer rate for post-LOCA containment conditions per vendor data for an unfouled RBCU. This heat transfer rate is 80 Mbtu/hr-sqft-°F at 286°F post-LOCA containment conditions. The RELAP5 RBCU model prediction of post-LOCA heat transfer was also compared to a detailed RBCU model using the GOTHIC 5.0e (Reference 2) code. GOTHIC 5.0e has a sophisticated fan cooler model (Reference 3) that includes all of the heat exchanger design, fluid flow, and heat transfer considerations that are important for simulating the containment side of an RBCU. The purpose of the RELAP5/GOTHIC RBCU comparison was to ensure that the trend of the reduction in heat transfer as the RBCU fan coasts down and the containment atmosphere conditions change is reasonably and conservatively predicted. The RELAP5 RBCU model was confirmed as sufficiently accurate and conservative.

The RELAP5 RBACU model heat transfer area uses the RBCU heat transfer area adjustment factor since no post-LOCA design data exists. The heat transfer on the outside of the RBCUs and RBACUs is a function of the transient boundary conditions representing containment pressure, temperature, and air fraction, and a flowrate based on fan speed. The RBCU fan speed coastdown on loss of power was obtained from the vendor for post-LOCA conditions. The RBACU fan speed coastdown uses the normalized RBCU fan coastdown data. The RELAP5 heat transfer correlation package is used. The LPSW pumps are modeled with the dynamic centrifugal pump model. This provides a mechanistic prediction of the rate of pump coastdown and restart as power to the pump motor is lost and restored. The boundary conditions associated with the LOOP or LOCA/LOOP scenarios, such as LPSW pump and RBCU fan restart time, valve actuation and stroke times, etc., are all modeled.

The RELAP5 LOOP and LOCA/LOOP cases analyzed investigated several scenarios involving different assumptions for key initial and boundary conditions. One and two LPSW pumps in operation were considered. Early and late power restoration delays (LPSW pump and fan restart times) were considered. Low and high lake levels were considered.

IV. Waterhammer Assessment Methodology

The methodology for identifying waterhammer conditions during the LOOP and LOCA/LOOP transients is as follows. This methodology was developed based on References 4, 5, and 6.

1. The RELAP5 model is run to obtain the thermal-hydraulic response of the transient.

2. The results are reviewed to identify all locations in the LPSW System where significant two-phase conditions evolve.

3. The void is evaluated to determine the potential for condensation-induced waterhammer according to the following procedure:

   a. Only horizontal nodes with significant voiding need to be considered.

   b. Determine any nodes with the slug flow regime during draining.

   c. For those nodes identified in (a), determine those nodes with liquid subcooling >2°F. Any such nodes must be evaluated further for the potential for and magnitude of a condensation-induced waterhammer.

   d. Determine any nodes with counter-current flow following pump restart.

e. For those nodes identified in (d), determine those nodes with liquid subcooling >2°F while the conditions in (c) exist.

f. For those nodes identified in (e), determine those nodes with a Froude number of less than 0.5 while the conditions in (e) exist. Any such nodes must be evaluated further for the potential for and magnitude of a condensation-induced waterhammer.

4. The void is evaluated to determine the magnitude of column rejoining waterhammer according to the following procedure:

a. Identify maximum liquid phase velocities with zero void fraction in all nodes. This maximum velocity can result from either steam expansion accelerating a liquid slug (if such a slug exists) or due to the refilling of the system following LPSW pump restart.

b. Calculate the column rejoining waterhammer overpressure.

V. Thermal-Hydraulic/Waterhammer Results

This section describes the results of the thermal-hydraulic analyses and waterhammer evaluation for the LOOP and LOCA/LOOP events.

LOOP Results

With the initial conditions prior to the LOOP, two-phase conditions exist in the high point downstream of the discharge of the RBCUs. The flow returns to single-phase as it flows vertically downward and the local pressure increases due to elevation head. The service water is subcooled at all other locations during normal operation. The LOOP causes the LPSW pumps to begin a rapid coastdown. The RBCU and RBACU fans would also begin a gradual coastdown, but since there is no significance to the fan response during a LOOP, the fans are assumed to remain at full speed in the RELAP5 model. The decrease in pressure due to LPSW pump coastdown causes the elevated piping runs to go subatmospheric and draw a vacuum at the highest elevation equal to the saturation pressure of the service water. The service water falls to the lower elevations except where trapped in a loop seal. All of the RBCUs and RBACUs are in loop seals and remain subcooled.

Heat transfer from the containment is limited by the normal containment temperature, approximately 110°F, and so heat transfer is unimportant to the thermal-hydraulic response following a LOOP. No steam is generated except that which flashes during the draining phase, and the steam is approximately in thermal equilibrium with the liquid. Sensitivity cases were run considering bounding power restoration times of 11 and 33 seconds. The LPSW pump restarts when power is restored, and flow rapidly increases as predicted by the RELAP5 centrifugal pump dynamic model and the hydraulic losses predicted by the model. Due to the absence of any heat transfer and the approximate thermal equilibrium throughout the LPSW System, no conditions exist with the potential for condensation-induced waterhammer.

The acceleration of the flow column following pump restart does result in collapsing the voids and column rejoining waterhammer. The maximum waterhammer overpressure results from the maximum water velocity during the refill period. RELAP5 predicts a maximum velocity of 10.6 ft/sec in the 8 inch pipe, 9.2 ft/sec in the 6 inch pipe, and 10.4 ft/sec in the 4 inch pipe. Assuming the maximum service water density at the coldest LPSW temperature of 40°F, the waterhammer overpressures are conservatively calculated to be 642 psi in the 8 inch pipe, 558 psi in the 6 inch pipe, and 630 psi in the 4 inch pipe.

LOCA/LOOP Results

With the initial conditions prior to the LOCA/LOOP, two-phase conditions exist in the high point downstream of the discharge of the RBCUs. The flow returns to single phase as it flows vertically downward and the local pressure increases due to elevation head. The service water is subcooled at all other locations during normal operation. The LOCA/LOOP causes the LPSW pumps to begin a rapid coastdown, and the RBCU and RBACU fans also begin a gradual coastdown. The rapid increase in containment temperature and pressure results in an increase in heat transfer to the service water flowing through the RBCU and RBACU tubes.

The LPSW pressure initially decreases due to the rapid decrease in pump head, and then subsequently increases due to the heat transfer. Voids appear at all RBCUs and RBACUs except RBCU "B", which is isolated at the initial condition. The generation of voids displaces water into the discharge of the RBCUs and RBACUs. Sensitivity cases were run considering bounding power restoration times of 11 and 33 seconds. The LPSW pump and RBCU fans restart when power is restored. Flow rapidly increases as predicted by the RELAP5 centrifugal pump dynamic model and the hydraulic losses and backpressure predicted by the model. The "B" RBCU fan starts, and service water flow is realigned from the RBACUs to the "B" RBCU. The RBCU fan speed goes to the low setpoint. Restart of the LPSW pump refills the system except for the higher elevation RBCU discharge piping. The RBACU piping is not refilled because LPSW flow is isolated by valve closure.

During the draining and refill phases of the LOCA/LOOP analysis the voids predicted by RELAP5 are individually evaluated for the potential for condensation-induced waterhammer according to the methodology described above. For the voids in the RBCU piping, conditions for condensation-induced water hammer do not exist. For the voids in the RBACUs, it is concluded that conditions may exist during the draining phase with the potential for the occurrence of condensation-induced waterhammer. Further analysis and evaluation may revise this conclusion.

Based on these results, Oconee has elected to isolate the RBACUs and sufficiently drain portions of the LPSW System servicing these units. These actions eliminate the possibility of condensation-induced waterhammer conditions following a LOCA/LOOP scenario. The RBACU fans will be left in operation to assist in circulating air in the Reactor Building. These compensatory actions will remain in place until further analyses are completed.

