#2 .9403180250

Duke Power Company
Oconee Nuclear Site
P.O. Box 1439
Seneca, SC 29679

J. W. Hampton
Vice President
(803)885-3499 Office
(803)885-3564 Fax

DUKE POWER

March 14, 1994

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Site
Docket Nos. 50-269, -270, -287
Inspection Report 50-269, -270, -287/93-25

Dear Sir:

By letter dated February 11, 1994 the Service Water Inspection Report for Oconee Nuclear Site was issued by the NRC. Within the report, violations, deviations and other items were identified which required a response by March 13, 1994.

Attached is Duke Power's response to Items 5, 6, and 7, as requested in your cover letter.

The response for the violations and deviations will be submitted on or before May 12, 1994, as was requested in our March 1, 1994 correspondence.

Very truly yours,

J. W. Hampton

cc: Mr. S. D. Ebneter, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II

Mr. L. A. Wiens, Project Manager
Office of Nuclear Reactor Regulation

Mr. P. E. Harmon
Senior Resident Inspector
Oconee Nuclear Site

8-2
A001
1

Item No. 5 - There was only one valve isolating the safety-related portion of the LPSW system from the nonsafety-related turbine building portion.

RESPONSE:

Design Criterion 41, contained in Section 3.1 of the Oconee FSAR, states that an engineered safety feature (such as the LPSW System) shall provide its required safety function assuming a failure of a single active component. This criterion ensures that sufficient redundancy is provided so that the failure of a single component will not impair the required safety function of that system. However, the Oconee FSAR documents the fact that this criteria was not applied to isolation of a postulated, seismically-induced break in the non-seismic portion of a safety system, during a design basis accident.

Section 3.7.3.9 of the Oconee FSAR specifically addresses the issue of seismic/non-seismic isolation valves. It states that "seismic/non-seismic boundaries are established by valves which are designed to meet the seismic design criteria. Failure in the non-seismic portion of the system cannot cause loss of function to the safety system in that automatic or remote manual-operated valves are used for valves that are normally open during Reactor Operation." This section goes on to state that "the seismic/non-seismic boundary valve is protected from seismic effects by restraining the non-seismic portion of the system downstream of the valve." By describing the characteristics of a single boundary valve, this section, in effect, represents an explicitly documented exception to Design Criterion 41.

The exact wording found in Section 3.7.3.9 of the Oconee FSAR can also be found in Appendix C of the July 1972 version of the Oconee FSAR. The wording has been constant over the life of Oconee. The use of a single isolation valve for seismic boundaries has been designed into several systems at Oconee (e.g., Main Steam, Main Feedwater). The LPSW System is not the only occurrence of this design feature.

Additional evidence of the acceptability of a single isolation valve for seismic boundaries can be found in the Safety Evaluation Report (SER) the NRC issued Oconee for Generic Letter (GL) 81-14, "Seismic Qualification of Auxiliary Feedwater Systems." This evaluation also considered portions of the LPSW System since LPSW is a support system for the EFW pumps. GL 81-14 requested licensees to consider EFW piping up to and including the second isolation valve. Oconee's position was that the piping is seismically qualified out through the first isolation valves, which are normally closed, and that this situation is consistent with other safety related systems at Oconee. Section 2.2 of the SER states that, during the evaluation process, a condition was identified outside Oconee's licensing basis in that certain manually operated isolation valves were not normally closed. This section also identified the following corrective actions as acceptable:

"Normally open boundary valves will be closed, or will be modified to be remotely operated, or analysis will demonstrate that failure of the piping beyond these valves will have no impact on system function."

Although five normally open LPSW valves were identified as deviating from the FSAR criteria, remotely operated valves 3LPSW-45 and LPSW-139 (the boundary valves between the safety and non-safety portions of the systems) were not identified as being unacceptable. The staff found single, remotely operated valves to be in accordance with the Oconee's design basis and acceptable for assuring seismic qualification.

We have also evaluated this issue from a risk perspective. The only time that valve 3LPSW-45 or valve LPSW-139 would have to close during an accident would be if a break occurred in the non-seismic piping downstream of the valve. This piping was designed and installed to power piping code USAS B31.1, which is the same code that the piping in the seismic portion of the system was designed to. The only difference is the lack of seismic design. The Electric Power Research Institute (EPRI) has collected and documented significant experience at power plants around the world that indicates piping systems designed to code survive seismic events. This position is also documented in many Probabilistic Risk Assessments throughout the nuclear industry. We consider the probability of the piping in question actually failing during a seismic event to be very low. It should be noted that, if the piping remains intact, there is no need for either of these valves to close.

Duke's Fossil/Hydro Department conducted an inundation study of Duke dams in 1993, in response to the FERC Emergency Action Plan (EAP) requirement. This study utilized the latest computer models for inundation evaluation, including a number of FERC-required conservative assumptions. In this analysis, the Keowee Dam was also assumed to fail due to overtopping of Keowee from the Jocassee failure. The resulting Oconee yard level flood was estimated to be 12 ft. This result is understandably different from the assessment made as part of the PRA study. An attempt was made to reproduce the results of the PRA study using assumptions consistent with original Civil Engineering evaluation in support of the PRA. This was unsuccessful since the details of the analysis could not be located and the responsible engineer was no longer employed at Duke.

Key differences in the methodology and assumptions used for the two studies are as follows:

- For the PRA study, routing of flood wave was accomplished by a step approach using an older version of National Weather Service DAMBRK program. For the FERC study, a later version of the DAMBRK program was used for routing the flood wave, using a full dynamic routing that makes use of downstream information. The FERC study considered consequences of flow restrictions downstream of Keowee Dam while the PRA study did not.

- The Keowee reservoir consists of the Keowee basin and the Little River basin that are connected by a man-made channel. Fifty-six percent of the storage volume for the Keowee reservoir is in the Little River basin. The man-made channel is located immediately upstream and west of the Keowee Dam. The PRA study takes full credit for the Little River basin acting as a storage basin in the event of a Jocassee Dam failure. Flooding of the Oconee yard would result from overtopping of the Oconee intake dike from the Little River basin. Keowee Dam is also overtopped but assumed not to fail. The FERC EAP study assumes the flood wave from the Jocassee dam failure will overtop and fail the Keowee Dam before a significant amount of the flood volume can be spread into the Little River basin. Water from both the Jocassee and Keowee reservoirs will flood the Keowee tailrace area and enter the Oconee yard prior to dissipating downstream.

In summary, a recent analysis for the Jocassee Dam failure using FERC-required modeling and assumptions produced more severe flooding than the magnitude considered in the PRA analysis. The Oconee yard flood level estimated in the PRA analysis cannot be reproduced as stated above. Nevertheless, it is believed that the 5 ft. SSF flood wall would provide some protection for best estimate types of dam failure modes. Considering that the estimated dam failure frequency is very small (1.58E-05/yr.), additional analytical effort to more precisely quantify the flood level is considered not warranted.

Concerning the description of the SSF door flood wall in the previous IPE submittal, the 8 ft. value is incorrect. The height of the wall is 5 ft. Nevertheless, the IPE analysis and conclusions are not affected by the 8 ft. versus 5 ft. discrepancy.

In 1995, Duke is scheduled to submit to the NRC a supplemental response considering seismic and fire events as part of the Individual Plant Examination for External Events. Duke will include at that time a re-analysis of the risk impact of external flood (Jocassee flood) to include the possibility that the yard flooding model might be greater than the capability of the 5 ft. SSF door flood wall. Further enhancements will be evaluated as a result of this re-analysis.

2. THE SSF WATERTIGHT GATE WAS OBSERVED WITH THE WATERTIGHT LATCH NOT PROPERLY SECURED.

The SSF "watertight" gate is designed to prevent unacceptable amounts of flood water through the passage in the wall surrounding the SSF. The design of this gate is such that the water pressure from the flood surge acts to seal the gate tighter against its rubber sealing gasket. In addition, the operators sent to the SSF in response to a flood scenario would be expected to close and secure the door following entry. The abnormal procedure will be reviewed to assure that there is adequate guidance regarding the closing and securing of the door.

Item No. 7 - The High Pressure Service Water system was not designed or maintained commensurate with its importance to safety.

RESPONSE:

1. **THE HPSW SYSTEM WAS NOT CLASSIFIED, CONSTRUCTED, TESTED OR MAINTAINED COMMENSURATE WITH ITS IMPORTANCE TO SAFETY.**

The High Pressure Service Water System (HPSW) System was not required to be classified or constructed QA1, which would have included the seismic design criteria of Section 3 of the Oconee FSAR. However, the system was designed and installed to power piping code USAS B31.1, which is the same piping code that several of Oconee's safety related piping systems are designed to. The only difference is the lack of seismic design. The Electric Power Research Institute (EPRI) has collected and documented significant experience at power plants around the world that indicates piping systems designed to code survive seismic events. This position is also documented in many Probabilistic Risk Assessments (PRA) throughout the nuclear industry.

The inspection report states that failure in virtually any portion of the system, during a loss of power event, would cause loss of Condenser Circulating Water (CCW) pump sealing and motor cooling water flow due to diversion of flow depleting the Elevated Water Storage Tank (EWST). If the event was a seismically induced break with a loss of offsite power (emergency onsite power was available), the system could maintain some level of system pressure because the standby HPSW pumps would start as level in the EWST dropped. Pressure regulating valves in the sealing lines to the CCW pump shafts would adjust to compensate for decreases in system pressure due to breaks. With both 6,000 gpm standby HPSW pumps running, it is likely that the flow diversion resulting from all but the largest breaks would not prevent adequate sealing and cooling flow to CCW. If the event was a seismically induced break with a loss of all ac power, the standby pumps would not be able to start and flow diversion would deplete the EWST. However, a seismic event, in combination with a loss of all ac power, is beyond Oconee's design basis.

The report is correct in stating that the HPSW pump discharge check valves, HPSW-2, -5, and -8, perform a safety function and should be tested accordingly. The check valves must not leak significantly during a loss of all ac power (SBO) or they could adversely impact EWST inventory. Leakage during an event involving a loss of offsite power (emergency onsite power was available) would have no effect since the standby HPSW pumps would start on low EWST level and force flow through the check valve in the forward direction to pressurize the system. A PIP (0-094-0329) has been written that identifies the need to add these valves to the In-service Testing Program and to the Quality Standards Manual (QSM). However, it should be noted that these valves are indirectly tested every time that PT/0/A/250/38, "Elevated Water Storage Tank Drain Test," is performed. This test is run to ensure that the EWST inventory will supply the required loads during a 4 hour SBO. If the check valves in question were leaking significantly, the EWST would not pass this test.

The report is correct in stating that the rotameters which measure CCW pump sealing flow and motor cooling flow may not provide accurate indication. A PIP (0-O94-0313) has been written to document the fact that a number of the rotameters were found to be reading high (offscale). This PIP will track the corrective action(s) for this problem to completion. The corrective action(s) will take into account the maintainability of the flow indication. It should be noted, however, that the CCW system operates continuously with at least two pumps in operation. If pump sealing and motor cooling flow were actually inadequate, it would adversely affect pump performance. This has not been the case to date.

2. **COMMUNICATION THAT HPSW WAS NOT CLASSIFIED, CONSTRUCTED, TESTED, OR MAINTAINED COMMENSURATE WITH ITS IMPORTANCE TO SAFETY WAS NOT TIMELY. ALSO, ENGINEERING CORRECTIVE ACTIONS WERE NOT OF SUFFICIENT SCOPE.**

The inspection report states that while Oconee's engineering organization recognized the safety significance of the HPSW System, this awareness had not yet been fully conveyed to the rest of the organization. HPSW performs several functions important to safety. It supports Emergency CCW siphon flow and CCW pump restart following a design basis LOCA. It also supports turbine-driven emergency feedwater pump operation during a 4 hour station blackout event. A number of HPSW valves were designated as "active" during the 1980s and included in Oconee's In-service Testing (IST) program. Plant personnel familiar with and/or responsible for testing were, therefore, aware of the relative importance of the system and its components.

