July 29, 1997

EA 97-297 and EA 97-298

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

SUBJECT: PREDECISIONAL ENFORCEMENT CONFERENCE SUMMARY - OCONEE NUCLEAR STATION (NRC INSPECTION REPORT NOS. 50-269, 270, 287/97-07 AND 50-269, 270, 287/97-08)

Gentlemen:

This letter refers to the predecisional enforcement conference held at our request on July 23, 1997, at the Region II office in Atlanta, Georgia. The purpose of the meeting, which was open to the public, was to discuss the apparent violations associated with the April 21, 1997, Unit 2 High Pressure Injection (HPI) line/nozzle weld crack event and the Unit 3 HPI system degradation on May 3, 1997. We considered the meeting beneficial, and it provided us a better understanding of several of the issues discussed in the subject inspection reports.

Enclosed is a list of the Conference Attendees, the NRC's Conference Agenda and Apparent Violations, and Oconee Nuclear Station handouts. The results of the NRC's deliberations regarding the apparent violations, discussed at the predecisional enforcement conference, will be forwarded to you by separate correspondence.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this letter, please contact us.

Sincerely,

ORIGINAL SIGNED BY
JOHNS P. JAUDON

Johns P. Jaudon, Director
Division of Reactor Safety

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

Enclosures: (See page 2)

IE45 /1

Enclosures:
1. List of Attendees
2. Predecisional Enforcement Conference Agenda
3. Apparent Violations
4. Oconee Nuclear Station Handouts

cc w/encls:
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D. LaBarge, NRR
R. Carroll, RII
C. Payne, RII
PUBLIC

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COPY	YES [✓] NO [ ]	YES [✓] NO [ ]	YES [ ] NO [ ]	YES [ ] NO [ ]	YES [ ] NO [ ]

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List of Attendees

Nuclear Regulatory Commission

L. Reyes, Regional Administrator, Region II (RII)
J. Lieberman, Director, Office of Enforcement (OE)
B. Mallett, Acting Deputy Regional Administrator, RII
J. Jaudon, Director, Division of Reactor Safety (DRS), RII
J. Johnson, Director, Division of Reactor Projects (DRP), RII
H. Berkow, Director, Project Directorate (PD) II-2, Office of Nuclear Reactor Regulation (NRR)
L. Plisco, Deputy Director, DRP, RII
B. Uryc, Director, Enforcement and Investigation Coordination Staff (EICS), RII
C. Evans, Regional Counsel, RII
N. Dudley, Regional Coordinator, Office of Executive Director for Operations
W. Holland, Chief, Maintenance Branch, DRS, RII
S. Shaeffer, Acting Chief, Projects Branch 1, DRP, RII
C. Ogle, Vogtle Senior Resident Inspector, DRP, RII
T. Reis, Enforcement Specialist, OE
D. LaBarge, Senior Project Manager, PD II-2, NRR
R. Bernhard, Senior Reactor Analyst, DRS, RII
M. Scott, Oconee Senior Resident Inspector, DRP, RII
E. Girard, Senior Reactor Inspector, DRS, RII
A. Boland, Enforcement Specialist, EICS, RII
R. Carroll, Jr., Project Engineer, Branch 1, DRP, RII
P. Harmon, Senior Operations Engineer, DRS, RII
M. Ernstes, Project Engineer, DRP, RII
D. Billings, Oconee Resident Inspector, DRP, RII

Duke Power Company

M. Tuckman, Executive Vice President
W. McCollum, Jr., Site Vice President, Oconee Nuclear Station (ONS)
M. Nazar, Engineering Manager, ONS
J. Davis, Station Support Division Manager
B. Peele, Station Manager, ONS
T. Curtis, Superintendent of Operations, ONS
B. Foster, Safety Assurance Manager, ONS
J. Burchfield, Regulatory Compliance Manager, ONS
R. Zuercher, Media Relations

Enclosure 1

PREDECISIONAL ENFORCEMENT CONFERENCE AGENDA

OCONEE NUCLEAR STATION

JULY 23, 1997, AT 10:00 A.M.