The acceleration of the flow column due to steam generation in the RBCUs and RBACUs and the LPSW pump restart does result in collapsing the voids and column rejoining waterhammer. The results of the RELAP5 analysis are evaluated to identify the maximum fluid velocities in water-filled piping during the analysis. The maximum waterhammer overpressure results from the maximum water velocity, which occurs during the refill period. Since the RBACUs do not completely refill due to isolation of service water flow (assuming that the RBACUs are initially in operation), only the RBCUs and related piping are subject to column rejoining overpressure. RELAP5 predicts a maximum velocity of 12 ft/sec in the 8 inch pipe, 10 ft/sec in the 6 inch pipe, and 11.3 ft/sec in the 4 inch pipe. Assuming the maximum service water density at the coldest LPSW temperature of 40°F (no credit for heat transfer), the waterhammer overpressures are conservatively calculated to be 727 psi in the 8 inch pipe, 606 psi in the 6 inch pipe, and 685 psi in the 4 inch pipe.

VI. Basis for Operability

As described above, Duke has thoroughly analyzed the potential for condensation-induced and column rejoining waterhammer phenomena to occur in the Oconee LPSW System. The results of this analysis conclude that column rejoining waterhammers will occur following a LOOP or LOCA/LOOP event. Conditions conducive to a condensation-induced waterhammer are not predicted for any LOOP event. The analysis concludes that, following a LOCA/LOOP event, condensation-induced waterhammer will not occur in the LPSW piping that serves the safety-related RBCUs. However, the analysis concludes that it is possible condensation-induced waterhammer may occur in the LPSW piping that serves the non-safety-related RBACUs. As a conservative measure until further analysis can be performed, ONS is isolating service water flow to the RBACUs. Therefore, with these compensatory actions, conditions conducive to condensation-induced water hammer will be prevented for any postulated LOOP or LOCA/LOOP event.

ONS has performed substantive testing which demonstrates the loads induced by a column-rejoining waterhammer are well within the structural capacity of the LPSW System. An Emergency Power Engineered Safeguards Functional Test was performed from January 2, 1997, to January 5, 1997. The test was divided into six parts as described in a Duke submittal to the NRC dated December 11, 1996. The test, which involved all three Oconee units, resulted in sixteen cases where power was interrupted to the LPSW System and RBCUs. The RBCU discharge piping located in the Unit 3 East Penetration Room was observed during Test 1. A LOOP was simulated on Unit 3 during this test. During Test 2, where a LOOP was simulated on Unit 2, the RBCU discharge piping located in the Unit 2 East Penetration Room was observed. During Test 5, which simulated a simultaneous LOCA/LOOP on Unit 3, the RBCU discharge piping in the Unit 3 East Penetration Room was observed. In all cases, no pipe movement was observed and no unusual noises were heard. The LPSW System responded as expected during these tests with no anomalies.

The sections of LPSW piping on Unit 2 where column-rejoining waterhammers are predicted to occur during a LOOP were inspected by civil engineers on January 18, 1997. The inspected piping included the LPSW discharge piping from RBCUs 2A, 2B, and 2C to the embedment in the Turbine Building and the LPSW discharge piping from RBACUs 2A, 2B, 2C, and 2D to the tie-in point at the RBCU discharge piping. The same scope of piping was inspected on Units 1 and 3 on January 22, 1997.

These inspections looked for evidence of significant waterhammer loads, such as:

- Broken or deformed pipe supports.

- Signs of pipe movement such as damaged insulation, scrapes on piping and supports, or a damaged fire seal at floor or wall sleeves.

- Deformed welded attachments (those that could be viewed without removing insulation).

- Deformation at coils, coil nozzles, and coil support structures.

No adverse conditions were found during the inspection. All components and piping supports were in good material condition. No signs of overloading from waterhammer existed. No insulation damage from piping movement was found, nor any condition indicating pipe movement from waterhammer forces.

In addition to this testing, Unit 2 experienced a LOOP event on October 19, 1992, and Unit 3 experienced a LOOP event on

March 16, 1996. The corrective action database for ONS was reviewed to determine if any evidence of waterhammer was identified during these or other LPSW pump restart events. This review did not identify any problems caused by a waterhammer induced by LPSW pump restart.

This operating experience demonstrates that the column-rejoining waterhammers occurring during LOOP events are minor in nature and are well within the structural capacity of the piping and its supports. Column-rejoining waterhammer loads during a LOCA/LOOP event are predicted to be comparable to the loads during a LOOP event. Based on this conclusion, a sound engineering judgment can be made that the operating experience at ONS demonstrates there are no operability concerns associated with column-rejoining waterhammers occurring in the LPSW System during a LOOP or LOCA/LOOP event. Based on the field conditions observed during the Civil Engineering walkdown, it is concluded that there is sufficient capacity in the LPSW piping and supports to withstand a column-rejoining waterhammer in conjunction with seismic loads and still remain within operable allowables.

Corrective Actions

Duke's thermal-hydraulic analyses predict that conditions conducive to condensation-induced waterhammer may exist in certain portions of the LPSW System that serve the RBACUs following a postulated LOCA/LOOP event. As a conservative measure, corrective actions will be taken prior to the startup of each Oconee unit to eliminate the possibility of condensation-induced waterhammers. LPSW flow to the four RBACUs (total of 16 coils) will be isolated. Trapped sections of fluid will be sufficiently drained to eliminate the possibility of thermal overpressurization.

Isolation of the RBACUs will result in Reactor Building temperatures that are higher than normal. The RBACU fans will remain in operation to circulate air in the Reactor Building. Reactor Building temperatures will be monitored to ensure that design limits are not exceeded. Two of the three RBCUs will operate in high speed to remove heat from containment during normal operation. Duke has previously operated the Oconee units with the RBACUs isolated. This historical data, along with monitoring of the Reactor Building temperatures, will be used to extrapolate trends in temperature and ensure that the design limits are not exceeded.

Duke will be performing additional thermal-hydraulic analyses to further quantify the loads associated with waterhammers in the LPSW System. Upon completion of these analyses, the resulting dynamic loads will be evaluated. The results of the thermal-hydraulic analyses, as well as a schedule for completion of the piping structural analyses and any required modifications that may result from this effort, will be submitted to the staff by August 1, 1997.

VII. Conclusions

The potential for condensation-induced and column-rejoining waterhammer phenomena to occur in the Oconee LPSW System following LOOP and LOCA/LOOP events has been analyzed and evaluated. A detailed RELAP5 model of the LPSW System was used to simulate the LOOP and LOCA/LOOP transients including all relevant initial and boundary conditions. The results indicated significant voiding due to loss of power to the LPSW pumps and the elevation of the system relative to the lake level, and also due to the heat transfer from the containment atmosphere to the RBCUs and the RBACUs following a LOCA.

For the LOOP transient, all voids were evaluated for the potential for condensation-induced waterhammer. It was concluded that no such conditions occur. This conclusion is mainly resulting from the absence of significant heat transfer in the RBCUs and RBACUs due to the nominal containment temperature. The refilling of the piping following restart of the LPSW pumps does result in column-rejoining waterhammer. The overpressure resulting from such waterhammers was conservatively quantified assuming no credit for any air or steam cushioning effects.

For the LOCA/LOOP transient, all voids were evaluated for the potential for condensation-induced waterhammer. For the voids in the RBCU piping, conditions for condensation-induced water hammer do not exist. For the voids in the RBACUs, it is concluded that conditions may exist during the draining phase with the potential for the occurrence of condensation-induced waterhammer. Further analysis and evaluation may revise this conclusion. In order to ensure that such potential waterhammer conditions are prevented following a LOCA/LOOP scenario, the RBACUs will be isolated from the service water flow and not restored following accidents. The RBACUs will also be drained to prevent overpressure. These actions will be completed prior to the restart of each Oconee unit. The RBACU fans will be left in operation to assist in circulating air in the Reactor Building.

The refilling of the piping following restart of the LPSW pumps does result in column-rejoining waterhammer. The overpressure resulting from such waterhammers was conservatively quantified assuming no credit for any air or steam cushioning effects.

Previous analyses performed with a similar RELAP5 model of the LPSW System, and the RELAP5 analyses described above, have shown that the existence of two-phase flow conditions in the LPSW System following a LOCA or SLB, with or without a LOOP, do not significantly degrade the post-LOCA heat rejection function of the RBCUs. The two-phase conditions result in additional hydraulic losses and some reduction in LPSW flow through the RBCUs for a period of time. This flow reduction and resulting heat transfer has been evaluated relative to the flow assumed and heat transferred in the LOCA and SLB containment response analyses. Based on sensitivity studies, heat transfer from the RBCUs is not needed during the first 30 minutes following a LOCA, and the RBCUs are not needed at all following a SLB. The RBCUs are credited in the UFSAR analyses after a conservative delay time. The results of these sensitivity studies indicates the low relative importance of the RBCUs on the containment response to LOCA in the short-term, and on the SLB. It is concluded that two-phase flow in the LPSW piping has an insignificant impact on the containment heat load rejection capability.