More recently, the HPSW design basis document (DBD) was issued in September 1993 and an update to the Oconee specific appendix to the QSM was issued in October 1993. The HPSW DBD and the QSM are 2 separate, formal mechanisms for communicating the functions and importance of HPSW. It was understood at the beginning of the DBD program that, as these documents were created, the roles performed by various systems would be clarified. During the creation of the recent update to the QSM, a significant amount of discussion focused on understanding the reason specific equipment was being added, the impact of this increased scope of equipment, and the communication of this information. Due to the relatively recent release dates of these documents, it is to be expected that some operations and maintenance personnel were not yet aware of the relative importance of the system and its components. The most recent additions of SSCs to the QSM are in the process of being reviewed by the station maintenance engineering teams for impact. As discussed in our January 12, 1994 conference call with the NRC, a plan for implementation of these changes will be developed by April 15, 1994.

The inspection report states that the most recent draft revision to the QSM was of insufficient scope since it did not include the HPSW pumps or their discharge check valves. This statement is correct. A PIP (0-O94-0329) has been written to update the QSM, the IST program and the HPSW DBD.

The audit team recognized Oconee's efforts to resolve concerns associated with HPSW not being seismically qualified. This issue was evaluated as part of design study ONDS-327 and further evaluated in response to PIPs 0-O92-0084 and 0-O93-0695. However, the inspection report goes on to state that Oconee's evaluation on the seismic adequacy of the HPSW System failed to consider the actuation of any of the system's fire deluge functions due to a seismic event. It is not clear that Oconee's licensing basis requires consideration of the potential actuation of the fire deluge functions during a seismic event. Duke has not been able to find any documentation of the intent of Duke Power design engineers and of the AEC (NRC) at the time Oconee was licensed. The changes occurring in the design criteria and codes at the time, especially the seismic design criteria, apparently created some inconsistencies when the overall design is reviewed on a comprehensive basis. However, the original plant design was based on good judgement and sound engineering practices.

The spurious actuation of the fire deluge functions of HPSW, due to a seismic event, can be postulated. If the event were a seismic event only, the full operation of the HPSW System would not adversely impact the safe shutdown of the plant. The two standby, 6,000 gpm HPSW pumps would start as EWST level decreased to pre-set limits. HPSW would continue to provide its sealing and cooling function to the CCW pumps. The effect on Low Pressure Service Water (LPSW) pump NPSH would be negligible since the LPSW System would not be in its worst case configuration. If, instead of a seismic event alone, the event were a LOCA with loss of offsite power and concurrent seismic event which ruptured all the compressed air systems, then there would be an impact on LPSW pump NPSH. The effect of full HPSW flow has recently been incorporated into Revision 5 of engineering calculation OSC-2280, "LPSW $NPSH_A$ and Minimum Required Lake Level." The effect was to raise the required lake level 2 feet, from an elevation of approximately 784 to an elevation of approximately 786. This new, minimum lake level is being incorporated into a revision to SLC 16.9.7.

An evaluation was performed to determine the integrity of the HPSW System following a seismic event. Oconee recognizes that the HPSW System was not required to be designed and constructed to the seismic design criteria presented in Section 3.7 and 3.9 of the Oconee FSAR. However, a significant amount of experience at power plants around the world indicates that piping systems designed to piping codes typically survive seismic events. This experience has been collected and documented under the auspices of EPRI in EPRI Report NP-5617, "Recommended Piping Seismic Adequacy Criteria Based on Performance During and After Earthquake", and also represents the bases for the SQUG program.

As stated in the inspection report, Oconee's evaluation of the seismic adequacy of HPSW was based on the following 3 criteria:

1. A system walkdown which qualitatively verified that all major piping and equipment in the system was adequately supported and/or anchored.
2. A system walkdown which verified that no large pipes, which could undergo movement during a seismic event, were being restrained by small piping which could not move.

3. An evaluation which determined that any existing corrosion in the HPSW System did not threaten the structural integrity of the piping.

The inspection report raised a concern with the conclusions drawn on the extent of corrosion in the HPSW System. The service water systems (including HPSW) undergo UT testing as part of the Service Water System Piping Corrosion Management Program. Data from this testing is collected and compared against initial screening criteria. That data which does not pass the initial screening criteria is evaluated by stress analysis engineers to determine if the stresses are within code allowables. In all cases, the wall thicknesses were found acceptable.

A second concern raised by the inspection report was the scope of the walkdown itself. The report states that the walkdown only included that portion of the system from the HPSW water source to the CCW pumps. The scope of the walkdown was more extensive than stated in the report. It included the piping and equipment in the Turbine and Auxiliary Building (for instance, the HPSW piping to the HPI pump motor coolers was included). While the walkdown did not include every pipe in the Turbine and Auxiliary Buildings, it did include a significant, representative sample so that conclusions could be drawn based on the criteria stated earlier.

The inspection report raised a third concern with the threaded rod hangers on some of the HPSW piping at the CCW intake structure, their connection to the concrete, and the interval between these hangers. The hanger design described in the report is an acceptable support design used in seismically qualified piping systems installed in the Oconee construction time frame.

We are currently evaluating several options to enhance the reliability of LPSW suction following a design basis accident involving a seismic event. One group of options includes the installation of a safety grade sealing and cooling system for ECCW siphon flow and CCW pump restart following a seismic event. Upgrading portions of the existing CCW pumps is also under consideration as part of this group of options. A second group of options involves the installation of a separate, safety grade suction source for LPSW. Both groups of options eliminate the need for the current HPSW System to support ECCW siphon flow and CCW pump restart, in order to provide LPSW suction during a design basis accident involving a seismic event.

3. WEAKNESSES IN MANAGEMENT CONTROLS TO ASSURE THE HPSW SYSTEM IS CAPABLE OF PERFORMING ITS FUNCTION DURING A STATION BLACKOUT (SBO).

Oconee's SBO submittal takes credit for the EWST providing gravity flow cooling to certain loads for up to 4 hours. This capability is tested annually, using PT/0/A/250/38, "Elevated Water Storage Tank Drain Test." We recognize that the test procedure contains several weaknesses:

1. The calculation in the procedure for determining the capacity of the tank, in minutes, should use the minimum full level of 90,000 gallons, instead of the actual initial EWST level.
2. Stricter controls should be placed on re-performing the test if the original test failed due to leakage of the check valves on the HPSW pump discharge.
3. The procedure should direct the test performer to notify the operating manager when the calculated HPSW flow rate exceeds 375 gpm, not 500 gpm. In addition, the procedure should provide guidance for the performer and the operating manager on actions to take should the results of the test fall outside the acceptance criteria.
4. Clarification is needed of actions to be taken in the event of HPSW pump or jockey pump discharge check valve leakage or any other unacceptable leakage.

A PIP (0-O94-0307) has been written on these procedure weaknesses and the procedure will be revised. Past test procedures have been retrieved and the recorded data has been reviewed. This review indicated that the identified weaknesses do not change the conclusion arrived at upon completion of each test. None of the tests would have failed due to using 90,000 gallons in the calculation instead of the initial volume. None of the tests were re-performed due to check valve leakage.

The inspection report specifically discussed the test performed on February 27, 1993 which had an adjusted flow rate of 481.5 gpm (after accounting for HPI pump motor cooler flow); this did not meet the 240 minute acceptance criteria. The flow rate did not account for the 87 gpm going to the Instrument Air compressor, which would be automatically isolated following a loss of all ac power (481.5 - 87 = 394.5 gpm). The procedure was revised April 27, 1993 to account for this inventory loss. The inspection report states that even after accounting for this loss the test would still not have met the acceptance criteria. At the time of this test, functional checkout of the new Unit 3 Breathing Air compressors was in progress. While it cannot be determined for certain, HPSW may have been valved in for the Breathing Air System and would have accounted for an additional 26 gpm. This would have further reduced the adjusted flow to 368.5 gpm and allowed the test to meet its acceptance criteria.

The inspection report states that five of the CCW seal water rotameters and all of the bearing cooling water flows were greater than the values used to calculate the four hour availability of the EWST. The report also states that the operator rounds sheets contain no upper limits for these flows. A PIP (0-O94-0313) has been written to document the fact that a number of the rotameters were found to be reading high (offscale). However, test procedure PT/0/A/250/38 does not record or use the measured flow to the CCW pumps to verify the 4 hour availability of the EWST. Whatever the sealing and bearing cooling water flows are at the time of the test is reflected in the EWST level at the end of the test. There is no reason to expect these flowrates to vary significantly over time. The lines providing sealing water, for instance, contain pressure regulating valves which maintain the pressure, and the flowrate, constant. No additional value would be provided by adding an upper limit to the operator rounds sheets for these flows.

December 19, 1994

Duke Power Company
ATTN: Mr. J. W. Hampton
Vice President
Oconee Site
P. O. Box 1439
Seneca, SC 29679

SUBJECT: NOTICE OF VIOLATION
(NRC INSPECTION REPORT NOS. 50-269/94-31, 50-270/94-31, AND 50-287/94-31)

Gentlemen:

This refers to the Service Water System Operational Performance Follow-up Inspection conducted by Mr. Walter G. Rogers and others of this office on September 20 through November 3, 1994. The inspection included a review of activities authorized for your Oconee Nuclear Power Station. At the conclusion of the inspection, the findings were discussed with members of your staff identified in the enclosed inspection report.

The enclosed inspection report identifies areas examined during the inspection. The team assessed your progress in resolving the deficiencies from the initial Service Water System Operational Performance Inspection of 1993. Also, the team performed detailed reviews of your recent response to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

The team noted that while you have made some progress, your efforts in resolving a number of the original SWSOPI findings were not timely or complete as discussed in paragraph 3 in the inspection report details. Please indicate what actions you are and will be taking to improve corrective action capability in terms of timeliness and adequacy. Also, additional information is needed with respect to your GL response as outlined in paragraph 3.a of the inspection report. Please provide the additional response within 30 days of receipt of this letter.

Based on the results of this inspection, certain of your activities appeared to be in violation of NRC requirements as specified in the enclosed Notice of Violation (Notice). One violation reflected multiple corrective action program inadequacies. One violation concerned inadequate testing of the Emergency Condenser Cooling Water system's ability to supply the suction source for the Low Pressure Service Water pumps. Another violation dealt with insufficient scope of the American Society of Mechanical Engineer's Section XI testing program.

In addition, the enclosed inspection report identified certain activities that violated NRC requirements that will not be subject to enforcement action because your efforts in identifying and/or correcting the violation met the criteria specified in Section VII.B of the Enforcement Policy. These activities included inadequate procedural direction when operating the Auxiliary Service Water System.