NRC REGION II OFFICE - ATLANTA, GEORGIA

I. OPENING REMARKS AND INTRODUCTIONS
L. Reyes, Regional Administrator

II. NRC ENFORCEMENT POLICY
B. Uryc, Director
Enforcement and Investigation Coordination Staff

III. SUMMARY OF THE ISSUES
L. Reyes, Regional Administrator

J. Lieberman, Director
Office of Enforcement

IV. STATEMENT OF CONCERNS / APPARENT VIOLATIONS
J. Jaudon, Director, Division of Reactor Safety

V. LICENSEE PRESENTATION

VI. BREAK / NRC CAUCUS

VII. NRC FOLLOWUP QUESTIONS

VIII. CLOSING REMARKS
L. Reyes, Regional Administrator

ENCLOSURE 2

NRC OPEN PREDECISIONAL ENFORCEMENT CONFERENCE

OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

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

UNDETECTED CRACKS IN HIGH PRESSURE INJECTION PIPING RESULTED IN THROUGH WALL CRACK ON OCONEE - UNIT 2

AND

DEGRADATION OF THE OCONEE - UNIT 3 HIGH PRESSURE INJECTION SYSTEM DURING A UNIT COOLDOWN

JULY 23, 1997

ENCLOSURE 3

APPARENT VIOLATIONS

A. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures 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.

As of April 21, 1997, the licensee failed to establish measures to assure that conditions adverse to quality were promptly identified and corrected. Specifically, the licensee failed to effectively implement the periodic augmented inspection program for high pressure injection (HPI) system piping intended to detect cracks or their precursors, as prescribed by the 1982 "Babcock & Wilcox Owner's G177 Fuel Assembly Owner's Group Safe End Task Force Report on Generic Investigation of HPI/MU Nozzle Component Cracking" and endorsed by the licensee in a February 15, 1983, letter to the NRC. Due to (1) the lack of definitive acceptance criteria for certain radiographic testing (RT) conducted in 1996, (2) the failure to conduct ultrasonic testing (UT) of susceptible piping areas, (3) the failure to develop RT procedural requirements to assure the quality of RT performed to detect sleeve/safe end gap, and (4) the failure to properly record indications found during UT conditions, indications of cracking went unrecognized until a crack resulted in an unisolable reactor coolant leak on April 21, 1997.

B. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that measures 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.

The licensee failed to assure that conditions adverse to quality were promptly identified and corrected in that the licensee measured temperature differentials indicative of thermal stratification, a potential cause of pipe cracking, in their HPI lines in June 1990; however, no subsequent actions were initiated to identify, evaluate, correct, or assess the impact of the temperature differentials on the HPI augmented inspection program as of April 1997.

NOTE: THE APPARENT VIOLATIONS DISCUSSED IN THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND ARE SUBJECT TO CHANGE PRIOR TO ANY RESULTING ENFORCEMENT DECISION.

APPARENT VIOLATIONS

C. Technical Specification (TS) 3.2.1, "High Pressure Injection and Chemical Addition Systems," requires that the reactor shall not be critical unless two high pressure injection pumps per unit are operable except as specified in TS 3.3.

TS 3.3.1.a(1), "High Pressure Injection System," requires that when the reactor coolant system (RCS), with fuel in the core, is in a condition with temperature above 350 degrees Fahrenheit (°F) and reactor power less than 60 percent full power, two independent trains, each comprised of an HPI pump and a flow path capable of taking suction from the borated water storage tank and discharging into the reactor coolant system automatically upon Engineered Safeguards Protective System actuation, shall be operable. Specification 3.3.1.c(1) further requires, in part, that when reactor power is greater than 60 percent full power that the remaining HPI pump shall be operable.

From March 6 until May 2, 1997, with fuel in the Oconee Unit 3 core and RCS temperature greater than 350°F, the licensee failed to maintain the HPI system operable, as required by Technical Specifications. Specifically, the licensee operated with the HPI system outside of the letdown storage tank level versus pressure analyzed limitation curve which resulted in all of the high pressure injection pumps being inoperable due to inadequate net positive suction head.