The steam line break transient can result in containment temperatures that exceed the LOCA transient. The heat transferred by the RBCUs to the service water following a steam line break can exceed the heat transfer resulting from a LOCA. The heat transfer rates assumed in the RELAP5 analyses are conservative for all LOCAs and steam line breaks. Therefore, the conclusions reached for the LOCA/LOOP scenario are valid for the MSLB/LOOP scenario.

The analyses assumed conservatively high heat transfer in the RBCUs (zero fouling) and conservatively high containment temperature and pressure transients corresponding to the maximum LOCA and MSLB mass and energy releases. These assumptions are considered to be bounding in terms of maximizing the volume of voids generated in the LPSW System, and the resulting potential consequences and magnitude of any two-phase flow conditions and hydrodynamic loadings.

The analyses and evaluations considered the effects of different initial and boundary conditions including single failures. These included the timing of restoration of offsite power, the number of LPSW pumps restarting, the lake level, and the lake temperature. Single failures were also considered. The analysis was performed specifically for Oconee Unit 3. A review of the system design differences relative to Units 1 and 2 has concluded that the results of the Unit 3 analysis is applicable to all units.

For LOCA and MSLB with continued offsite power, the LPSW pumps continue running and no interruption of flow to the RBCUs occurs. For these scenarios no boiling will occur in the RBCUs. Two-phase conditions are predicted at the discharge of the RBCUs at the higher elevations.

The column-rejoining waterhammer overpressures are comparable to those reported by others in the industry, including those that have experienced and measured the magnitude of waterhammers in service water systems following a LOOP. The Oconee units have a significant experience base of tests and a few events which included the type of column-rejoining waterhammer events that are predicted by the RELAP5 model and waterhammer calculations used in this methodology. There has been no indication of any damage in any of the Oconee LPSW Systems during any of these LOOP-type tests or events. Since the conservatively calculated loads for the LOOP and LOCA/LOOP events are comparable in magnitude, it can be concluded that no damage will result from LOOP and LOCA/LOOP loadings on the LPSW piping.

Provided that the RBACUs are isolated from the LPSW flow and drained during normal operation and following accidents, the potential for condensation-induced waterhammer will be precluded until further analysis and evaluation can be completed to determine a long-term solution. With the implementation of this action, the LPSW System is operable with respect to the potential consequences of two-phase conditions following LOOP, MSLB/LOOP, and LOCA/LOOP events.

VIII. References

1. RELAP5/MOD3 Code Manual, NUREG/CR-5535, Lockheed-Martin Idaho Technologies Company

2. GOTHIC Containment Analysis Package User Manual, Version 5.0, NAI 8907-02, Rev. 6, Numerical Applications Incorporated, December 1995

3. "Containment Air Cooler Heat Transfer During Loss of Coolant Accident With Loss of Offsite Power Conditions," EPRI interim report, Numerical Applications, Inc. November 1996

4. NUREG/CR-5220, Diagnosis of Condensation-Induced Waterhammer, October 1988

5. "Water Hammer Prevention, Mitigation, and Accommodation, EPRI NP-6766, July 1992

6. Letter, B. Link, Wisconsin Electric Power Co, to USNRC PDR, September 9, 1996

Table 1
Oconee Unit 3 LPSW RELAP5 Model

RELAP 5 Component Number	System Component	Number of Nodes in RELAP5 Volume	Pipe Diameter	Description
300	pipe	1	41.3"	LPSW pump 3B suction
301	pump	1		LPSW pump 3B
302	pipe	1	23.3"	LPSW pump 3B discharge
303	pipe	1	19.3"	LPSW piping
304	pipe	1	8.0"	component cooling flow path
305	pipe	1	19.3"	LPSW piping
306	pipe	1	15.3"	LPI cooler 3B flow path
307	pipe	1	13.3"	LPSW piping
309	pipe	1	8"	LPSW piping
311	pipe	1	8"	LPSW piping
313	pipe	1	13.3"	LPSW piping
314	pipe	3	15.3"	LPI cooler 3A flow path
315	pipe	1	13.3"	RBCU discharge piping
316	pipe	16	17.25" (all nodes)	RBCU discharge header
317	pipe	1	23.3"	LPSW discharge pipe
318	pipe	1	23.3"	LPSW discharge pipe
319	pipe	1	23.3"	LPSW discharge pipe
320	pipe	3	23.3" (all nodes)	LPSW header A

RELAP 5 Component Number	System Component	Number of Nodes in RELAP5 Volume	Pipe Diameter	Description
321	pipe	1	23.3"	LPSW pump 3A discharge
322	pump	1	23.3"	LPSW pump 3A
323	pipe	1	41.3"	LPSW pump 3A suction
325 (nominal flow path modeled inside containment)	pipe	17	10"	RCP motor coolers piping
331	pipe	18	8" (all nodes)	RBCU 3C inlet
333	RBCU and pipe	33	6" (Nodes 1 to 3, 13 to 17) 4" (Nodes 4, 5, & 12) 0.5268" (Nodes 7 to 10) 8" (Nodes 18 to 33)	RBCU 3C
334	valve	1	8"	Valve 3LPSW-24
335	pipe	4	8" (all nodes)	RBCU 3C outlet
341	pipe	17	8" (all nodes)	RBCU 3B inlet
342	valve	1	8"	Valve 3LPSW-566
343	RBCU and pipe	19	8" (Nodes 1, 2, & 19) 6" (Nodes 3 to 5, 15, to 18) 4" (Nodes 6, 7, & 14) 0.5268" (Nodes 9 to 12)	RBCU 3B
345	pipe	14	8" (all nodes)	RBCU 3B outlet
346	valve	1	8"	Valve 3LPSW-21
347	pipe	4	8" (all nodes)	RBCU 3B outlet
351	pipe	16	8" (all nodes)	RBCU 3A inlet

RELAP 5 Component Number	System Component	Number of Nodes in RELAP5 Volume	Pipe Diameter	Description
353	RBCU and pipe	27	6" (Nodes 1, 2, 12 to 15) 4" (Nodes 3, 4, & 11) 8" (Nodes 16 to 27)	RBCU 3A
354	valve	1	8"	Valve 3LPSW-18
355	pipe	5	8" (all nodes)	RBCU 3A outlet
361	valve	1	8"	Valve 3LPSW-565
362	pipe	1	8"	Aux. Cooler inlet header
363	aux. cooler and pipe	29	8" (Nodes 1 to 3, 28 to 29) 4" (Nodes 4 to 10, 22 to 27) 3" (Nodes 11& 21) 2.5" (Nodes 12, 13, 19, & 20) 0.5268" (Nodes 15 to 17)	Aux. Cooler 3A
364	aux. cooler and pipe	45	4" (Nodes 1 to 16, 28 to 45) 3" (Nodes 17 & 27) 4" (Nodes 18, 19, 25, & 26) 0.5268" (Nodes 21 to 23)	Aux. Cooler 3B
365	aux. cooler and pipe	54	8" (Nodes 1 to 12) 4" (Nodes 13 to 21, 33" to 54) 3" (Nodes 22 & 32) 2.5" (Nodes 23, 24, 30, & 31)	Aux. Cooler 3C

RELAP 5 Component Number	System Component	Number of Nodes in RELAP5 Volume	Pipe Diameter	Description
366	aux. cooler and pipe	19	4" (Nodes 1 to 4, 16 to 19) 3" (Nodes 5 & 15) 2.5" (Nodes 6, 7, 13, & 14) 0.5268" (Nodes 9 to 11)	Aux. Cooler 3D
367	pipe	3	8" (all nodes)	Aux. Cooler outlet header
386	valve	1		RCP motor cooler inlet
387	valve	1		LPI 3B cooler inlet
388	valve	1		LPI 3A cooler inlet
389	valve	1		LPI 3A cooler discharge
391	pipe	1	13.3"	LPSW piping
392	pipe	1	8"	LPSW piping
393	pipe	1	13.3"	LPSW piping
394	pipe	1	15.3"	LPSW piping
700		1	Time Dependent Volume	CCW Suction Pipe - Lake Keowee
701		1	Time Dependent Volume	Main turbine oil tank cooler
703		1	Time Dependent Volume	CCW discharge
704		1	Time Dependent Volume	CCW suction pipe - Lake Keowee
710		1	Time Dependent Volume	Containment Atmosphere
711		1		Containment Atmosphere
712		1	Time Dependent Junction	RBCU 3A air flow
713		1	Time Dependent Junction	RBCU 3B air flow
714		1	Time Dependent Junction	RBCU 3C air flow