8-4

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. In your response, you should document the specific actions taken and any additional actions you plan to take to prevent recurrence. After reviewing your response to this Notice, including your proposed corrective actions and the results of future inspections, the NRC will determine whether further NRC enforcement action is necessary to ensure compliance with NRC regulatory requirements.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

Albert F. Gibson, Director
Division of Reactor Safety

Docket Nos. 50-269, 50-270, 50-287
License Nos. DPR-38, DPR-47, DPR-55

Enclosures:
1. Notice of Violation
2. NRC Inspection Report

cc w/encls: (See page 3)

cc w/encls:
Mr. J. E. Burchfield
Compliance
Duke Power Company
P. O. Box 1439
Seneca, SC 29679

Mr. A. V. Carr, Esq.
Duke Power Company
422 South Church Street
Charlotte, NC 28242-0001

Mr. Robert P. Gruber
Executive Director
Public Staff - NCUC
P. O. Box 29520
Raleigh, NC 27626-0520

Mr. Robert B. Borsum
Babcock and Wilcox Company
Nuclear Power Generation Division
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Mr. J. Michael McGarry, III, Esq.
Winston and Strawn
1400 L Street, NW
Washington, D. C. 20005

Office of Intergovernmental
Relations
116 West Jones Street
Raleigh, NC 27603

County Supervisor of
Oconee County
Walhalla, SC 29621

Mr. Max Batavia, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, SC 29201

Manager, LIS
NUS Corporation
2650 McCormick Drive
Clearwater, FL 34619-1035

Mr. G. A. Copp
Licensing - ECO8O
Duke Power Company
P. O. Box 1006
Charlotte, NC 28201-1006

Ms. Karen E. Long
Assistant Attorney General
N. C. Department of Justice
P. O. Box 629
Raleigh, NC 27602

Distribution w/encl:
L. A. Wiens, NRR
M. W. Hodges, RI
G. E. Grant, RII
T. P. Gwynn, RIV
R. Carroll, RII
G. A. Hallstrom, RII
OB Oconee Read File

NRC Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, SC 29678

NRR assigned 12/6/94 for typing

RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:DRS	Consultant FOR
WRogers:wgr/tr	LMellen	LKing	CRapp	PKellogg	DPrevette
12/1/94	12/1/94	12/1/94	12/5/94	12/5/94	12/2/94
YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

RENO	NRC	RII:DRP	RII:DRS	RII:DRS		
TO	NAME	MSinkule	TPeebles	AGibson		
PDR?	DATE	12/2/94	12/6/94	12/6/94	/94	/94
YES NO	COPY?	YES NO	YES NO	YES NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: G:\RPTS\OCO9431.WGR

NOTICE OF VIOLATION

Duke Power Company
Oconee Nuclear Power Plant
Units 1, 2, and 3

Docket Nos. 50-269, 270, 287
License Nos. DPR-38, 47, 55

During an NRC inspection conducted September 20 through November 3, 1994, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," 10 CFR Part 2, Appendix C, the violations are listed below:

1. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures established shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

   The licensee's site directive on the problem identification process requires the initiation of a condition adverse to quality report PIP when there are errors in design bases documents and when documents are not updated.

   Contrary to the above, the licensee failed to promptly identify, adequately document and/or take adequate corrective action for several examples of conditions adverse to quality listed below:

   a. As of September 28, 1994, the corrective action to Deviation 50-269, 270, 287/93-25-01 of creating Keowee Hydroelectric Power Station operating procedures was inadequate in that numerous valves in the Keowee service water systems, including all the generator thrust bearing cooler inlet valves and drain valves WL-1, 2, 5, and 6 for both units, were omitted.

   b. As of September 28, 1994, the corrective actions to problem investigation form 0-093-0986 for drawings at Keowee not reflecting the as-built condition of the facility and to Violation 50-269, 270, 287/93-25-12C of verifying and reflecting the as-built condition of Keowee mechanical systems on drawings were inadequate, in that the revised drawings failed to show a drain valve on Unit 2 downstream of valve 2WL-3.

   c. As of October 23, 1994, technical errors associated with actual available suction source inventory in calculation OSC-0864, "RC System DH Removal following a Loss of Intake Structure," although known by engineering personnel, were not identified as a condition adverse to quality through the initiation of a problem investigation form.

Notice of Violation

d. As of October 11, 1994, the condition adverse to quality corrective actions to problem investigation form 0-94-213 of poor condenser cooling water pump cooling/sealing flow rotameters material condition were inadequate in that 10 of the rotameters contained significant slime contamination, one contained 3 clams and 11 were pegged at the high end of the scale.

This is a Severity Level IV violation (Supplement I).

2. 10 CFR 50, Appendix B, Criterion XI, "Test Control," requires in part that operational test procedures be established to demonstrate that systems and components will perform satisfactorily within the acceptance limits contained in applicable design documents.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented procedures. The procedures shall include appropriate quantitative or qualitative acceptance criteria.

Technical Specification Surveillance Requirement 4.1.2, Table 4.1-2, Item 7, specifies a condenser cooling water system gravity flow test be performed each refueling.

FSAR Section 9.2.2.2.1, "Condenser Circulating Water System (CCW)," states that the CCW systems are designed to take advantage of the siphon effect so the pumps are required only to overcome pipe and condenser friction losses. Under a loss-of-power situation, the emergency discharge line will automatically open and the CCW system will continue to operate as an unassisted siphon system... the siphon system is the Emergency Condenser Circulating Water (ECCW) system and can be divided into two distinct parts. The "first siphon" takes suction from the CCW intake canal and supplies flow to the CCW crossover header in the Turbine Building basement, where the LPSW System takes its suction... In a loss of off-site power (LOOP) situation, the CCW pumps will be tripped by a load shed command from the Engineered Safeguards System and the first siphon is required to supply suction to LPSW System until a CCW pump can be manually restarted by the control room operator."

Contrary to the above, as of October 27, 1994, the condenser cooling water system gravity flow test, an operational test procedure performed each refueling by the licensee, did not demonstrate that the system would perform satisfactorily within the acceptance criteria identified in the FSAR (applicable design document). Specifically, the test would not demonstrate that the portion of the condenser cooling water system providing the suction source for the low pressure service water system when the condenser cooling water pumps are off will perform satisfactorily within the acceptance limits contained in the FSAR section 9.2.2.1 and did not contain appropriate quantitative or qualitative acceptance criteria for that portion of the condenser cooling water system.

This is a Severity Level IV violation (Supplement I).

3. 10 CFR 50.55(a) paragraphs (f)(1) and (4) requires in part that the safety-related valves of a pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, be classified as ASME Code Class 1, 2 or 3 and meet the test requirements set forth in section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda.

   Contrary to the above, as of October 27, 1994, atmospheric relief valves, check valves from the ASW pump to the HPI motor coolers, and the turbine oil cooling bypass valve which perform safety-related functions, were not tested to the requirements set forth in section XI of the ASME Boiler and Pressure Vessel Code.

This is a Severity Level IV violation (Supplement I).

Pursuant to the provisions of 10 CFR 2.201, Duke Power Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission for items 1, 2 and 3, ATTN: Document Control Desk, Washington, DC 20555, with a copy to the Regional Administrator, Region II, and a copy to the NRC Resident Inspector, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved.

If an adequate reply is not received in the time specified in this Notice, an order or demand for information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Dated at Atlanta, Georgia
this 11th day of December 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W., SUITE 2900
ATLANTA, GEORGIA 30323-0199

Report Nos.: 50-269/94-31, 50-270/94-31, and 50-287/94-31

Licensee: Duke Power Company
422 South Church Street
Charlotte, NC 28242

Docket Nos.: 50-269, 50-270, and 50-287 License Nos.: DPR-38, DPR-47, and DPR-55

Facility Name: Oconee Nuclear Station Units 1, 2, and 3

Inspection Conducted: September 20 through November 3, 1994

Inspector: Walter G. Rogers, Team Leader — Date Signed: 12/16/94

Accompanying Personnel: Paul Kellogg
Larry Mellen
Larry King
Curt Rapp

NRC Consultants: D. Prevatte

Approved by: Thomas A. Peebles, Chief — Date Signed: 12/16/94
Operations Branch
Division of Reactor Safety

SUMMARY

A follow-up to the initial Service Water System Operational Performance Inspection (SWSOPI) of 1993 was conducted on September 20 through November 3, 1994 according to NRC inspection module 37550 and Temporary Instruction (TI) 2515/118.

RESULTS

The NRC Temporary Instruction for Service Water Inspections (Safety Issues Management System item TI 2515/118), was not closed since the licensee has yet to accomplish a number of committed Generic Letter 89-13 actions.

Numerous corrective actions from the previous inspection had not been adequately accomplished or completed consistent with committed schedules. Testing of the suction source capability for the low pressure service water pumps via the siphon mode was not adequate. Deficiencies were identified in the operational guidance and testing of the Auxiliary Service Water system. Also, infrastructure weaknesses persist inhibiting efforts to keep design calculations current.

Enclosure 2

REPORT DETAILS

1. Inspection Objectives

Numerous problems identified at various operating plants in the country have called into question the ability of the SWSs to perform their design function. These problems have included: inadequate heat removal capability, biofouling, silting, single failure concerns, erosion, corrosion, insufficient original design margin, lapses in configuration control or improper 10 CFR 50.59 safety evaluations, and inadequate testing. NRC management concluded that an in-depth examination of SWSs was warranted based on the identified deficiencies.

In the original SWOPI conducted in 1993, the team focused on the mechanical design, operational control, maintenance, and surveillance of the SWS and evaluated aspects of the quality assurance and corrective action programs related to the SWS. Numerous deficiencies were identified by the team which the licensee responded to in letters on March 14, April 20, May 12 and September 1, 1994. This inspection's primary objectives were to:

- assess the licensee's progress in accomplishing the committed actions specified in these letters,

- verify whether the corrective actions resolved the identified deficiency and did not compromise SWS performance requirements or design bases,

- assess the licensee's planned and completed (since February 1994) actions in response to Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment," and

- ascertain why there were discrepancies between information provided to the team and a recent motor operated valve inspection about LPSW turbine building isolation valves thrust capabilities and types of tests performed on these valves in the past.

Another issue not associated with the SWS follow-up was included as part of this inspection. This issue dealt with a postulated feedwater piping break within containment. The licensee had provided information and future corrective actions to mitigate the consequences of such an event. Therefore, another primary objective of this inspection was to understand the licensee's present response capability to such an event and ascertain the status of any design changes to reduce the consequences of such an event.

The team observations and concerns identified are described in sections 3 through 7 of this report. Personnel contacted and those who attended the exit on October 27, 1994, are identified in Attachment A. Acronyms and abbreviations are identified in Attachment B.

Report Details

2. General Description of SWSs

The SWSs at Oconee encompass numerous systems. These are the CCW (including the ECCW subsystem), LPSW, HPSW, ASW, SSF DG, HVAC, and steam generator cooling, the SSF submergible pump subsystem, Keowee hydroelectric station generator air, thrust bearing, packing box, and main bearing cooling.

The CCW system is common to all three units and takes suction from the Lake Keowee intake canal. Twelve pumps (four per unit) supply a common cross-connected 42-inch discharge header from which numerous other SWSs take suction. From this header, cooling water passes through the three condensers. Upon leaving the condensers, the water discharges through six lines (two per unit) and returns to Lake Keowee upstream of the intake canal.

A subsystem of CCW is the ECCW system. If the CCW pumps lose power, ECCW actuates establishing siphon or gravity flow from the intake canal to the 42-inch header and through the condenser sections. Emergency condenser discharge lines connect the condensers with the Keowee hydroelectric station's tailrace. Prior to entering the tailrace, all the discharge lines connect into one line. ECCW actuation involves the automatic closure of the condensers' normal outlet valves, opening of the condensers' emergency outlet valves and opening the emergency discharge valve to the Keowee tailrace, CCW-8, located in the common discharge piping. The high points of the ECCW piping are connected to a vacuum priming system which would remove air entrapped within the system that could impede siphon operation. The licensee considers CCW supplying the LPSW pumps as the first siphon and CCW passing through the condensers as the second siphon.