NOTE: THE APPARENT VIOLATIONS DISCUSSED IN THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND ARE SUBJECT TO CHANGE PRIOR TO ANY RESULTING ENFORCEMENT DECISION.

APPARENT VIOLATIONS

D. TS 6.4, "Station Operating Procedures," requires, in part, that the station be operated and maintained in accordance with approved procedures. Included in this specification is the requirement that procedures be provided for normal startup, operation, and shutdown of the complete facility and of all systems and components involving nuclear safety of the facility.

The licensee failed to operate the station in accordance with approved procedures as evidenced by the following:

1. Licensee Operation Management Procedure (OMP) 2-1, "Duties and Responsibilities of On Shift Operations Personnel," Revision (Rev.) 40, Enclosure 4.5, "Responsibilities of the Reactor Operators," describes the responsibilities of the Operator at the Controls (RO) and the Balance of Plant Operator (RO). Step 2 of the section on shared responsibilities states: "The Reactor Operators assigned to any Control Room are charged with the responsibility of operating their assigned unit. They are to operate the plant with a questioning attitude, keeping nuclear safety and 'Operations Conservatism' in mind." Step 5 of this section further states: "In addition to normal plant monitoring, Reactor Operators are responsible for making <u>at least</u> three complete control room rounds per shift, as defined in Enclosure 4.8." Enclosure 4.8, Step 1 requires that each control room operator make a complete, detailed board walkdown soon after relieving to verify turnover items and to ensure their understanding of plant and control room equipment. Step 9 of the shared responsibilities delineated in Enclosure 4.5 states: "All Reactor Operators shall ensure that his/her normal or selected instruments monitoring their associated parameters are responding as expected for the existing condition. If an instrument is responding contrary to what would be expected, the redundant instrument should be checked to verify the indication."

   During the period of time between 7:45 a.m. and 9:12 a.m., on May 3, 1997, the reactor operators (RO) failed to ensure that letdown storage tank level (LDST) indication was responding as expected for the reactor cooldown. During a cooldown, with the pressurizer being maintained at a constant level, the LDST level is expected to be constantly decreasing, as was demonstrated during the previous shift, when the operations crew was having to repeatedly add water to the LDST.

NOTE: THE APPARENT VIOLATIONS DISCUSSED IN THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND ARE SUBJECT TO CHANGE PRIOR TO ANY RESULTING ENFORCEMENT DECISION.

D. CONTINUED

2. OMP 2-1, Rev. 40, Enclosure 4.5, Step 8 of the section on shared responsibilities states: "A Reactor Operator shall acknowledge all alarms. When an alarm is received, he/she shall take appropriate actions in response to the alarm. This action may include a comparison/check of relevant supporting parameters to validate the alarm, [and] taking such actions as designated in the Alarm Response Guide, Emergency or Abnormal Procedures. When an alarm is received that is unexpected for the existing plant conditions or without apparent cause, he/she shall notify the Control Room SRO immediately."

   During the loss of makeup event on May 3, 1997, licensed operators failed to promptly implement any Annunciator Response Guide, Emergency, or Abnormal Procedures, following receipt of an alarm. Specifically, at 9:13 a.m. on May 3, 1997, alarm 2SA-2/C-2, "HPIP DISCH PRESSURE LOW," at 9:13 a.m. was received in the control room; however, Abnormal Procedure AP/1700/14, "Loss of Normal HPI Makeup and Letdown, Rev. 1," was not entered until 9:32 a.m.

3. OMP 2-1, Rev. 40, Enclosure 4.5, Step 3 of the section on the responsibilities of the Operator at the Controls (OATC) states: "Under the direction of the Control Room SRO, the OATC shall have the responsibility for the operation of the assigned unit. Step 4 of this section further states, in part: "The OATC shall provide surveillance of operations and instrumentation monitored from the Control Room to ensure the safe operation of the Unit."