RELAP 5 Component Number	System Component	Number of Nodes in RELAP5 Volume	Pipe Diameter	Description
715		1	Time Dependent Junction	Aux. Cooler 3A air flow
716		1	Time Dependent Junction	Aux. Cooler 3B air flow
717		1	Time Dependent Junction	Aux. Cooler 3C air flow
718		1	Time Dependent Junction	Aux. Cooler 3D air flow
721		4		RBCU 3A cooler atmosphere
731		4		RBCU 3B cooler atmosphere
741		4		RBCU 3C cooler atmosphere
751		4		Aux. Cooler 3A atmosphere
761		4		Aux. Cooler 3B atmosphere
771		4		Aux. Cooler 3C atmosphere
781		4		Aux. Cooler 3D atmosphere
790		1		Containment atmosphere
791		1	Time Dependent Volume	Containment atmosphere

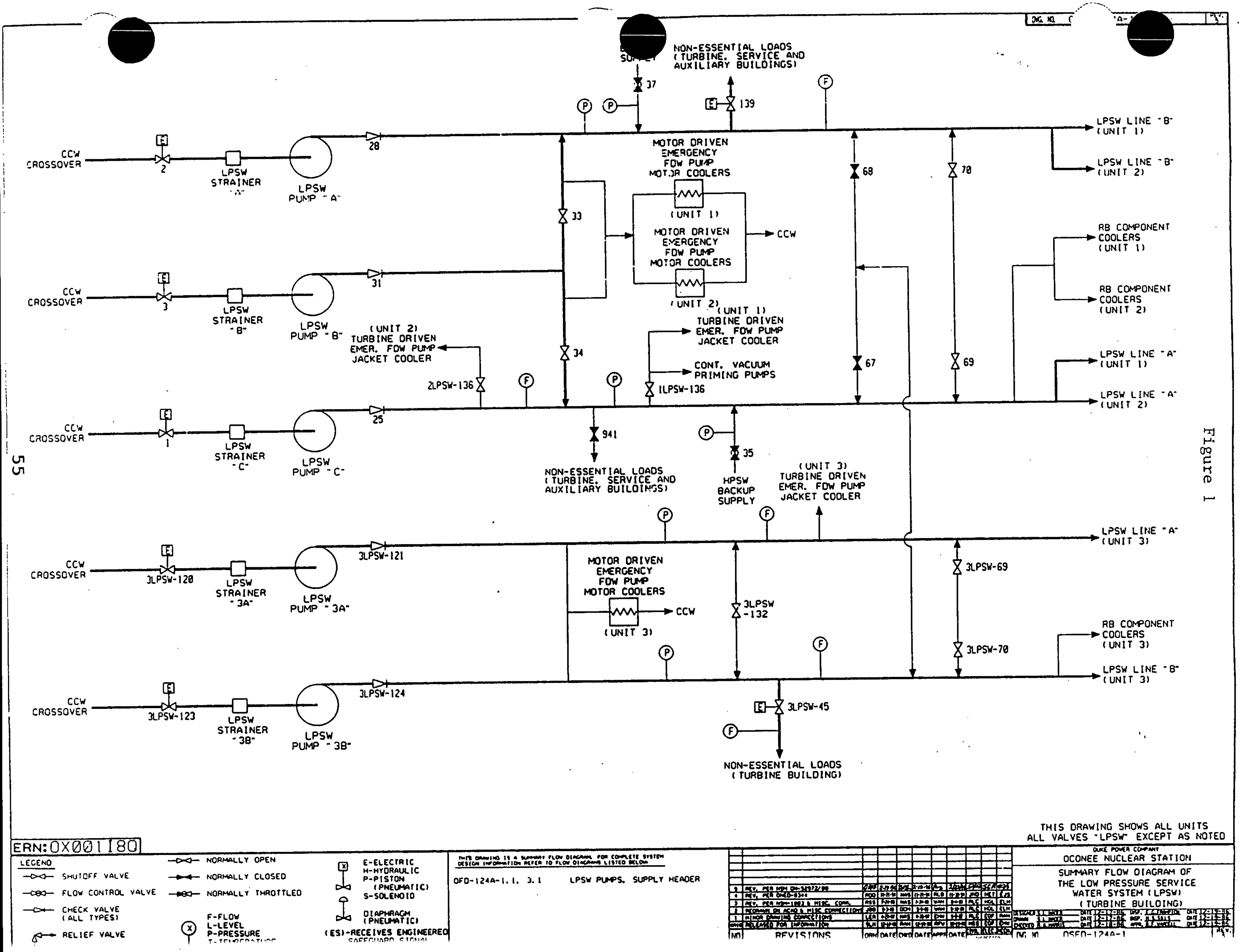
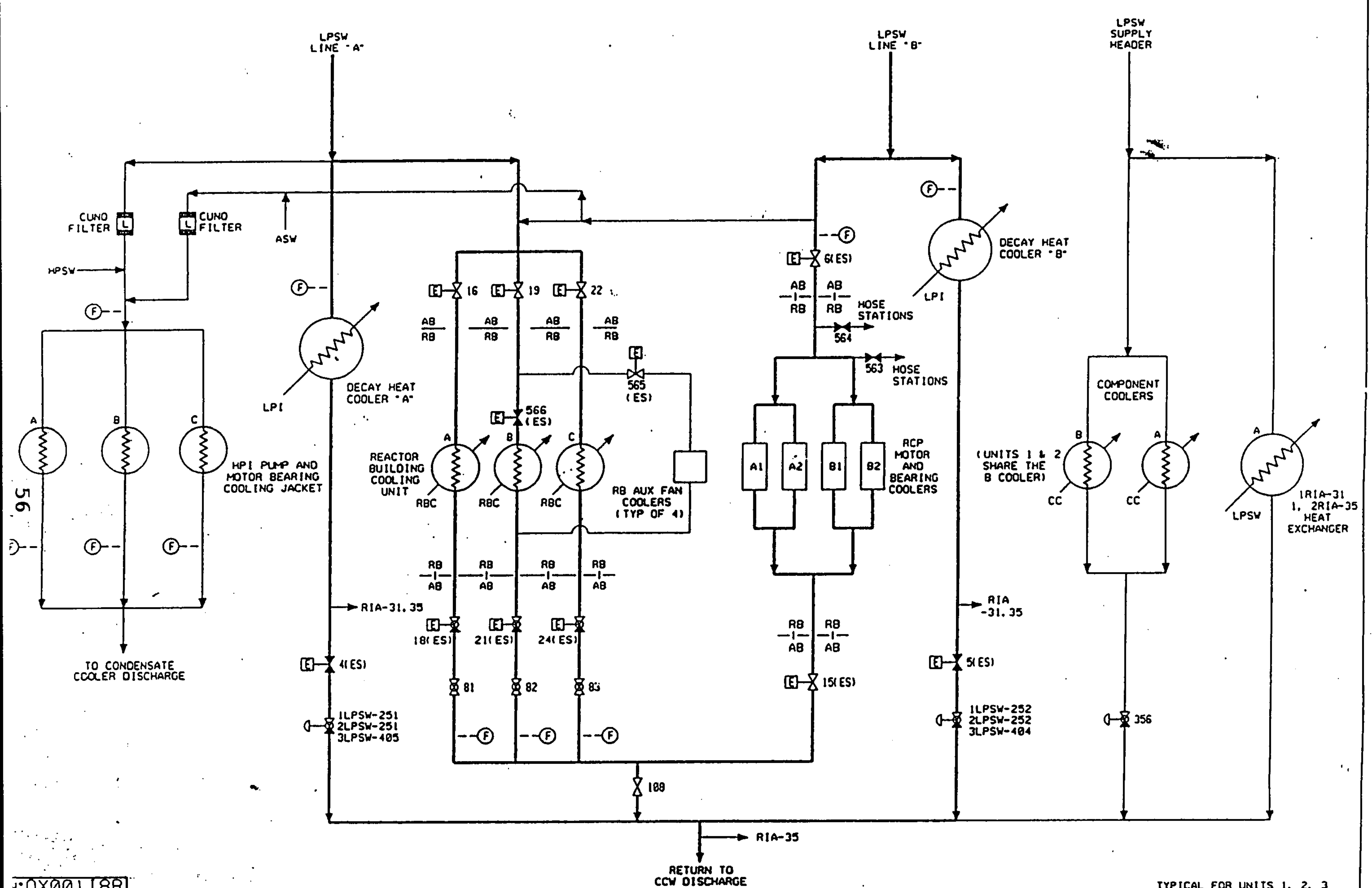