The CCW system performs two distinct safety functions during the LOCA/LOOP event. First, it provides a suction source for other systems including the safety-related LPSW system, and second it provides cooling water to the condenser to remove decay heat in the ECCW mode. The CCW pumps contribute to those safety functions in two ways. First, when power is lost and they are not operating, they provided a siphon conduit from the intake canal to the CCW piping from which the LPSW takes suction, and to the condenser for the ECCW system. Second, at the time when the pumps can be restarted (up to 1/2 hours per emergency procedures), they continue to provide water for these same functions. Since dissolved air will tend to come out of solution when the system is in the siphon mode, at least one of the CCW pumps must be operated after power is restored in order for the water to continue to be supplied to the CCW piping.

Within the intake canal, is an underwater dam which can trap approximately 67,000,000 gallons of water if Lake Keowee were to fall below the 770-foot level. With the CCW pumps operating, the system is capable of recirculating water from this impounded area, through the condensers, through the condenser emergency discharge lines and through normally closed valve, CCW-9, which discharges into the intake canal.

The LPSW system provides cooling to the RBCUs, LPI coolers, the motor and turbine driven EFW pump coolers, HPI pump motor coolers, the control room chilled water system, numerous room coolers, and nonsafety-related turbine building loads. Units 1 and 2 share three 15,000 gpm pumps with one pump capable of being powered from two separate safety-related busses. The Unit 3 LPSW system has two 15,000 gpm pumps. The LPSW pumps take a suction from the 42" CCW discharge header within the turbine building. The Unit 1/2 LPSW pumps discharge into a common header that splits into two supply lines; one supply line for each unit. The unit supply lines further divide into two separate headers supplying the two trains of safety-related equipment. The two equipment supply lines then interconnect into a common line which enters containment. This common line then splits into three parallel lines; each line supplying one RBCU. These three RBCU supply lines reconnect into one line on the discharge side of the RBCUs before exiting containment. Also, branching from the common discharge header is a supply line to the turbine building loads. The turbine building supply line then splits to provide cooling to each unit's turbine building equipment. The Unit 3 RBCU and turbine building cooling arrangement is similar. A normally closed crosstie line allows either LPSW system to supply the discharge header of the other LPSW system.

The HPSW system normally functions as the site's fire protection system. The system is composed of three pumps, an elevated storage tank, and interconnecting piping to fire protection deluge valves throughout the site and to the CCW pumps. The three pumps, two 6000 gpm capacity and one jockey, take suction from the 42" CCW discharge header. The jockey pump maintains system pressure. The other two pumps are to make up lost water inventory in the 100,000 gallon capacity elevated storage tank. The system constantly supplies cooling and sealing water to the CCW pumps. The system is capable of supplying cooling water to specific components normally cooled by the LPSW system such as the HPI pump motor coolers and the EFW pumps. Though at reduced capacity, the system can provide backup cooling to the LPSW system through interconnections at the discharge of the LPSW pumps.

ASW is a system common to all three units. It is designed to provide cooling to the steam generators. The system was originally designed for the loss of the intake canal/structure. However, following NUREG 0737 review of the facility for tornado vulnerabilities, the system was credited in the July 28, 1989, NRC SER to mitigate the consequences of a tornado. The system can supply cooling water to the high pressure injection pump motors if the low pressure service water system is unavailable. The ASW system consists of a suction connection to the unit 2 CCW system at elevation 759.5, a low head, high capacity pump, piping, and manual valves connected to the emergency feedwater piping in the penetration rooms of all three units. The ASW pump is operated from the safety-related Aux Service Switchgear, and the alignment to the SGs is accomplished manually.

The SSF is a separate onsite building housing the necessary equipment to maintain all three units in a safe shutdown condition following turbine building flood, fire, sabotage, and certain classes of tornados or station blackout. The SWS portion of the SSF is composed of a high head, low capacity pump and interconnecting piping to all steam generator EFW discharge lines, solenoid operated flow control valves in the discharge lines to the steam generators, a pump and piping to cool a tandem diesel with a common generator, two pumps with a condenser unit to cool the HVAC within the SSF, and a moveable submersible pump. The SSF ASW, HVAC, and DG pumps take suction from the Unit 2's CCW pump's discharge header. The HVAC and DG pumps discharge to the CCW header. There is an option to divert the DG pump discharge water to the yard drainage system when high temperature constraints warrants. The submersible pump allows replenishment of the CCW header from the intake canal.

Lake Keowee is the motive and cooling source for the two hydroelectric generators which function as Oconee's onsite emergency power. Water flows from a common penstock, through the turbines and into the tailrace. Cooling flow comes from a single pipe located in the penstock. Once the line enters the building housing the hydroelectric units it splits into two lines, one for each unit. Cooling flow for the turbine bearing oil cooler, the stuffing box, eight thrust bearing heat exchangers, and six generator air coolers comes from the unit specific main line.

3. Follow-up on Previously Identified Items

The team reviewed all outstanding violations, unresolved items, and inspector follow-up items identified in the original SWSOPI. For violations, the corrective actions described by the licensee were evaluated for adequacy and completeness. The team reviewed whether the implementation of the corrective actions were accomplished within the time frames specified by the licensee's NOV response and strengthened the licensee's QA program procedures or practices to prevent recurrence. If the corrective action deadline had not occurred, the status of licensee efforts was ascertained. Unresolved items were updated or dispositioned depending upon the status of NRC review of these matters. The team continued to evaluate the matters selected for inspector follow-up. The status of licensee efforts in these matters also were ascertained. The team reviewed all matters to ensure the generic implications, if applicable, were addressed. Inspection findings were:

a. (Open) DEV 50-269, 270, 287/93-25-01 (Deviation A in NRC Inspection Report 93-25), "Failure to Adequately Perform SWS GL Actions." The licensee's original response to GL 89-13 was not inclusive of some of the systems utilizing service water.

   In response the licensee indicated a revised response to the GL would be submitted by September 1, 1994. Included in the September 1, 1994 GL response would be those actions necessary to deal with Keowee stagnant or intermittent flow. In addition, a schedule for implementation of modifications and testing of Keowee heat exchangers

would be provided by September 1, 1994. Finally, operating procedures were created for the Keowee thrust bearing oil heat exchangers and the generator air coolers by February 4, 1994.

The revised GL response was submitted by September 1, 1994. However, there were numerous omissions and inadequacies associated with this response. Therefore, the licensee continues to have not fully responded to the GL. The omissions and inadequacies outstanding are:

- Action I - No date was provided as to when the CCW system hydraulic model would be benchmarked. No date was provided as to when the HPSW system hydraulic model would be benchmarked. The frequency of simultaneous SSF SWS pump testing was not specified.

- Action III - No administrative controls existed to ensure the committed inspection program for the SSF, ASW, and Keowee SWSs would be accomplished. System engineers interviewed who were to do the inspection program were unaware of their inspection responsibilities. There was no criteria by which to judge piping condition acceptability. Also, no technical bases for the adequacy of the piping inspection program in terms of scope, frequency, or corrective action could be ascertained.

- Action IV - No date was provided as to when the Keowee single failure analysis would be completed.

- Action V - The GL response referenced a Duke Power Company letter to the NRC dated April 20, 1994, as providing the discussion on procedures and training. However, this letter was not applicable to Action V. Also, the presently docketed correspondence on this matter indicated that SWS procedures were receiving a two-year review. This was not correct. The licensee had revised their procedure review cycle to as long as every six years following a change to their QA plan.

As indicated above, the licensee's corrective actions to the deviation included the creation of operating procedures for Keowee thrust bearing oil heat exchangers and the generator air coolers. However, the newly created Keowee operating procedures omitted numerous valves in the Keowee SWSs. These valves included all the generator thrust bearing cooler inlet valves, and drain valves WL-1, 2, 5, and 6 for both units. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected. The licensee had developed procedures, but they were not complete. Failure to adequately correct this condition adverse to quality is an example of Violation 50-269, 270, 287/94-31-01A, "Inadequate Corrective Action Controls."

b. (Closed) URI 50-269, 270, 287/93-25-02, "Turbine Building Isolation Single Failure Vulnerabilities."

The licensee's original SER stated that the plant was designed such that no single failure would prevent safety system functionality in the event of a LOCA and a LOOP caused by a seismic event. NRC reviewed the licensee's original SER and because seismic event is a qualification standard and not an event that requires mitigation, interpreted that single failure did not apply to seismic event. Because the reactor coolant system was seismically qualified, a seismic event did not result in a significant challenge to core cooling capability. Additionally, NRC credited the SSF as providing the same safety function as LPSW in the licensee's EFW SER. Based on NRC's interpretation of seismic event as a qualification standard and NRC's credit of the SSF in the licensed's EFW SER, no further licensee action is required. This item is closed.

c. (Open) VIO 50-269, 270, 287/93-25-03 (Violation B in NRC Inspection Report 93-25), "Failure to Perform Adequate Calculations and Evaluations to Support Facility Design." There were seven parts associated with this violation.

(1) In Item 1, the NPSH of the LPSW pumps was not adequately considered as a design input in that calculation DSC-5019 was accepted by the license's engineering personnel with inadequate NPSH and inadequate technical justification.

The licensee contested this example of the violation.

To better understand the safety significance associated with this matter the licensee performed a PRA of the conditions necessary to have inadequate NPSH. The analysis indicated the event was not a significant accident precursor. However, the information provided by the licensee in their docketed correspondence continued to indicate that there was inadequate consideration of net positive suction head as a design input for the Low Pressure Service Water pumps. This matter remains open.

(2) In Item 2, no administrative control existed to assure the LPSW's pump flows used as hydraulic computer model input for the LPSW system remained valid during quarterly testing of the LPSW pumps.

The licensee responded that the test procedures for LPSW and other select systems, which do not have clear test acceptance criteria for pump performance, would be revised by September 1, 1994. In the procedure revision system, engineers would compare quarterly pump test data, along with full system flow test data, against computer models and other calculations to ensure the validity of design basis analyses.

The licensee had included guidance in the quarterly LPSW pump test procedure ensuring the test results did not invalidate the hydraulic model. Discussions with the system engineer indicated this guidance would be removed and the LPSW pump curve used in the hydraulic model degraded. The amount of degradation would be determined by reviewing the decrease in LPSW pump performance data based on the previous 10-year trend. The quarterly LPSW pump test acceptance criteria will be modified to reflect the LPSW pump degradation used in the hydraulic model. This violation example is closed. However, transition from the test procedure to the hydraulic model is considered an Inspector Follow-up Item 50-269, 270, 287/94-31-02, "Hydraulic Model Controls Transition."

(3) In Item 3, the commercial grade evaluation for Belzona as a suitable material for application to the Unit 2 Reactor Building Cooling Unit tubes was inadequate.

In response the licensee documented that an effort was underway to obtain dynamic material property data and to analyze Belzona for usage under cyclic loading and LOCA conditions by December 16, 1994. Also, a modification to replace the RBCU cooling coils had been completed on U3 and U1, and was currently underway on U2.

The licensee was continuing to analyze Belzona with the targeted completion date of December 16, 1994. Also, the RBCU cooling coils had either been replaced or were being replaced during the inspection period. This example remains open pending completion of the licensee's Belzona analysis.

(4) In Item 4, the design basis of the ECCW system was not adequately translated into design documents in that the calculations supporting ECCW decay heat removal capability did not include numerous aspects of the design that would reduce decay heat removal capability.

Two calculations were involved. The first calculation was OSC-2346, "ECCW System Performance Evaluation." This calculation was generated to show that the main condenser in the ECCW mode had the capacity to transfer the required decay heat without exceeding the condenser pressure limits or causing flashing on the CCW side which could cause loss of siphon. The initial inspection report noted that this calculation contained several non-conservatisms. As partial response to the violation example the licensee revised the calculation on September 29, 1994 (Rev. 5).