   Licensee Operations Procedure OP/3/A/1104/49, "Low Temperature Overpressure Protection (LTOP)," Rev. 6, Step 2.8 requires, in part, that a dedicated LTOP operator be assigned whenever RCS temperature is less than or equal to 325°F, the RCS is closed (no LTOP vent path is established), an HPI pump is operating and capable of injecting into the Reactor Coolant System via 3HP-120 (Reactor Coolant Volume Control), and the 3HP-120 travel stop is inoperable. OP/3/A/1104/49, Enclosure 4.3, "Dedicated LTOP Operator Guidelines," Step 1.3, states: "Prevention of low temperature overpressurization is the <u>only</u> responsibility and duty of the dedicated low temperature overpressure protection operator."

   At 11:58 pm, on May 2, 1997, LTOP operation was established with the Operator at the Controls as the designated dedicated operator. This resulted in the dedicated operator having other responsibilities in addition to his responsibility to prevent low temperature overpressurization.

NOTE: THE APPARENT VIOLATIONS DISCUSSED IN THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND ARE SUBJECT TO CHANGE PRIOR TO ANY RESULTING ENFORCEMENT DECISION.

APPARENT VIOLATIONS

E. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires that measures 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 shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition.

The licensee failed to take corrective action for significant conditions adverse to quality to preclude repetition as evidenced by the following:

1. As of May 3, 1997, maintenance procedures used for the installation of tube fittings which had been revised based on previous fitting failures and 1991 industry guidance did not prevent the fitting failure which resulted in draining of the letdown storage tank level instrumentation reference line following its calibration on February 22, 1997.

2. As of May 3, 1997, the licensee failed to implement actions to address design vulnerabilities and operational concerns identified in the high pressure injection system subsequent to a November 14, 1979, loss of HPI suction pressure event while at cold shutdown, as a consequence the deficiencies went uncorrected, contributing to the May 3, 1997, HPI degradation event.

NOTE: THE APPARENT VIOLATIONS DISCUSSED IN THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND ARE SUBJECT TO CHANGE PRIOR TO ANY RESULTING ENFORCEMENT DECISION.

APPARENT VIOLATIONS

F. 10 CFR 50, Appendix B, Criterion III, Design Control, requires that measures be established to assure that applicable regulatory requirements and the design basis for systems, structures, and components which affect the safety-related functions of those systems that prevent or mitigate the consequences of postulated accidents, are correctly translated into specifications, drawings, procedures, and instructions.

Oconee Nuclear Station Specification Number OSS-0254.00-00-1001, "HPI & Purification & Deborating Demineralizers Systems," Revision 5, Section 33.2.1.7 discussed the design and accident aspects of the Unit 3 letdown storage tank, including that the "Letdown Storage Tank Pressure versus Indicated Level" curve is included in plant procedures to define letdown storage tank operating parameters.

As of May 3, 1997, the licensee failed to accurately translate the design basis for the letdown storage tank instrumentation and the associated valves into plant drawings, procedures, and instructions. Specifically the design basis for the letdown storage tank instrumentation did not incorporate design specification requirements for components which affect the safety-related functions of the high pressure injection system (which is used to mitigate the consequences of postulated accidents). This condition affected the safety-related function of the high pressure injection pumps due to inaccurate water level indication in the letdown storage tank. In addition, design configuration control was not maintained for six of twelve valves on the instrumentation lines for the Unit 1, 2, and 3 letdown storage tanks as verified by field observations on May 3, 1997.

G. 10 CFR 50.72 (b)(2) requires, in part, that the licensee shall notify the NRC as soon as practicable and in all cases, within four hours of the occurrence of any event or condition that alone could have prevented the fulfillment of the safety functions of systems that are needed to mitigate the consequences of an accident.

The licensee failed to report within four hours a condition that alone could have prevented the fulfillment of the safety function of the HPI system. Specifically, at 3:15 p.m. on May 5, 1997, a licensee engineering evaluation concluded that the Unit 3 High Pressure Injection System would not have been able to perform its intended safety function (mitigate a small break loss of coolant accident) during power operating conditions from "February 22, 1997 until May 3, 1997." On May 6, 1997, at 6:56 p.m., the licensee submitted a facsimile reporting the condition, a period in excess of four hours.