NON-ESSENTIAL LOADS (TURBINE, SERVICE AND AUXILIARY BUILDINGS)
37
139
LPSW LINE "B" (UNIT 1)
LPSW LINE "B" (UNIT 2)
CCW CROSSOVER
2
LPSW STRAINER "A"
LPSW PUMP "A"
28
MOTOR DRIVEN EMERGENCY FDW PUMP MOTOR COOLERS
(UNIT 1)
68
70
33
MOTOR DRIVEN EMERGENCY FDW PUMP MOTOR COOLERS
CCW
RB COMPONENT COOLERS (UNIT 1)
RB COMPONENT COOLERS (UNIT 2)
CCW CROSSOVER
3
LPSW STRAINER "B"
LPSW PUMP "B"
31
(UNIT 2)
(UNIT 1) TURBINE DRIVEN EMER. FDW PUMP JACKET COOLER
(UNIT 2) TURBINE DRIVEN EMER. FDW PUMP JACKET COOLER
34
CONT. VACUUM PRIMING PUMPS
67
69
LPSW LINE "A" (UNIT 1)
LPSW LINE "A" (UNIT 2)
2LPSW-136
1LPSW-136
CCW CROSSOVER
1
LPSW STRAINER "C"
LPSW PUMP "C"
25
941
35
NON-ESSENTIAL LOADS (TURBINE, SERVICE AND AUXILIARY BUILDINGS)
HPSW BACKUP SUPPLY
(UNIT 3) TURBINE DRIVEN EMER. FDW PUMP JACKET COOLER
LPSW LINE "A" (UNIT 3)
CCW CROSSOVER
3LPSW-120
LPSW STRAINER "3A"
LPSW PUMP "3A"
3LPSW-121
MOTOR DRIVEN EMERGENCY FDW PUMP MOTOR COOLERS
CCW
(UNIT 3)
3LPSW -132
3LPSW-69
3LPSW-70
RB COMPONENT COOLERS (UNIT 3)
LPSW LINE "B" (UNIT 3)
CCW CROSSOVER
3LPSW-123
LPSW STRAINER "3B"
LPSW PUMP "3B"
3LPSW-124
3LPSW-45
NON-ESSENTIAL LOADS (TURBINE BUILDING)
THIS DRAWING SHOWS ALL UNITS
ALL VALVES "LPSW" EXCEPT AS NOTED
ERN: 0X001180
LEGEND
SHUTOFF VALVE
FLOW CONTROL VALVE
CHECK VALVE (ALL TYPES)
RELIEF VALVE
NORMALLY OPEN
NORMALLY CLOSED
NORMALLY THROTTLED
F-FLOW
L-LEVEL
P-PRESSURE
E-ELECTRIC
H-HYDRAULIC
P-PISTON (PNEUMATIC)
S-SOLENOID
DIAPHRAGM (PNEUMATIC)
(ES)-RECEIVES ENGINEERED SAFEGUARD SIGNAL
THIS DRAWING IS A SUMMARY FLOW DIAGRAM. FOR COMPLETE SYSTEM DESIGN INFORMATION REFER TO FLOW DIAGRAMS LISTED BELOW
OFD-124A-1.1, 3.1 LPSW PUMPS, SUPPLY HEADER
REVISIONS
DUKE POWER COMPANY
OCONEE NUCLEAR STATION
SUMMARY FLOW DIAGRAM OF THE LOW PRESSURE SERVICE WATER SYSTEM (LPSW) (TURBINE BUILDING)
OSFD-124A-1

Figure 1



TYPICAL FOR UNITS 1, 2, 3

N:0X00118R

LEGEND

SHUTOFF VALVE	NORMALLY OPEN	E-ELECTRIC	
FLOW CONTROL VALVE	NORMALLY CLOSED	H-HYDRAULIC	
CHECK VALVE (ALL TYPES)	NORMALLY THROTTLED	P-PISTON (PNEUMATIC)	
RELIEF VALVE	F-FLOW, L-LEVEL, P-PRESSURE, T-TEMPERATURE	S-SOLENOID	
		DIAPHRAGM (PNEUMATIC)	
		(ES)-RECEIVES ENGINEERED SAFEGUARD SIGNAL	

THIS DRAWING IS A SUMMARY FLOW DIAGRAM. FOR COMPLETE SYSTEM DESIGN INFORMATION REFER TO FLOW DIAGRAMS LISTED BELOW

OFD-124B-1.1, 2.1, 3.1	DECAY HEAT COOLERS, HPI PUMP COOLERS COMPONENT COOLERS
OFD-124B-1.2, 2.2, 3.2	REACTOR BUILDING COOLING UNITS
OFD-124B-1.3, 2.3, 3.3	REACTOR BUILDING AUX. FAN COOLERS
OFD-124B-1.4, 2.4, 3.4	RC PUMP MOTOR AND BEARING COOLERS
OFD-124B-1.5, 3.5	RIA HEAT EXCHANGERS
OFD-124B-1.6	LPSW - AUX. BLDG. AHUs

NO.	REVISIONS	DRN	DATE	CHKD	DATE	APPD	DATE
6	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
5	REV. PER OKED-0467	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
4	REV. PER OKED-0344	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
3	MISC. CORRECTIONS	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
1	MINOR DRAWING CORRECTION	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
0	RELEASED FOR INFORMATION	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

SUMMARY FLOW DIAGRAM OF
LOW PRESSURE SERVICE
WATER SYSTEM (LPSW)
(REACTOR BLDG. AUX. BLDG.)