The new calculation was poorly performed and contained new errors or did not completely address the original concerns. These deficiencies included:

- On page 32o, the specific volume of air was used in a computation to determine the amount of air that would come out of solution. The value used was 1L/kg - the correct value for water, but not for air. The specific volume of air at one standard atmosphere is 847L/kg. Therefore, the result was non-conservative by a factor of 847. When this was brought to the responsible engineer's attention, he re-reviewed the calculation and found that in the computation of the dissolved air on page 32n, there was another offsetting error in the number of gram moles/liter by a factor of 502. The net result was that the air coming out of solution, before correction for other non-conservative factors, increased from 940 ft^3 to 1,590 ft^3. Per the calculation, the non-conservative limit on outgassing was 1,584 ft^3. Therefore, this result was unsatisfactory.

  In subsequent discussions with the licensee the reason for the gram moles/liter error was due to an incorrect reference. The team inquired what actions had been taken by the licensee to deal with the ramifications of such an error and whether the matter had been identified as a condition adverse to quality. The licensee responded that the chemistry section (where the reference had been acquired) had been contacted but, no PIP had been initiated. The licensee personnel stated that they were waiting to understand all the problems associated with this calculation before initiating a PIP.

- Outgassing of the CCW water had not been accounted for in the calculation initially reviewed by the team in 1993. The revised calculation did consider outgassing but not completely. Specifically, the calculation only considered the outgassing effects due to the increase in temperature as the CCW water passed through the condenser; it did not consider the outgassing due to the reduction in pressure due to the siphon effect. Also, the calculation did not consider the expansion of the air after it had outgassed due to the reduction in pressure from atmospheric, again, due to the siphon effect.

- On page 32m, the revised calculation determined the allowable volume of air to be outgassed based on the volume of the condenser waterboxes above the ECCW outlet piping. This computation was based on the height from the top of the waterbox to the centerline of the outlet pipe, 7 feet. It should have been to the top edge of the pipe, 6.5 feet, which is the point where the pipe would begin to be uncovered. This reduced the allowable outgassing volume by 7.1 percent from what was calculated.

- The revised calculation did not reconcile competing assumptions associated with the number of available tubes in

the condenser for heat transfer. The revised calculation assumed one condenser section was out of service. However, 1/2 of a section can be taken out of service under operating procedures without engineering involvement. Further tube reduction can be accomplished by tube plugging and Amertap ball clogging. The licensee responded that only 4.5 percent (approximately 700 tubes) were plugged, and this would produce an insignificant effect. This response only accounted for the tubes that had been taken out of service and those which the licensee felt might trap the Amertap balls due to denting or other damage. In generating that number, the licensee did not consider that in the siphon flow mode the differential pressure across the condenser would be extremely low compared to normal operation - sufficiently low that none of the Amertap balls were likely to pass through the tubes. Therefore, all of the balls which entered the inlet waterbox would potentially plug tubes. The licensee had not verified that the assumed out of service condenser section would compensate for the other non-conservative mechanisms capable of reducing the number of tubes for heat transfer.

- The original calculation assumed an even flow split to all three condensers, whereas the piping configurations were significantly different for each condenser; therefore, this assumption was not necessarily valid. This assumption continued in the revised calculation. The licensee responded that the three units displayed similar condenser flow values which are considerably higher than the required flow value. However, the data cited is not for conditions similar to the conditions described in the calculation; i.e., flow through all three of the condensers at the minimum flow rate. Therefore, the conclusion was not supported by relevant data.

The responsible engineer stated the calculation would be re-performed making the appropriate corrections and also taking into consideration a number of significant conservatisms that had not been considered before. Revising the calculation is considered Inspector Follow-up Item 50-269, 270, 287/94-31-03, "Reperformance of Calculation OSC-2346."

The second calculation was OSC-2349, "CCW Intake Piping Degassing in the ECCW Mode." The calculation was performed to verify the CCW system's "first siphon" capability for the four-hour SBO event. It considered the air inleakage into the system and air outgassing from the water that would tend to break the siphon, and it established the acceptance criteria for the ECCW system flow test. The initial inspection report noted that this calculation contained several discrepancies and nonconservatisms

as well as several conservative assumptions; the net effect of which could not be determined without a rigorous re-performance of the analysis.

In the response letter of May 12, 1994, the licensee stated that this calculation was intended to be applicable for the SBO event only, and not for the LOCA/LOOP event; a separate calculation would be generated to analyze that scenario. Therefore, the 30,000 gpm flow assumed for this calculation was appropriate. It was also stated that the calculation's use of the incorrect atmospheric pressure would have an insignificant effect on the analysis. Therefore, no changes were required.

Considering this response, the licensee had no analysis or test which verified the "first siphon's" design basis safety function for events involving LOOP such as the LOCA/LOOP event (90,000 gpm for 1.5 hours) during the initial inspection. Subsequently, a new calculation was generated (OSC-5670, Rev. 1, March 17, 1994, "Required Number of CCW Intake Flow Paths") to provide the analytical basis for the "first siphon" for the LOCA/LOOP event.

This new calculation revealed a new requirement for the system configuration to support the design basis conditions; at least two CCW pump discharge valves on the same eleven foot CCW piping section were required to meet the siphon flow requirements.

Also, the new calculation contained the following minor deficiencies which were discussed with the responsible engineer and were found to have no effect on the results. The responsible engineer indicated the required revisions would be made to correct these deficiencies. The deficiencies were:

- The test data used as benchmark data in the calculation was of questionable validity. However, the team reviewed additional test data which appeared to be valid and would provide appropriate benchmarks.

- The calculation did not address the differences in water temperature and flow rate between the test data used as benchmark cases and the design basis case. Although this left the calculation incomplete, the effects appeared minimal.

Further team review as to the confirmation of the new calculation via testing did not exist. There was a test that acquired test data that would be imputed into the LPSW hydraulic model. However, this test did not contain acceptance criteria associated with first siphon performance. Therefore, the test data was not evaluated as to whether the siphon would operate in the most demanding design bases conditions (lowest lake level, highest required flow, etc.). Also, the first siphon test did not assure proper protection from air intrusion via a pump flange connection

normally underwater. This connection would be uncovered at lower acceptable lake elevations. The only other test procedure associated with the ECCW system verified only the second siphon, not the first, with all test acceptance criteria associated with the second siphon.

Independently, the licensee had identified that the ECCW system testing did not accomplish the technical elements of the team's findings. However, the licensee viewed these deficiencies as areas for improvement to the present testing, not as deficiencies. This position was partially based on the licensee's belief that the Technical Specifications did not require a test of the first siphon capability of the ECCW system.

Technical Specification Surveillance Requirement 4.1.2, table 4.1-2, item 7, specifies a condenser cooling water system gravity flow test be performed each refueling. 10 CFR 50, Appendix B, Criterion XI, "Test Control," requires, in part, that operational test procedures demonstrate systems and components will perform satisfactorily within the acceptance limits contained in applicable design documents. 10 CFR 50, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by procedures which include appropriate quantitative or qualitative acceptance criteria. Failure to perform a test that included the first siphon as part of the test acceptance criteria and assuring that such acceptance criteria bounded the acceptable operating range of the ECCW first siphon is Violation 50-269, 270, 287/94-31-04, "Inadequate LPSW Suction Source Testing Via the ECCW System."

(5) In Item 5, the design basis of the CCW's system capability to withstand loss of Lake Keowee was not translated into any design document.

In response the licensee indicated that a "loss of lake" analysis would be performed and completed by June 1, 1995. The team confirmed the licensee had targeted June 1, 1995, for completion of the analysis.

(6) In Item 6, the design basis of the LPSW's system capability to function as described in Case B of Abnormal Procedure AP/1/A/1700/13, "Loss of Condenser Circulating Water Intake Canal/Dam Failure," Step 5.5.1, was not translated into any design document.

During this follow-up inspection, the licensee stated that the analysis had not been started, pending completion of the heatup analysis of the pond area for the loss-of-dam event (see part E above). This was based on the assumption that the latter analysis was required for input to the former. The team pointed out that these two heatup conditions occur in completely different time frames - the former over a period of minutes, or

at most hours, the latter over a period of days. Therefore, the LPSW heatup analysis could be performed starting with the initial conditions of the pond, and could be performed in parallel with the pond heatup analysis. The licensee indicated the analysis would be completed by June 1, 1995.

(7) In Item 7, the design basis of the SSF ASW system capability to remove decay heat was not adequately translated into design documents in that a minimum flow less than required by 23 gpm per steam generator pair was established in calculation OSC-4171.

The calculation had been revised and this concern has been corrected.

In summary, Items 2 and 7 were closed. All other items remain open pending licensee actions.

d. (Open) VIO 50-269, 270, 287/93-25-04 (Violation A in Inspection Report 93-25), "Inadequate Evaluation of Conditions adverse to Quality by Engineering." There were two parts associated with this violation.

(1) In Item 1, the evaluation of PIP 92-454 for a postulated water hammer within the LPSW piping downstream of the RBCUs did not include the consequences on the structural integrity of the piping.

In response to the violation, PIP 93-1031 was written and OSC-6020 performed indicating turbine building flood was the bounding event. To eliminate the water hammer a flow orifice would be installed downstream of the potential cavitation and a schedule for modification implementation would be provided by September 1, 1994.

The provided documentation indicating projected orifice installation at the next respective refueling outages. However, the documentation also indicated that corporate engineering had been requested to conduct detailed computer analysis to determine if the water hammer would occur on the discharge of the RBCUs. Based on corporate engineering results, the discharge from the RBCUs was determined to always be in a condition of two-phase flow. Any waterhammer would be dampened by the two-phase state. Therefore, installation of a flow orifice to increase downstream pressure would have the detrimental effect of eliminating the dampening effect of the two-phase state. The results of these tests were to be available at a later date.

(2) In Item 2, the evaluation for corrective action to design study ONDS 327 and Problem Investigation Report 92-084 of the postulated response of the HPSW system to the maximum hypothetical earthquake did not include spurious fire protection component activations. In response to the violation, the

licensee stated calculation OSC-2280,"LPSW NPSHa and Minimum Required Lake Level," was revised to account for HPSW system malfunction. Consequently, SLC 16.9.7 was revised.

The team confirmed the calculation had been performed and the SLC revised. Also, the licensee indicated that the deluge valve had been included in the components to be qualified via SQUG.

Item 1 of this violation remains open pending orifice installation or completion of additional analyses showing the orifices are not required. Item 2 is considered closed.

e. (Open) IFI 50-269, 270, 287/93-25-05, "Additional Validation of RBCU Evaluation Inputs." There was two inputs in question dealing with LPSW flow orifice accuracy and assumptions on air flow distribution

The licensee reinstalled the flow orifices after replacing the RBCUs in all units. The team inspected the piping that was replaced on the inlet and outlet of the RBCUs for Unit 1 and determined that the corrosion buildup was minimal. A review of the calculations to show that the new coolers were operable used data from the cooling water side to calculate the heat removal capability. It was assumed that the heat removal was the same for the air side and this was used to calculate the air flow. The team's review of the calculations showed that there was adequate margin in the coolers' heat removal capability.

The team observed licensee attempting to take air flow measurements during the current refueling outage but the attempt was poorly coordinated and not accomplished. The licensee stated air flow measurements would be obtained during the next refueling outage. This inspector follow-up item remains open pending the licensee acquiring the air flow data.

f. (Open) IFI 50-269, 270, 287/93-25-06, "Actions to Improve Operator Responses to Abnormal Events." There were three parts associated with this item.