NOTE: THE APPARENT VIOLATIONS DISCUSSED IN THIS PREDECISIONAL ENFORCEMENT CONFERENCE ARE SUBJECT TO FURTHER REVIEW AND ARE SUBJECT TO CHANGE PRIOR TO ANY RESULTING ENFORCEMENT DECISION.

Oconee Nuclear Station

Predecisional Enforcement Conference

July 23, 1997

ENCLOSURE 4

Agenda

- Introduction
- Perspective on Recent ONS Performance
- Management Focus
- Unit 2 HPI Line Crack Apparent Violations
  - » Root Causes
  - » Corrective Actions
  - » Safety Significance
- Unit 3 HPI System Degradation Apparent Violations
  - » Root Causes
  - » Corrective Actions
  - » Safety Significance
- Closing Remarks

Oconee Performance

- Too many events
- Situation assessment
  - » Personal observations
  - » NSRB
  - » Engineering Review
  - » SEITs
- Addressing issues

Oconee Performance

- Corporate commitment to the long-term safe operation of Oconee

Oconee Performance

- Activities
  - » Walking, talking, observing, listening
- Perspective
  - » Dedicated staff
  - » Aware of issues and root causes
  - » Initiatives are addressing issues
  - » Real equipment and design issues
  - » Process improvements can help
  - » Performance improvements needed
  - » Accountabilities need strengthening

Short-term strategy

- Defenses to protect front line
  - » Control room activities
    - Implemented management oversight of startup and shutdown evolutions
    - Implemented "Operations Core Values"
    - Implemented enhanced expectations for control room monitoring

Short-term strategy

- Defenses to protect front line
  - » Engineering/Operations/Maintenance Communications
    - Established daily Engineering management focus meetings
    - Assigned ownership for daily operational concerns
    - Implemented action register
    - Implemented biweekly Operations/Engineering management meetings
    - Implemented monthly Maintenance/Engineering management meetings
  - » Troubleshooting activities
    - Revised expectations and I&C troubleshooting procedure

Short-term strategy

- Defenses to protect front line
  - » ISI activities
  - » Post-modification/maintenance testing activities
  - » Chemistry and Operations interfaces
  - » Others as appropriate

Self Assessment

	Oconee	Management	Industry/Outside
Completed (10/96 - 6/97)	• Procedure Development and Use • Management Expectations • Corrective Actions • Inservice Augmented Inspections • Maintenance Rule • Programs for Maintaining Licensing Basis • Mod Selection and Activation • Mispositioned Events • FIP for Unit 2 HPI line crack • FIP for 3A and 3B HPI Pump Event • FIPs for Keowee Equipment • Innage Planning • Training • PIP Screening Team • Operations and Work Control Interfaces • Common Cause Analysis	• Independent Restart Review from 3 Unit Outage • Engineering Resources • Unit 2 Drain Line Rupture EIT • SEIT for 3A and 3B HPI Pump Event	• WANO Peer Review • Maintenance Work Practices INPO Assist • NRC Resident Inspections • NRC Maintenance Rule Inspection • Other NRC Inspections
Ongoing	• Chemistry • Radiation Protection • Operating Experience • Engineering Commitments • Reportability	• Special Operations Assessment	• NRC Resident Inspections
Planned	• SITA Corrective Action • Consolidated Performance • Corrective Action	• Special Maintenance Assessment • Special Engineering Assessment • MS Tuckman progress updates • MS Tuckman employee feedback sessions	• SOER 96-01 Control Room Supervision Assist • NRC Resident Inspections

Near Term Assessment

- Assessments
  - » Site internal assessments
  - » General Office led assessments
- Identify immediate actions needed
- Prioritized recommendations
- Fills gap between longer-term initiatives and temporary defenses
- Will identify other initiatives needed

Performance Improvement

- Clarify expectations
  - » Management and "all hands" meetings
  - » Face-to-face interaction
- Site focus adjustment
- Identify action items from field observations and self assessments
- Human performance improvement initiatives
  - » tools in place
  - » accountability
- Monitor, feedback, adjust