DG NO. OSFD-124B-1

Figure 2

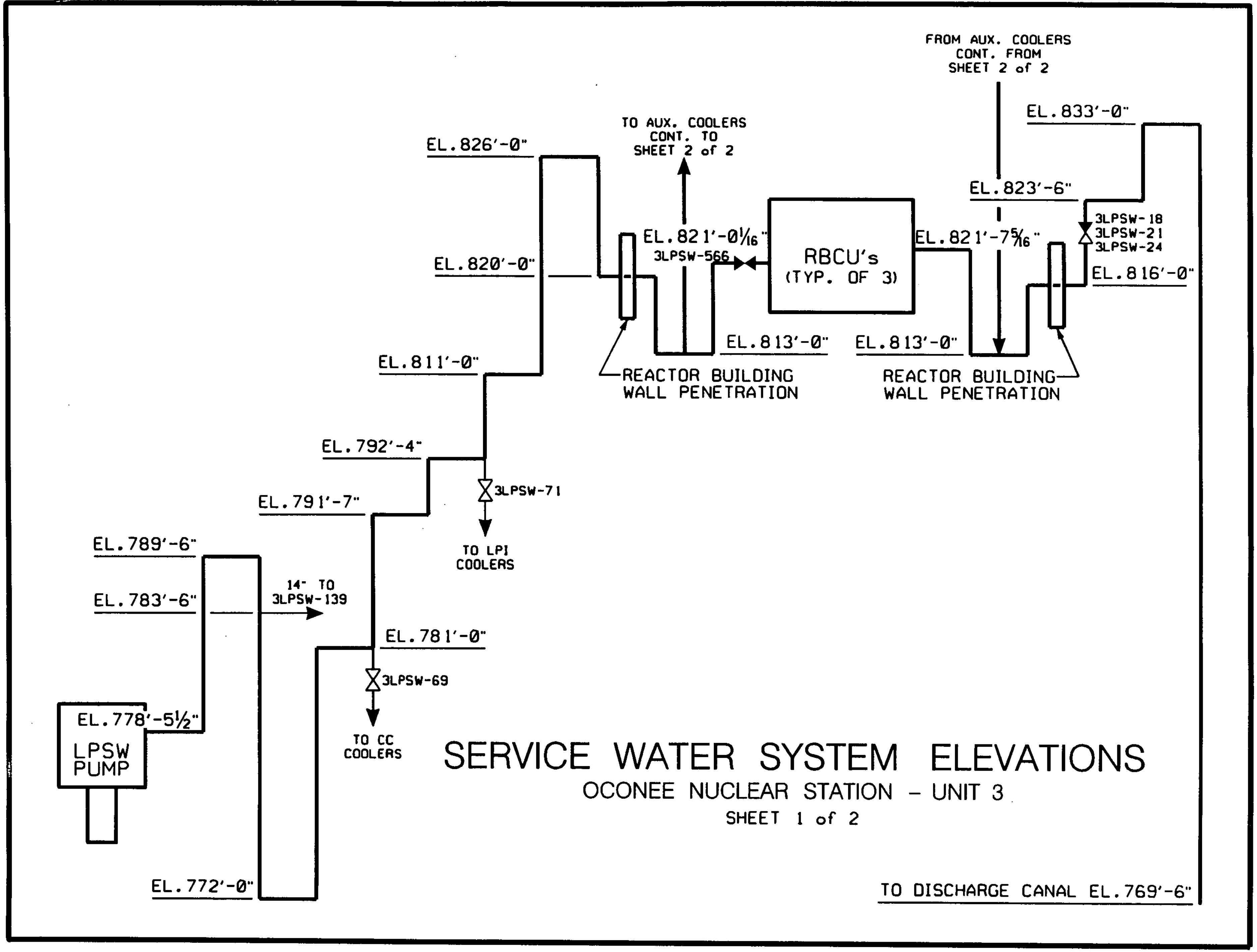
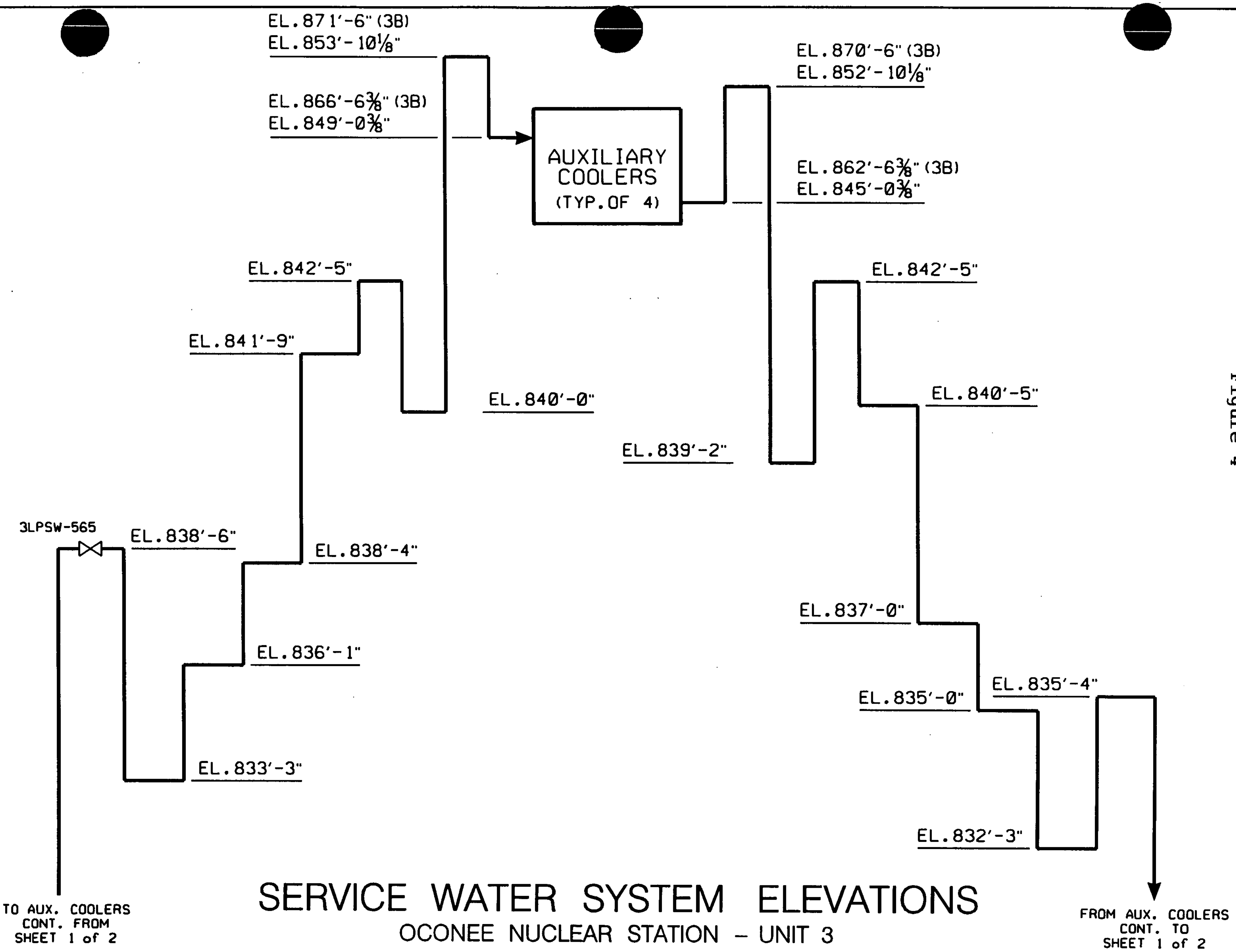

FROM AUX. COOLERS CONT. FROM SHEET 2 of 2
EL.833'-0"
TO AUX. COOLERS CONT. TO SHEET 2 of 2
EL.826'-0"
EL.823'-6"
3LPSW-18
3LPSW-21
3LPSW-24
EL.821'-0 1/16"
3LPSW-566
RBCU's (TYP. OF 3)
EL.821'-7 5/16"
EL.820'-0"
EL.816'-0"
EL.813'-0"
EL.813'-0"
EL.811'-0"
REACTOR BUILDING WALL PENETRATION
REACTOR BUILDING WALL PENETRATION
EL.792'-4"
3LPSW-71
EL.791'-7"
TO LPI COOLERS
EL.789'-6"
14" TO 3LPSW-139
EL.783'-6"
EL.781'-0"
3LPSW-69
EL.778'-5 1/2"
LPSW PUMP
TO CC COOLERS
SERVICE WATER SYSTEM ELEVATIONS
OCONEE NUCLEAR STATION - UNIT 3
SHEET 1 of 2
EL.772'-0"
TO DISCHARGE CANAL EL.769'-6"