(1) In part A, the prerequisite for the total loss of LPSW was no LPSW pump operating; not inadequate LPSW flow. The licensee revised the prerequisite of the procedure with no problems identified.

(2) In part B, Abnormal Procedure AP/1/A/1700/13, "Loss of Condenser Circulating Water Intake Canal/Dam Failure," had several weaknesses. The licensee generated PIP 0-094-0514 to revise the procedure and improve operator training to address these concerns.

Operator training had been revised and taught to the 1994-95 licensed reactor operator and senior reactor operator class. Also, other licensed operators would be taught during the 1994-95

PTRQ classroom training. The procedure revision was scheduled to be completed by November 30, 1994.

(3) In part C, potential weaknesses in the operator guidance for response to a severe tornado were identified. The licensee indicated that a tornado would be considered as part of an upcoming exercise to ascertain whether further operator guidance was warranted.

In summary, part A is closed. Part B will remain open until the revision to Abnormal Procedure AP/1/A/1700/13, "Loss of Condenser Circulating Water Intake Canal/Dam Failure," is issued. Part C will remain open until the exercise is accomplished and any additional guidance provided.

g. (Open) VIO 50-269, 270, 287/93-25-08 (Violation D in NRC Inspection Report 93-25), "Inadequate SSF and ECCW Testing." There were three parts associated with this violation.

(1) In Item 1, ECCW flow test procedure PT/1/A/0261/07 did not account for the potential ± 2,000 gpm error which could result from the method used to measure flow - observation of the impact point of the ECCW discharge flume.

In response the licensee committed to produce an analysis by July 1, 1994, and incorporate its results in the test procedure by August 1, 1994.

A revision to the analysis, OSC-5629, "ECCW Test Acceptance Criteria Inputs," was issued on July 13, 1994. There were no technical discrepancies associated with the new analysis. However, the results were not incorporated into the test procedure at the time of the follow-up inspection. The licensee stated that they had intentionally delayed incorporation until just before the next required procedure performance at the next refueling outage so that all other ensuing changes can be incorporated at the same time in one overall revision. The NRC had not been informed of the implementation schedule change. The licensee indicated a letter would be submitted on the schedule change.

(2) In Item 2, the preoperational test program for the SSF's SWS and the post-construction flushing procedure for the SSF's discharge lines to all the SGs were inadequate.

In response the licensee committed to performing reverse flow testing of each unit's SSF ASW supply piping. Adequate flush velocities would be achieved during the testing along with water samples to verify the flush was adequate.

Procedures were revised and the first test (Unit 1) had been completed. The preliminary results appeared to address the

initial concerns. There was provision for assuring the [illegible] velocities would be adequate and that condensate would be used to flush the lines. The procedure also had steps to ensure that the condensate flow out of sample lines were clear. Also, the licensee had replaced the SSF feedwater control valves and added bypass valves to assist in the control of the SSF feedwater flow to the steam generators. The modifications appeared adequate.

(3) In Item 3, "Periodic Safe Shutdown Facility Auxiliary Service Water Pump Operability Test," PT/0/A/0400/05, was not performed under suitable environmental conditions in that the pump was preconditioned in step 12.2 by venting the pump just prior to it being started.

In response the licensee committed to eliminate the preconditioning from the procedures by September 1, 1994. The team verified the procedure had been adequately revised.

In summary, Item 3 is closed. Items 1 and 2 remain open pending completion of corrective action by the licensee.

h. (Closed) IFI 50-269, 270 ,287/93-25-09, "CCW Pump NPSH Information." The initial inspection report noted incomplete documentation of the NPSH requirements and availability for design basis conditions for the CCW pumps for the Keowee Dam failure event.

The licensee provided three letters from the pump manufacturer, one dated June 29, 1967 and two dated July 3, 1967, which had been discovered after the initial inspection when a vendor letter dated February 16, 1968, had been provided. These verified the capability of the pumps to operate at conditions down to lake elevation 770' (the elevation of the top of the impounded pond weir) and 90°F (the design temperature for plant components served by LPSW), and down to 767 at an unspecified temperature, without NPSH problems. This information encompassed the LOOP/LOCA situation. The licensee indicated there would be additional attempts to obtain more definitive information from the vendor on the minimum NPSH requirements.

However, the worst case design basis conditions of available NPSH were for the loss-of-Lake Keowee event which produced the lowest intake level and highest intake temperature. This event begins with the level at the top of the weir and the water at the design maximum temperature. During the event, the level decreases due to evaporation and seepage from the impounded area, and the temperature increases as the CCW water is recirculated to the impounded area carrying the plant decay heat. The analyses of this event will be completed in response to Violation 50-269, 270, 287/93-25-03F. Therefore, the acceptability of the CCW pump NPSH requirements will be reviewed in conjunction with the completed licensee analysis.

Report Details 16

1. (Open) DEV 50-269, 270, 287/93-25-10 (Deviation B in NRC Inspection Report 93-25), "Inadequate HPSW SBO Test." The test did not properly establish initial testing conditions and provided weak guidance when problems were identified. In response the licensee committed to revise the test procedure.

   The test procedure had been revised, and all but one initial issue had been addressed. For the condition of a severely leaking HPSW pump discharge check valve, the updated procedure still directed the operator to isolate the faulty valve and allowed the test to proceed and the results to be accepted, when the test would not have passed with the valve un-isolated. The PIP contained a statement that "The test can continue with the leaking check valve isolated", and a note still existed in the procedure to make an entry in the turnover sheets to inform the Operators to shut the affected pump's discharge valve upon loss of power to prevent excessive losses of the EWST, implying that a leaking check valve was acceptable with regard to the test results. As noted in the original inspection report, this is not acceptable since it is not reasonable to expect that an operator would isolate the valve before excessive losses from the EWST could have occurred in an actual SBO event. Numerous, complex operator actions are necessary to respond to an SBO event. The guidance for performing the isolation is not contained in an emergency procedure but on a rounds checklist.

   This deviation remains open pending appropriate resolution to a leaking HPSW pump discharge check valve when performing the test.

J. (Closed) IFI 50-269, 270, 287/93-25-11, "Jocassee Dam Failure IPE." Inaccuracies: contrary to the IPE submittal the SSF could not withstand the postulated external flood. Also, IPE Submittal report, Section 3, Subsection 13, indicated there was an 8' waterproof flood wall around the SSF ground level entrances. The wall was actually 5' in height.

   The licensee stated a supplemental response to the IPE on external events in 1995 would include re-analysis of risk impact of external flood. Also, further enhancements would be evaluated as a result of this re-analysis.

   The licensee had not completed the analysis of the Jocassee dam failure. Calculation OSC-6781, "USQ Evaluation for Change in FSAR Concerning SSF and Jocassee Flood," provided the justification for changing the FSAR. The change removed the requirements of SSF mitigation for a rapid Jocassee Dam failure. This was based, in part, on PRA data, IPE information, and other information submitted to the Commission. The revised IPE will be reviewed by the NRC as part of the external events IPE submittal. This item is closed.

k. (Open) VIO 50-269, 270, 287/93-25-12 (Violation C in NRC Inspection Report 93-25), "SWS Procedure/Drawing Content or Procedure Implementation Inadequacies." There were five parts associated with this violation.

(1) In item 1, engineering administrative procedures did not establish a definitive length of time for revising calculations following design changes.

In the licensee's original response to the violation the licensee stated that Procedure EDM-101, Engineering Calculations/Analysis, would be revised to provide the necessary guidance by September 1, 1994. However, in a letter dated September 13, 1994, almost two weeks beyond the committed dated, the licensee stated that the plant modification process would be revised by November 1, 1994, to provide the necessary guidance.

The original inspection finding dealt with calculations associated with the SSF SWSs. With regard to these, many of the older calculations have been deleted. Some calculation revisions are on hold while the DBD is being completed. This is scheduled for mid-1995.

The definition of design documents to be updated with a facility change had been modified to include affected design calculations. However, the infrastructure to implement the requirement did not exist. Other than engineer memory, there was no method to identify all other calculations affected by the calculation needing revision. Even if recognized by the engineer, there was no method to flag that a calculation needed revision. Also, a calculation could be revised without using the administrative requirements of the facility change process. Further licensee actions were necessary to reasonably assure administrative requirements were properly implemented. This violation remains open pending such corrective action.

(2) In item 2, no flow instruments existed to confirm 200 gpm was being provided to each steam generator or 400 gpm to an unisolated steam generator by the Auxiliary Service Water pump as directed by Emergency Procedure EP/1,2,3/A/1800/01, Section 502.

In the response to the violation, the licensee stated the emergency procedure would be revised and training on the procedure revision completed by October 1, 1994.

The procedure was revised by the committed date. However, step 7 of the procedure controlled flow via the recirculation valve. Such an action could not actually control flow to all the units or SG pairs.

10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires that activities affecting quality shall

be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances. Failure to provide adequate procedural direction to accomplish this task is violation 50-269, 270, 287/94-31-05, "Inadequate ASW Procedure." However, the licensee had previously identified the error and was in the process of revising the procedure. This violation will not be subject to enforcement action because the licensee's efforts in identifying and correcting the violation meet the criteria specified in Section VII.B of the Enforcement Policy.

(3) In Item 3, Keowee Turbine Generator Cooling Water system drawings, KFD-100A-1.1 and KFD-100A-2.1 did not indicate the existence of an additional valve downstream of valve 2WL-3 for Unit 2, the supply line to the air compressor coolers was interconnected to the 12" main piping for Unit 1, the piping downstream of valve WL-76 was copper for both Units, or a consistent piping class break in the supply line to the generator thrust bearing coolers for both Units.

In the response to the violation, the licensee stated PIP 0-093-0986 was initiated to address these items. Also, the drawings would be re-verified by walkdown with all identified errors to be corrected by July 1, 1994.

The KFD-100A-1.1 and KFD-100A-2.1 drawings had been revised in July 1994. A walkdown of select Keowee mechanical systems with the revised drawings reflected an additional valve downstream of valve 2WL-3 for Unit 2 not shown on the drawing. This was one of the original discrepancies documented in the Notice of Violation.

The team determined through interview with the engineering personnel involved in the resolution of the PIP that only a partial walkdown of the service water mechanical systems was performed through verbal mis-communication. Also, the drawing editorial change process was used to update the drawings which contained a limited check for accuracy of the walkdown activities.

Once the licensee was notified of the drawing discrepancy by the team, the additional valve downstream of valve 2WL-3 was removed. However, the licensee failed to adequately correct the design document error in July 1994 or adequately perform a walkdown of the mechanical systems. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires conditions adverse to quality be promptly identified and corrected. This is considered an example of Violation 50-269, 270, 287/94-13-01B, "Inadequate Corrective Action Controls."

(4) In Item 4, a condition adverse to quality report dealing with the removal of the Keowee Unit 2's turbine guide bearing oil cooler was neither properly processed nor did it receive a written operability evaluation.

In response to the violation, the licensee stated an operability evaluation would be completed on the ramifications of removing the oil cooler from service without declaring the unit inoperable by November 1, 1994. Also, involved personnel were retrained on the condition adverse to quality procedure.

The involved personnel were retrained. Also, on November 1, 1994 the licensee satisfactorily completed an operability determination in calculation KC Unit 1-2-0107, "Keowee Turbine Guide Bearing Temperature Calculation," with results indicating the oil cooler was not necessary for unit operability. This violation example is closed.

(5) In Item 5, the appropriate housekeeping zones were not being assigned to select maintenance activities at Keowee.