Equipment and Design Improvement

- Continue to address equipment and design issues
  - » Oconee Safety Related Designation Clarification (OSRDC) Project
  - » Emergency Power
  - » Service Water
  - » Integrated Control System (ICS)
  - » Operator Aid Computer (OAC)
  - » Reactor Coolant Pumps
  - » Others

Process Improvement

- More formalized, structured processes
  - » Prioritize actions from near-term assessment
  - » Selected changes as part of short-term defenses
  - » Strengthened self assessment processes and corrective action program trending will identify further process improvements

NRC Communications

- Regular updates with Region II
- Resident staff interface

Unit 2 HPI System Weld Crack

- April 21, 1997 RCS leakage caused by HPI nozzle safe-end to injection line weld crack on Unit 2
- RCS pressure boundary leak within makeup capacity of HPI System
- Unit 2 was safely shut down
- FIP team thoroughly investigated the pipe crack
- Breakdown of HPI nozzle/injection line augmented inspection program
- Issue previously identified by operating experience from the 1980s

Apparent Violation 97-07-01

- Failure to implement an effective HPI nozzle inspection program based on available industry recommendations
  - » Weaknesses in RT Procedures
    - – No definitive acceptance criteria
    - – Absence of measures to assure consistency from one exam to the next
  - » Failure to perform UT of safe-end/piping welds or adjacent piping
  - » Augmented examinations not performed in RFO 16

Apparent Violation 97-07-01

- Root Cause:
  - » Change Management
    - – "The organization responsible for determining and initiating the changes to satisfy the commitment requirements did not verify the effectiveness of their actions"

Apparent Violation 97-07-01

- Completed Corrective Actions:
  - » FIP team thoroughly investigated the cause of the leak
  - » Reviewed all thermal sleeve gap RT results since 1983
  - » Shut down Unit 3 to inspect nozzle components
  - » Replaced thermal sleeve and safe end with new design for the 2A1 and 3A1 HPI normal injection lines
  - » Per Unit 1 JCO, additional administrative precautions were implemented until Unit 1 was shut down on June 13, 1997
  - » Unit 1 examinations confirmed JCO bases--no change in gaps, no recordable UT indications
  - » Performed augmented inspections of HPI nozzle components for all three units (exceeded Generic Letter 85-20 requirements)

Apparent Violation 97-07-01

- Completed Corrective Actions:
  - » Detailed cause analysis completed for breakdown in augmented inspection program
  - » Evaluated current commitment management process to ensure it would be effective in minimizing the potential for this type of error
  - » Installed temporary instrumentation to monitor HPI nozzles/flow
    - – All four injection lines on Units 1 and 2
    - – Normal makeup injection lines on Unit 3
    - – Added computer point for makeup flow on all 3 units
  - » Verified compliance with all other augmented inspection requirements

Apparent Violation 97-07-01

- Remaining Corrective Actions:
  - » Complete assessment of other programmatic engineering commitments
  - » Improve nozzle component examination program
    - – The ISI Plan will be revised to include an entry for each nozzle "component", its inspection procedure number and outage numbers in which the examination is to be performed
    - – Develop specific RT and UT examination procedures for augmented examinations for the nozzle components
    - – Improvements will be complete by September 1997 and used during U1EOC17
    - – General Office audit of effectiveness of corrective actions in 1998
  - » Evaluate warming line flow and Operations procedures to minimize HPI nozzle component thermal stresses

Apparent Violation 97-07-01

- Remaining Corrective Actions:
  - » Operating Experience (OE) Assessment
    - – Reviewing approximately 1500 OE documents from 1982 to 1992
    - – Each OE document will be rescreened for applicability
    - – Acceptability of proposed corrective actions for applicable OE issues will be assessed
    - – Verify documents to assure corrective actions were implemented
  - » Near term assessments to prioritize key areas for process improvements