Figure 3



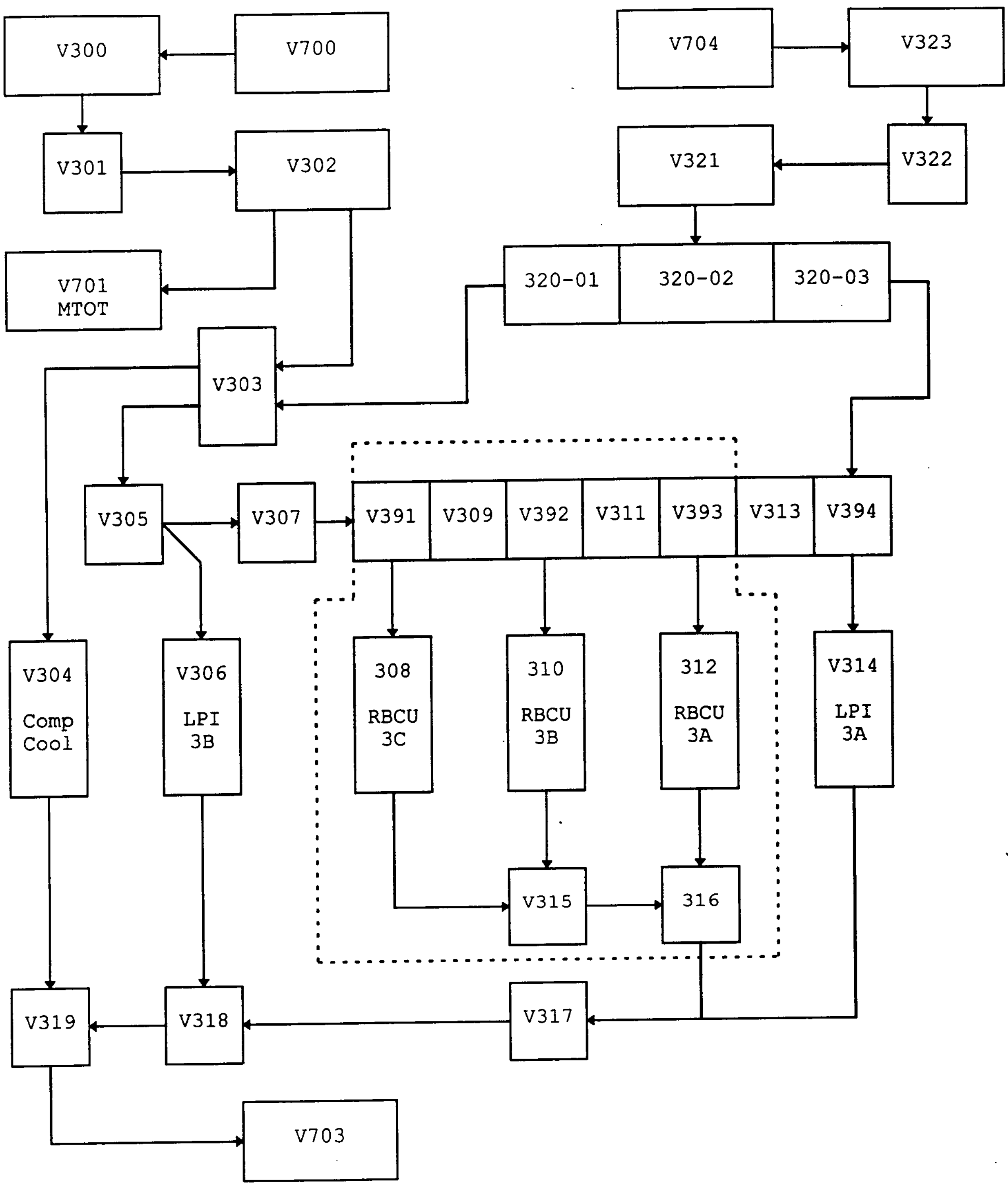
SERVICE WATER SYSTEM ELEVATIONS

OCONEE NUCLEAR STATION – UNIT 3

SHEET 2 of 2

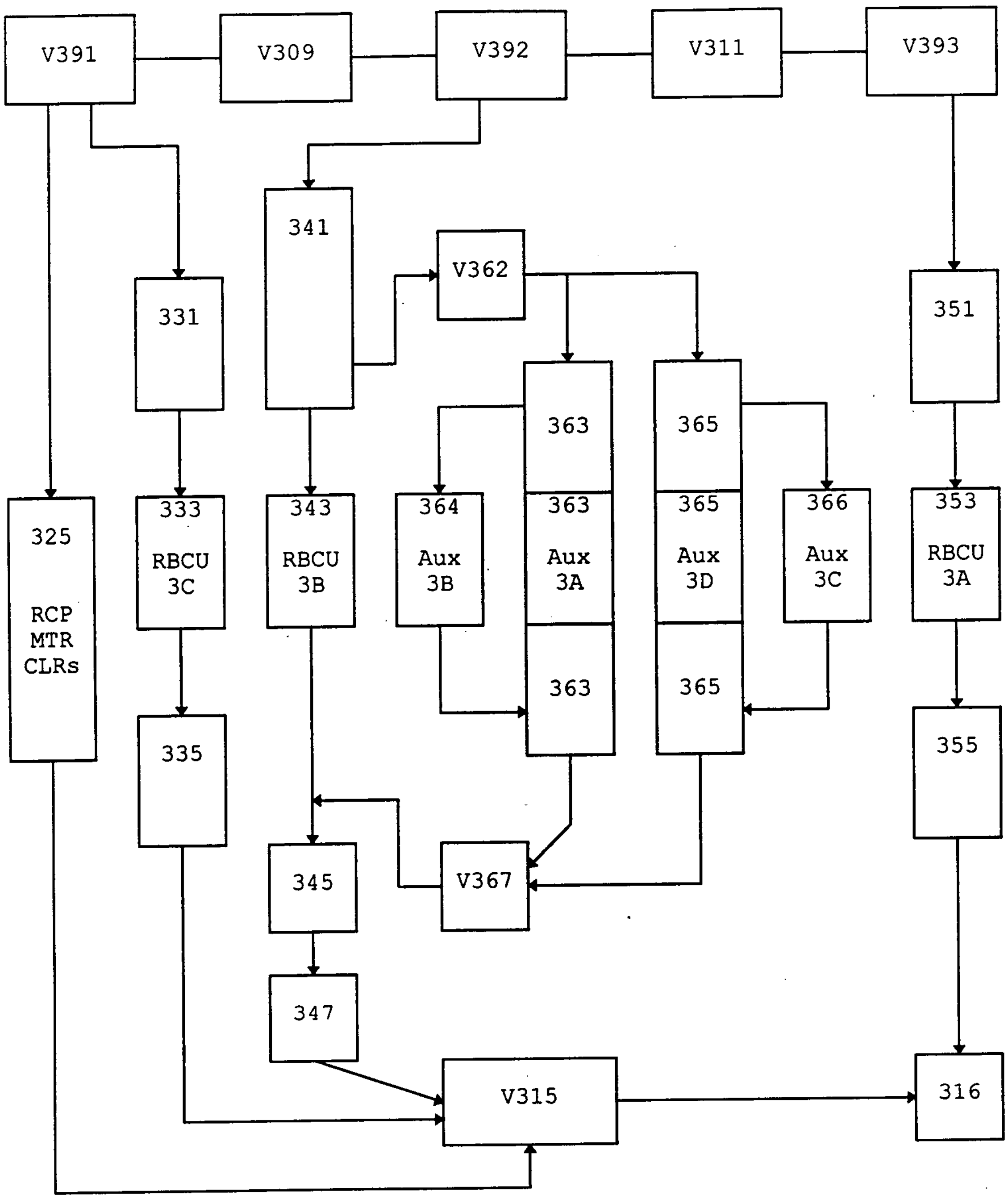
Figure 4

Figure 5



RELAP5 Model of Oconee Unit 3 LPSW System
(See Figure 6 for details inside dashed line)

Figure 6



RELAP5 Model of Oconee Unit 3 LPSW System

(Continued from Figure 5)

Requested Action 2

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

Response:

A water solid volume between two isolation valves, such as a reactor building penetration, may be subject to overpressurization due to thermal expansion. This phenomenon is recognized as a design consideration by USAS B31.1.0-1967 and USAS B31.7-1969. Additional design considerations include leakage limits for isolation valves and the requirement that no credible single failure will result in the loss of containment integrity.

Oconee Units 1, 2 and 3 have 212 mechanical Reactor Building penetrations. Nine of the 212 penetrations are equipped with pressure relief devices. The other 203 penetrations do not need thermal overpressure protection due to pipe expansion, valve leakage, fluid type, operating fluid temperature or service requirements during the accident.

I. Reactor Building Penetrations

A. Penetrations with relief devices installed between the isolation valves.

Penetration #- 21 - LPSW to Reactor Coolant Pump Motors Inlet
22 - LPSW to Reactor Coolant Pump Motors Outlet
57 - DHR Return (Unit 1)
62 - DHR Return (Units 2 and 3)

Total number of penetrations - 9.

B. Penetrations where both containment isolation valves are located outside containment and temperature variations are minimal. Thermal overpressure protection is not required.

Penetration #- 5a - Reactor Building Normal Sump
5b - Post Accident Liquid Sample
36 - Reactor Building Emergency Sump

37 - Reactor Building Emergency Sump
40 - Reactor Building Emergency Sump Drain

Total number of penetrations - 15.

C. Penetrations that utilize an inboard check valve for containment isolation, through which the pressure increase could flow unimpeded to a relief device or sufficient piping volume located inside the Reactor Building. Thermal overpressure protection is not required.

Penetration #- 3 - Component Cooling Inlet
38 - Quench Tank Cooler Inlet
44 - Component Cooling to CRD Inlet

Total number of penetrations - 9.