In response to the violation, the licensee stated that all applicable maintenance procedures would be revised by October 1, 1994. Also, training on housekeeping zone requirements would be given to Keowee personnel by August 1, 1994.

The applicable procedures were revised, and training was provided to the Keowee personnel. However, the training was given 50 days after the committed date. Keowee personnel were aware the date assigned was not met and had contacted regulatory assurance personnel. The regulatory assurance personnel involved did not understand NRC concurrence would be necessary to extend the commitment date and stated that the missed commitment date would be updated in quarterly correspondence with the NRC on electrical inspection issues.

In summary, Items 2, 3, 4, and 5 are closed. Item 1 remains open pending completion of additional licensee corrective actions.

1. (Closed) IFI 50-269, 270, 287/93-25-14, "Review of Revised ASW Pump NPSH Calculation."

During the original inspection calculation OSC-5125, "ASW NPSH Analysis," assumed siphon flow from the intake canal to the ASW pump suction would be in operation following the tornado. However, the ECCW siphon lacked tornado protection, and would not be operational. Therefore, the minimum suction height was contingent upon the inventory losses in the CCW piping as a result of ASW pump operation. Minimum NPSH for the ASW pump was -2.22 psig which meant that the pump could draw water from 5.12 feet below the pump's impeller eye and still have adequate NPSH. However, the licensee failed to consider the actual configuration of the CCW piping going to the suction of the ASW pump. Therefore, when the water in the CCW piping dropped to a height of 770.46 feet, inadequate NPSH would occur. Consequently, the amount of water available for ASW pump use was noticeably reduced.

The licensee failed to correct the calculation and failed to identify that the error in the calculation constituted a condition adverse to quality and required the initiation of a PIP. During the follow-up inspection this failure to initiate a PIP was identified to the licensee. PIP 94-1500 was subsequently initiated which also identified OSC-0004, "RC System DII Removal following a Loss of Intake Structure," as being deficient. The licensee's site directive on the problem identification process requires the initiation of a condition adverse to quality report (PIP) when there are errors in design bases documents and when documents are not updated. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires in part that conditions adverse to quality be promptly identified. Failure to originally identify this condition adverse to quality by initiating a PIP was an example of Violation 50-269, 270, 287/94-13-01C, "Inadequate Corrective Action Controls."

m. (Open) IFI 50-269, 270, 287/93-25-15, Administrative Controls for Lake Keowee: Calculation OSC-3520, "Keowee Lake Level Minimum Administrative Limits," had numerous technical weaknesses. The calculation was an attempt to establish a minimum lake level necessary to ensue operability of Oconee and Keowee for design basis events.

No further actions had been accomplished since the original inspection. The licensee indicated that this matter had "dropped through the cracks" following a reorganization and changes in personnel within the engineering department. The licensee indicated an individual would be assigned to address this matter. This issue remains open pending licensee action.

4. Inspection Report 93-25 Cover Letter Responses

The SWSOPI cover letter dated February 11, 1994, requested a written response describing analysis, rational or actions planned regarding:

- The presence of only one valve isolating the safety-related portion of the LPSW system from the nonsafety-related turbine building portion.
- The SSF could not withstand a postulated failure of the Jocassee Dam.
- The HPSW system was not designed or maintained commensurate with its importance to safety.

Also, the cover letter requested the licensee's design control measures, engineering evaluations, testing program, and the safety classification of components system be evaluated to determine any necessary programmatic corrective actions warranted by this inspection report.

In letters dated March 12, 1994, and April 20, 1994, the licensee discussed these issues defined in the cover letter to the SWSOPI report. The team reviewed the licensee responses and ascertained the status of the corrective actions identified within the body of the letters. The results of the reviews were as follows:

a. HPSW Single Isolation - This matter is closed as discussed in paragraph 3.b of this inspection report.

b. Ramifications of Jocassee Dam Failure on the SSF - This matter is closed as discussed in paragraph 3.j of this inspection report.

c. Design and Maintenance of the HPSW System -

(1) The initial inspection report discussed that covers for sections of the trench in which the power and control cables for the CCW pumps run, as well as the HPSW sealing/cooling water supply line to the pumps, were not bolted down and could fall into the trench in a seismic event, potentially damaging these components. It was also reported that the cover for the structure housing the ECCW valve CCW-8 was not restrained and could potentially fall on the valve in a seismic event.

The licensee generated PIP O-094-504 to perform a minor modification to provide a seismic design for the trench covers in question. However, this PIP was closed out on June 29, 1994, when the design package was completed, but the work still had not been started at the time of the follow-up inspection. When this was discovered by the team, the licensee indicated the PIP would be reopened to ensure completion of the installation work.

The team also found that no actions had been performed on the structure cover for valve CCW-8, and inspection revealed that one of the three cover plates was out of position and resting on one of the other plates, making it particularly vulnerable to a seismic event. The initial licensee response was that since functions for which this valve were required did not have to be considered in conjunction with a seismic event, it did not have to be seismically qualified. The team pointed out to the licensee that the requirement for safety-related equipment to be seismically qualified is solely because it is safety-related; it is not dependent on there being a seismic event linked to the event for which it is required.

The licensee indicated the cover would be restored to its proper location and signs installed on these covers and the trench covers indicating they must remain in place for the seismic qualification of the equipment to remain valid.

(2) The initial inspection report discussed that the operator rounds sheets contained no upper limits on the HPSW flows to the CCW pumps and motors for sealing and cooling respectively, and that 17 of the 24 flow instruments for these pumps were above the values used to calculate the 4-hour, SBO capacity of the EWST.

The licensee's response in the March 14, 1994, letter was that if the EWST SBO test passed, it was irrelevant if the flows were above the values used in the EWST capacity calculation. This is

not correct. Without upper limits on the flow, even with satisfactory test results, during the one year interval to the next test, the flow could be significantly increased, without constraint or indication, potentially rendering the test results invalid.

During the follow-up inspection, the licensee acknowledged that upper limits on the flows were appropriate and indicated necessary actions would be to taken.

(3) The initial inspection report discussed that HPSW performs the function of providing sealing and cooling water to the CCW pumps and motors respectively. These flows are set and monitored using rotameters at the CCW pumps. The initial inspection found that these instruments were being improperly set and poorly maintained.

In the licensee's letter response of March 14, 1994, it was stated that if the flow were actually inadequate, it would adversely affect pump performance, implying that this reduced performance would be detected during normal operation. This is not correct. Normal pump performance is not at all affected by loss of seal water. This loss only affects its safety-related function of maintaining the "first siphon" integrity when the pumps are not running.

(4) The initial inspection report discussed the poor material condition of the CCW rotameters.

The licensee responded in the March 14, 1994, letter that a PIP had been written to document the off-scale high reading of a number of rotameters. As corrective action to this PIP the licensee initiated a preventative maintenance activity to periodically clean the rotameters.

The team reinspected these devices during the follow-up inspection and found essentially the same conditions as the initial inspection. Ten of the twenty-four instruments contained significant slime contamination, and one contained three clams, making the accuracy of their indication questionable. Eleven of the flow instruments were pegged at the high end of the scale. Therefore, this parameter was not monitorable with these instruments.

Since readings from these instruments were taken as a part of the normal operator rounds, the team reviewed the non-licensed operator turnover sheets, OP/2/A/1102/20, Enclosure 5.7, and rounds sheets, Enclosure 5.11, dated October 9, 1994, and October 10, 1994, to determine if any notation had been made of their material condition. No notation had been made.

Operations supervisors were questioned on the training of operators regarding recognition of discrepant conditions and how conditions, such as were found, could exist. Operations personnel indicated that as long as flow could be determined or pegged high, since engineering had not established an upper limit, the condition would not be identified.

The licensee had initiated a PIP and took the corrective action identified. However, the actions were ineffective. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires in part that conditions adverse to quality be promptly corrected. The failure of the PIP corrective actions to adequately resolve the poor material condition of the CCW rotameters or even ensure that the material condition discrepancies were identified is an example of Violation 50-269, 270, 287/94-31-01D, "Inadequate Corrective Action Controls."

(5) The initial inspection report identified that the HPSW pump check valves, HPSW-2, 5, and 8 were not being properly tested. Also, the these valves and the pumps had not been properly classified.

The licensee responded in the March 14, 1994, letter that a PIP had been written to add these valves to the IST program. Also, a PIP was written to update the QSM and DBD for the HPSW system.

The team found that the QSM and IST corrective actions were predicated upon revision of the DBD. The DBD revision was still in draft form with the corrective action completion schedule extended to the end of the year.

(6) The initial inspection report discussed the structural capability of the HPSW system.

The licensee responded in the March 14, 1994, letter that inspected piping met minimum wall thickness criteria, the hanger design was acceptable, and appropriate consideration had been enacted with a revision to the SLC 16.9.7 to account for spurious HPSW actuation. The response also stated, "Oconee recognizes that the HPSW System was not required to be designed and constructed to the seismic design criteria presented in Section 3.7 and 3.9 of the Oconee FSAR." The team confirmed the SLC had bee revised. However, the overall design and construction requirements of the HPSW system continues to be reviewed by the NRC as part of URI 269, 270, 287/93-13-03, "ECCW System Design and Testing."

d. Design Control/Engineering Evaluations - The licensee indicated EDM-101, "Engineering Calculations/Analyses," would be revised to clarify management expectations on the verification and validation of design inputs and assumptions. Also, all required functions identified in

the mechanical systems DBDs would be reviewed against both the existing calculational support for that function and the existing emergency operating procedures and the abnormal procedures.

The team verified the revision to EDM-101 and determined the status of the mechanical systems DBD review. The review was in its initial stages.

Also, the team selected another subsystem of the SSF to determine whether there were additional problems within the design control area. A review of the SSF RC make-up calculations revealed deficiencies. Some of the calculational inconsistencies that were similar to those present in the SSF calculations reviewed during the previous inspection. The licensee indicated that many of these would be corrected when the DBD was issued. The DBD was scheduled to be issued in November 1994.

The most significant deficiency was in OSC-619, "Analysis for the use of Spent Fuel Pool Inventory for Standby Shutdown Facility." The deficiency was the failure to fully account for the high radiation levels of the Spent Fuel Pool in a timely manner when the water level dropped to one foot above the top of the spent fuel. The licensee's preliminary analysis indicated the radiation levels would be approximately 2,000,000 Roentgens per hour at the pool surface at 72 hours following loss of fuel pool cooling. There were no time-specific requirements for restoration of fuel pool make up. Additionally, there were no specific procedural steps for performing this evolution nor was equipment staged for this repair. This is Unresolved Item 50-269, 270, 287/94-31-06, "High Spent Fuel Pool Radiation Levels."

e. Testing - The licensee indicated that significant progress was being made in developing and implementing integrated system testing. Also, all required functions identified in the DBDs were being reviewed against existing test procedures to ensure the function was being appropriately verified.

The team ascertained the DBD/testing review was in progress.

f. Safety Classification - The licensee indicated that further review as to whether the QSM was complete was ongoing. For those cases maintenance and testing procedures would be upgraded consistent with the Quality Assurance program.

The team recognized the licensee's effort but additional evaluation of Generic Letter 83-28 and a review of the licensee's response to that GL would be necessary to fully understand the adequacy of the licensee's corrective actions in this area.

Completion of these programmatic reviews in the areas of Testing/QSM/Design Control is an Inspector Follow-up Item 50-269, 270, 287/94-31-07, "Quality Programs Review."