Apparent Violation 97-07-02

- Failure to evaluate known problems effectively and implement appropriate corrective actions in that ONS inadequately addressed thermal stratification
  - » Impact of 1990 temperature data on previous analyses (GL 85-20 and Bulletin 88-08) not addressed
  - » Changes to plant procedures based on 1990 data not implemented
- Our FIP team identified this error during its review of the Unit 2 event

Apparent Violation 97-07-02

- Root Cause:
  - » Lack of formality in turnover process and tracking system between the responsible engineers in early 1990s resulted in this issue not being appropriately addressed

Apparent Violation 97-07-02

- Completed Corrective Actions:
  - » Current processes to track open engineering issues to resolution are rigorous
  - » Installed temporary temperature instrumentation to monitor HPI nozzles
  - » Completed interim analysis of 1990 thermocouple data with acceptable results
  - » Established Structural Integrity Issues Management (SIIM) team to integrate component, system and stress engineering activities

Apparent Violation 97-07-02

- Remaining Corrective Actions:
  - » Collect temperature data that will be used to:
    - – Confirm appropriate augmented inspection program for HPI nozzles/injection lines
    - – Develop appropriate boundary conditions for thermal fatigue analyses
    - – Modify plant operation to minimize the number of thermal cycles on the HPI lines
  - » Develop the Engineering Support Program document for nozzles

Apparent Violation 97-07-02

- Remaining Corrective Actions:
  - » Submit supplemental information to NRC
  - » Replace HPI check valves and piping from safe-ends to check valves during the upcoming Unit 1 refueling outage
  - » Perform video inspections of all safe-ends and thermal sleeves during the upcoming Unit 1 refueling outage
  - » Complete Class 1 fatigue analysis on HPI System injection lines
  - » Perform design basis inspection of HPI/LPI Systems

Safety Significance

- Unit 2 HPI Line Leak
  - » RCS pressure boundary leak degraded a fission product barrier
  - » Leakage was well within makeup capacity of HPI System
  - » Unit 2 was safely shut down
  - » Leak was at a location analyzed in UFSAR LOCA analysis
  - » Duke analysis concluded event is not a precursor

Unit 3 HPI System Degradation

- LDST level lost during May 3, 1997 cooldown
- 3A and 3B HPI pumps rendered inoperable during the event
- Unit 3 safely brought to cold shutdown
- LDST level error resulted in operation of HPI System outside of Technical Specification requirements
- Control room performance contributed to the event
- Duke analysis concludes event is a precursor

Apparent Violation 97-08-01

- Failure to adhere to Technical Specification operability requirements for the HPI System on Unit 3
  - » HPI System was inoperable for some indeterminate period of time between March 6, 1997 and May 2, 1997

Apparent Violation 97-08-01

- Root Cause:
  - » "Combination of a design weakness of a common reference leg for the LDST level instruments and a leaking instrument fitting due to inadequate work practices"
  - » "Contributing cause was the failure to adequately apply available operating experience"

Apparent Violation 97-08-01

- Completed Corrective Actions:
  - » Completed detailed FIP and SEIT investigations
  - » Completed LDST modifications on all three units
    - – Added a separate reference leg for each LDST level transmitter
    - – Added a redundant LDST pressure transmitter
  - » Repaired, inspected, flushed, and tested Unit 3 HPI System
  - » Per Unit 1 JCO, monitored LDST level reference leg until Unit 1 was shut down on June 13, 1997
  - » Evaluated applicability of event to other tank level instruments
  - » Improved foreign material and damage inspection work practices for tubing caps and fittings

Apparent Violation 97-08-01

- Remaining Corrective Actions:
  - » HPI reliability study will be completed
    - – Objective of the study is to assess HPI System reliability for:
      - PRA events of interest
      - Design basis functions
      - Functions during normal operation
    - – Study will be submitted to the staff by December 31, 1997

Apparent Violation 97-08-01

- Remaining Corrective Actions
  - » Will complete design basis inspection of HPI and LPI Systems described under Violation 97-07-02
  - » Will complete operating experience assessment described under Violation 97-07-01