D. Penetrations that are in service and provide an accident mitigation function during a LOCA. Thermal overpressure protection is not required.

Penetration #- 8a - Pressurizer Auxiliary Spray
8b - Loop "A" Nozzle Warming Line
9 - RCS Normal Make-up and HP Injection "A" Loop
10a - RC Pump "B1" Seal Injection
10b - RC Pump "B2" Seal Injection
13 - Reactor Building Spray
14 - Reactor Building Spray
15 - LPI and DHR Inlet
16 - LPI and DHR Inlet
17 - OTSG "B" Emergency Feedwater
23a - RC Pump "A1" Seal Injection
23b - RC Pump "A2" Seal Injection
25 - OTSG "B" Feedwater Line
26 - OTSG "A" Main Steam
27 - OTSG "A" Feedwater Line
28 - OTSG "B" Main Steam

30 - LPSW Inlet to RBCUs
31 - LPSW Inlet to RBCUs
32 - LPSW Inlet to RBCUs
33 - LPSW Outlet to RBCUs
34 - LPSW Outlet to RBCUs
35 - LPSW Outlet to RBCUs
50 - OTSG "A" Emergency Feedwater
52 - HP Injection "B" Loop

Total number of penetrations - 72.

E. Penetrations which contain air, steam or other compressible gas mixtures that are not exposed to sufficient heat so as to experience overpressurization. Thermal overpressure protection is not required.

Penetration #- 18 - Quench Tank Vent
19 - Reactor Building Purge
20 - Reactor Building Purge
24a - Reactor Building Hydrogen Analyzer Train"A"
24b - Reactor Building Hydrogen Analyzer Train"A"
39a - Core Flood Tank Vent (Units 2 and 3)
39b - High Pressure Nitrogen Supply
41 - Instrument Air
42a - Reactor Building Hydrogen Analyzer Train"B"
42b - Reactor Building Hydrogen Analyzer Train"B"
45a - LLRT/ILRT Instrument Line
45b - LLRT/ILRT Instrument Line
45c - LLRT/ILRT Instrument Line (Units 2 and 3)
48 - Breathing Air
49 - Low Pressure Nitrogen (Unit 1)
51 - ILRT Pressurization Line
53a - High Pressure Nitrogen to CF Tank "A"
53b - Low Pressure Nitrogen (Units 2 and 3)
56 - Spent Fuel Canal Fill and Drain
60 - Reactor Building Sample
61 - Reactor Building Sample

Total number of penetrations - 58.

F. Penetrations that normally operate at fluid temperatures higher than peak containment accident temperature. Thermal overpressure protection is not required.

Penetration #- 1 - Pressurizer Liquid Sample (Unit 1)
2 - OTSG "A" Sample
4 - OTSG "B" Drain (Blow Down)
43 - OTSG "A" Drain (Blow Down)
58a - Pressurizer Liquid Sample (Units 2 and 3)
58b - OTSG "B" Sample.

Total number of penetrations - 15

G. Penetrations that are flanged inside the Reactor Building, but open-ended outside the Reactor Building such as the fuel transfer tubes. Thermal overpressure protection is not required.

Penetration #- 11a - Fuel Transfer Tube
12a - Fuel Transfer Tube

Total number of penetrations - 6.

H. Penetrations which utilize at least one soft seated valve capable of slight leakage or displacement to accommodate thermal expansion following an accident. This feature is attributed to diaphragm, butterfly, and soft seated valves. Thermal overpressure protection is not required.

Penetration #- 29 - Quench Tank Drain
46 - Filtered Water
54 - Component Cooling Outlet
55 - Demineralized Water

Total number of penetrations - 12

I. Penetrations that rely on valve leakage for thermal overpressurization protection.

Penetration # - 6 - Reactor Coolant Letdown
7 - Reactor Coolant Seal Return
11b - Reactor Coolant Makeup Pump Suction
12b - Reactor Coolant Makeup Pump Discharge
47 - Demineralized Water (Unit 1)
59 - Core Flood Tank Sample

Total number of penetrations - 16.

The need for additional overpressure protection for the penetrations listed in H. and I. above has been reviewed several times during the past. None of these penetrations are required to be opened to mitigate the consequences of any design basis accident that generates high containment temperatures. Our experience shows that these valves always have some amount of leakage, thereby eliminating the possibility of overpressurization. This assumption is supported by experience with reliable system operation over a wide temperature range. As an example, penetration 11b is located in the Reactor Building basement which experiences seasonal temperature changes of approximately 30°F. Chemical Engineering, May 1979 article, "Safer Relief Valve Sizing," states that a 30°F temperature increase could theoretically raise the pressure by approximately 1100 psi. Penetration 11b has not experienced any overpressurization nor have binding problems been experienced during any quarterly inservice pump test. This indicates that sufficient leakage is present to preclude overpressurization.

The assumption that valve leakage exists is further supported by our periodic containment isolation valve testing. ONS uses the 10CFR50 Appendix J local leak rate testing program to determine penetration leakage. Local leak rate testing is conducted with air when the system is drained of water. ONS recognizes the correlation between air leakage and water leakage is very difficult to quantify. However, the local leak rate test does establish that a leak path is present which provides reasonable assurance that overpressurization will not occur. The local leak rate test also provides quantifiable, predictable and known leakages.

II. Other Blocked-in Piping

The ONS review of thermal overpressurization was not limited to just containment penetrations. An additional review of all valves located in the Reactor Building that provide double isolation and are required to open to mitigate the consequences of a design basis accident was conducted. This review was used to determine if any safety significant piping sections could be subjected to the effects of thermal overpressurization. Two possible locations were identified:

A. Primary Boron Dilution Flow Path (LP-103 and -104).

Periodic valve stroking procedures leave the volume between the valves drained. Therefore, thermal overpressurization protection is not required. System alignment procedures will be revised to ensure the volume between the valves is always drained should maintenance be performed after the periodic procedures are completed. The potential of refilling the piping between LP-103 and LP-104 is very remote. To refill this piping both valves must leak; LP-103 to admit water into the piping and LP-104 to allow air to be displaced. If both valves leak sufficiently to refill the piping it is not credible to assume thermal overpressurization will occur.

B. Secondary Boron Dilution Flow Path (LP-1 and -2)

Valves LP-1 and LP-2 are the first and second isolation valves on the decay heat drop line. Following a LOCA, these valves can be opened to provide an alternate boron dilution flow path. As a conservative action, ONS has implemented a procedure change to drain a small volume of water from the pipe between these two valves. Thus, if no leakage exists in these isolation valves, this air space introduces a sufficient expansion volume to preclude overpressurization of the pipe. ONS has determined that the small air volume will not adversely impact operation of the Low Pressure Injection (LPI) and Reactor Building Spray (RBS) pumps. This procedure change is being implemented as an interim measure until more detailed analyses can be completed regarding the potential for overpressurization of this pipe.

III. Basis For Operability

ONS has evaluated the containment penetrations and closed sections of piping inside containment that may be subject to thermally-induced overpressurization. The basis for operability of each penetration and closed section of piping has been demonstrated. Depending on the penetration or pipe, the operability analysis credits valve leakage, fluid type, operating fluid temperature, or service requirements during an accident as the basis for concluding thermally-induced overpressurization does not impact operability of these components. This analysis does not credit thermal expansion of the piping system and valve components which would greatly reduce the amount of valve leakage required to prevent overpressurization. This provides an additional indication of the conservative nature of this analysis.

IV. Corrective Actions

In response to the issues identified in GL 96-06, Oconee has retained the services of Structural Integrity to perform additional thermal-hydraulic and structural analyses. This analysis will not credit leakage for thermal overpressurization protection and will quantify the potential for overpressurization of these penetrations. The results of these analyses will be evaluated to determine if any modifications to the penetrations are necessary from a long-term perspective. Any modifications to these penetrations must be carefully evaluated with respect to their impact on possibly increasing Reactor Building leakage and the resulting effects on off-site dose.

In addition, similar analyses will be performed to address thermally-induced overpressurization of the pipe between LP-1 and LP-2. Until these analyses are completed, the previously described corrective action of draining a small volume of water from this pipe will be implemented prior to the restart of each Oconee unit.

The conclusions of these additional thermal-hydraulic and structural analyses, along with a description and schedule for any longer-term corrective actions resulting from this effort, will be submitted to the staff by April 15, 1997.