5. Review of Valve LPSW 139 to Close

The licensee had recalculated the closing torque requirements using the EPRI methodology developed for butterfly valves. These calculations indicated that the valve operators had sufficient torque to close and seat the valves. This result differed from the calculation provided with the licensee's response to IR 93-04 in that there was no assurance that sufficient torque to seat the valves. The licensee had conducted laboratory testing for valve closure against high differential pressure for the new LPSW-139 valve, but no high differential pressure testing has been conducted on the valve currently installed. As documented in the licensee's response to IR 93-04, the licensee plans to install the new LPSW-139 valve in parallel with the current valve during the next Unit 1 outage currently scheduled for October 1995.

6. IST Program Scope versus Appendix B Testing Program Scope

During the initial SWS inspection certain aspects of the licensee's ASW system testing program as it related to the IST requirements arose. Specifically, the licensee used a combination of IST and an Appendix B test program to encompass the testing of the safety-related valves. As a result of the discussions with the licensee regarding the scope of their testing program, the team expanded the scope of review to include all valves included in the Appendix B program which were not included in the IST program. The results of that review were as follows:

a. The scope of the licensee's Section XI test program only encompassed that equipment used to mitigate accidents discussed in chapter 15 of the FSAR. Due to Oconee's unique design, other equipment not credited in the chapter 15 accidents is used to mitigate accidents discussed in other chapters of the FSAR such as a tornado. Therefore, there were omissions in the ASME Section XI test program.

b. The ASW check valves (LPSW-502) allowing flow to the HPI pump motor coolers had not been tested. HPI pump motor cooling via the ASW system was assumed to mitigate the consequences of certain types of tornadoes and to mitigate the consequences of a complete loss of the intake canal water. The licensee had previously identified the testing omission and initiated a PIP. Plans and preparations were being made to accomplish the testing before the end of the year. This situation was similar to NCV 50/269, 270, 287/93-26-13, "Omissions of LPSW Check Valves from IST Program."

c. Other valves necessary to function to mitigate the consequences of a tornado were not included within the IST program. Most notable were the atmospheric dump valves (MS 153-156 & 161-164) and the condenser water box isolation valves (CCW 21-25). These valves were tested during shutdown conditions but not at the frequency specified in Section XI of the ASME code. No relief request had been submitted to the NRC concerning these valves.

7 Review of High Energy Break Ramifications within Containment and Planned Corrective Actions

Report Details

d. Numerous valves associated with the air start and fuel supply systems to the SSF DGs were not identified as part of the IST program. However, the valves were being tested at the required frequency.

f. The manual MTOTC bypass valve, LPSW-51, was not identified as requiring testing, ASME Section XI nor Appendix B. The licensee's analysis credited its closure (when the MTOTC valve was out of service) during a LOCA/LOOP. When identified to the licensee PIP 94-438 was initiated.

10 CFR 50.55(a) requires safety related values be included within the Section XI valve test program. Failure to include the ASW check valves, atmospheric dump valves, and the MTOTC bypass valve within the scope of the ASME code Section XI test program is Violation 50-269, 50-270, 50-87/94-31-08, "ASME Section XI Test Program Omissions."

7. Review of High Energy Break Ramifications within Containment and Planned Corrective Actions

On June 2, 1993, the licensee determined that the original MSLB within containment analysis was nonconservative and LER 50-269/93-06 was submitted on the subject. In a letter dated May 27, 1993, the licensee discussed the rationale as to why there should be continued reliance upon the control grade integrated control system and operator action prior to implementation of a long term solution. In a letter dated August 19, 1993, the licensee also discussed the long term solution, an automatic feedwater line isolation scheme, to be implemented in the 1995/96 timeframe. In a letter dated October 6, 1993, NRC indicated this approach was viable.

The worst scenario discussed in the licensee's correspondence involved the MSLB with the accompanied failure of the feedwater control valve to close. The analysis concluded that containment design pressure would be exceeded (but structural yield pressure would not be reached) if manual action to close the feedwater block valve was taken within two minutes.

The team reviewed the containment temperature and pressure curves for this case of "Without Credit for Automatic Main Feedwater Control and Main Feedwater Control Valve Sticks Open". This case indicated that the EQ of equipment (mostly instrumentation) used to mitigate a MSLB was exceeded from a temperature perspective. The temperature EQ would be exceeded in the short term and long term.

This observation appeared to contradict other licensee statements in their correspondence of May 27th and August 19th which stated that the equipment required to mitigate the consequences of the MLSB was qualified and would perform its safety function. Discussions with the licensee indicated that the EQ statement was for the equipment after implementation of the long term solution for MSLB or in the design bases LOCA, and the equipment had not been EQ reviewed for the pressures and temperatures in the MSLB with a control valve failure case. Since this information was not clearly stated in the licensee's correspondence to the NRC (letters dated May 27, 1993

and August 19, 1993), the team notified the NRR Project Manager responsible for review of the adequacy of the licensee's response. Subsequently the Project Manager informed the team that the new information did not change the original NRC decision on this matter.

The team reviewed the current status of the proposed long term solution. The licensee stated that the modification would close non safety related valves and that the equipment would be EQ qualified for MSLB's. The schedule for implementation of the modification was feasible considering the fact that advantage will be taken of already installed safety-related equipment. The licensee indicated that some of the long term solution discussed in their letter of August 19, 1993, was going to be modified. A discussion with the NRR Project Manager indicated that the NRC understands that the licensee intends to submit surveillance and technical specifications that would ensure the operability of the modification.

8. Exit Interview

The team conducted an exit meeting on October 27, 1994, at the Oconee Nuclear Power Station to discuss the major areas reviewed during the inspection, the strengths and weaknesses observed, and the inspection results. Licensee representatives and NRC personnel attending this exit meeting are documented in Appendix A of this report. The team also discussed the likely informational content of the inspection report. The licensee did not identify any documents or processes as proprietary. There were three dissenting comments at the exit meeting associated with the lack of gravity/siphon flow testing, the scope of the ASME code Section XI valve test program and the failure to initiate a PIP when an inaccurate reference was used when performing calculation OSC-2346, "ECCW System Performance Evaluation." The licensee indicated a thorough review of the inspection findings would be necessary to ascertain the appropriate responses or corrective actions to the issues identified.

Also, an exit was held with the cognizant SSF engineer associated with the team's attempt to witness SSF SWS testing. There were no findings since the test could not be performed due to unanticipated scheduling delays in establishing the plant conditions for the testing.

ITEM NUMBER	STATUS	PARAGRAPH	DESCRIPTION
93-25-01	Open	3.a	DEV - Failure to Adequately Perform SWS QL Actions
93-25-02	Closed	3.b	UNR - Turbine Building Isolation Single Failure Vulnerabilities
93-25-03	Open	3.c	VIO - Failure to Perform Adequate Calculations and Evaluations to Support Facility Design
93-25-04	Closed	3.d	VIO - Inadequate Evaluation of Conditions Adverse to Quality by Engineering
93-25-05	Open	3.e	IFI - Additional Validation of RBCU Evaluation Inputs
93-25-06	Open	3.f	IFI - Actions to Improve Operator Responses to Abnormal Events
93-25-08	Open	3.g	VIO - Inadequate SSF and ECCW Testing
93-25-09	Closed	3.h	IFI - CCW Pump NPSH Information
93-25-10	Open	3.i	DEV - Inadequate HPSW SBO Test
93-25-11	Closed	3.j	IFI - Jocassee Dam Failure IPE Inaccuracies
93-25-12	Open	3.k	VIO - SWS Procedure/Drawing Content or Procedure Implementation Inadequacies
93-25-14	Closed	3.l	IFI - Review of Revised ASW Pump NPSH Calculation
93-25-15	Open	3.m	IFI - Administrative Controls for Lake Keowee
94-31-01	Open	3.a, 3.k(3) 3.l 4.c(4)	VIO - Inadequate Corrective Action Controls
94-31-02	Open	3.c(2)	IFI - Hydraulic Model Controls Transition
94-31-03	Open	3.c(4)	IFI - Reperformance of Calculation

OSC-2346

94-31-04	Open	3.c(4)	VIO - Inadequate LPSW Suction Source Testing Via the ECCW System
94-31-05	Closed	3.k(1)	NCV - Inadequate ASW Procedure
94-31-06	Open	4.d	UNR - High Spent Fuel Pool Radiation Levels
94-31-07	Open	4	IFI - Quality Programs Review
94-31-08	Open	8	VIO - ASME Section XI Test Program Omissions

Report Details 20

94-31-01	Open	3.a	NCV - Failure to Adequately Perform QA/QC Actions

LIST OF PERSONS CONTACTED

Duke Nuclear Power Plant

Persons Contacted

* L. Azzarello, Mechanical Engineering
* S. Baldwin, Mechanical Systems Engineering
* D. Coyle, Mechanical Systems Manager
* B. Dolan, Safety Assurance Manager
* W. Foster, Maintenance
* R. Harris, Mechanical Systems Engineering
* W. Horton, Operations Support
* D. Hubbard, Maintenance
$ H. Lefkowitz, Mechanical Systems Engineering
* G. McAninch, Mechanical Systems Engineering
* B. Millsaps, Mechanical/Civil Engineering
*$ J. Smith, Regulatory Compliance
* R. Swigant, Work Control

U.S. Nuclear Regulatory Commission

*$ L. Mellen, Reactor Inspector
* D. Prevatte, Powerdyne Corporation
* C. Rapp, Reactor Inspector
* W. Rogers, Team Leader
* L. King, Reactor Inspector
* P. Harmon, Senior Resident Inspector
* L. Keller, Resident Inspector

* Indicates those present at the exit meeting on October 27, 1994
$ Indicates those involved in the exit on November 3, 1994

LIST OF ACRONYMS AND ABBREVIATIONS

ASME	•	American Society of Mechanical Engineers
ASW	•	Auxiliary Service Water
CCW	•	Condenser Cooling Water
CFR	•	Code of Federal Regulations
DBD	•	Design Basis Document
DEV	•	Deviation
DG	•	Diesel Generator
DH	•	Decay Heat
ECCW	•	Emergency Condenser Cooling Water
EFW	•	Emergency Feedwater
EPRI	•	Electric Power Research Institute
EQ	•	Environmental Qualification
EWST	•	Elevated Water Storage Tank
FSAR	•	Final Safety Analysis Report
GL	•	Generic Letter
GPM	•	Gallons Per Minute
HPI	•	High Pressure Injection
HPSW	•	High Pressure Service Water
HVAC	•	Heating Ventilation and Air Conditioning
IFI	•	Inspector Follow-up Item
IPE	•	Individual Plant Examination
IST	•	Inservice Test
LOCA	•	Loss of Coolant Accident
LOOP	•	Loss of Offsite Power
LPI	•	Low Pressure Injection
LPSW	•	Low Pressure Service Water
MSLB	•	Main Steam Line Break
MTOTC	•	Main Turbine Oil Temperature Control
NCV	•	Non-Cited Violation
NPSH	•	Net Positive Suction Head
NPSHA	•	Net Positive Suction Head Available
PIP	•	Problem Investigation Process
PRA	•	Probabilistic Risk Analysis
PSIG	•	Pounds per Square Inch Gauge
PTRQ	•	Personnel Training Requiremental Qualification
QA	•	Quality Assurance
QSM	•	Quality Standards Manual
RBCU	•	Reactor Building Cooling Unit
RC	•	Reactor Coolant
SBO	•	Station Blackout
SER	•	Safety Evaluation Report
SG	•	Steam Generator
SLC	•	Selected Licensee Commitments
SQUG	•	Seismic Qualification Utility Group
SSF	•	Safe Shut Down Facility
SWS	•	Service Water System
SWSOPI	•	Service Water System Operational Performance Inspection
UNR	•	Unresolved Item
USQ	•	Unreviewed Safety Question
VIO	•	Violation