Apparent Violation 97-08-02

- Failure to follow procedures during unit cooldown and in response to off-normal conditions with three examples:
  - » Operations crew failed to appropriately monitor inventory parameters during May 3, 1997 cooldown
  - » Operations crew took actions for greater than 15 minutes without referring to procedures
  - » Dedicated LTOP operator was also the operator at the controls

Apparent Violation 97-08-02

- Duke agrees with the apparent violation
- We believe the second example is invalid based on additional facts

Apparent Violation 97-08-02

Root Causes:

» Insufficient Operations management and supervisory reinforcement of high standards for procedure use and adherence

» Insufficient emphasis on RCS inventory balancing and plant monitoring techniques

Contributing Causes:

» Weaknesses in related Operations procedures and directives

Apparent Violation 97-08-02

Completed Corrective Actions:

- Clarified Expectations:
  - » Operations Shift Managers instituted five "OPS Core Values " including: Plant Monitoring; Procedure Quality and Use; Log Keeping
  - » Revised Operations Management Procedures to clarify expectations for use of procedures and performance of operator rounds
  - » Revised OP/1,2,3/A/1104/49 - HPI System Operation- to clarify dedicated LTOP operator responsibilities
  - » Operations personnel instructed by Operations Superintendent and Shift Operations Manager regarding lessons from the loss of HPI event

Apparent Violation 97-08-02

Completed Corrective Actions:

- Improved Procedures & Tools:
  - » Benchmarked and revised two Abnormal Procedures for all three units, including AP/1,2,3/A/1700/14 - Loss of HPI Makeup/Letdown
  - » Provided computer graphic aid for LDST level/pressure monitoring
- Temporary defenses established for control room procedure use and adherence and plant monitoring

Apparent Violation 97-08-02

Remaining Corrective Actions:

» Raise standards through industry benchmarking and self assessment

- Improve operator requalification training on procedure use and adherence
- Simulator training on loss of HPI suction events with upgraded AP
- Benchmark and upgrade additional abnormal procedures
- Complete on-going assessment and upgrade of Operations procedures
- Improve techniques and standards for plant monitoring
- Improve guidance, computer-based monitoring tools, and associated training for RCS inventory monitoring during startup and shutdown

Apparent Violation 97-08-03

- Three examples of failure to take corrective actions for conditions adverse to quality
  - » Corrective actions for past problems associated with safety-related maintenance activities were inadequate
  - » 1983 loss of HPI suction modification was canceled in 1986
  - » 1988 request by Operations for loss of HPI suction modification not installed as of May 1997

Apparent Violation 97-08-03

- Duke admits to the violation
  - » Duke agrees that certain Maintenance work practices with fittings needed improvement
  - » Duke agrees that prior opportunities existed through Information Notices to address the common LDST reference leg design
  - » For Example 3, a previously considered design option may have prevented the event
- Modifications as scoped in Examples 2 and 3 would not have prevented the event

Apparent Violation 97-08-03

- Root Causes:
  - » Maintenance practices for fittings:
    - – Inadequate work practice guidance resulted in personnel not checking fittings for foreign material and material condition
    - – Neither mixed fittings nor over torquing contributed to the event
  - » Common reference leg design:
    - – Weaknesses in past processes for evaluating industry operating experience

Apparent Violation 97-08-03

- Completed Corrective Actions
  - » Revised procedure to:
    - – require visual inspections for damage and foreign material exclusion
    - – provide specific instructions to prevent overtightening of fittings
    - – detect and eliminate all mixed parts
  - » Trained Maintenance personnel on improved fitting work practices
  - » Corrective actions for LDST level design are addressed in Violation 97-08-01 response

Apparent Violation 97-08-03

- Remaining Corrective Actions:
  - » Operating experience assessment described under Violation 97-07-01 will be completed
  - » Revising work planning process to assure appropriate procedures are referenced
  - » Continuing emphasis of management expectations regarding the use of mixed fittings and installation work practices for fittings via classroom training and management observations

Apparent Violation 97-08-04

- Lack of design control for LDST instrumentation
  - » Mislabeled root valves
  - » QA classification of LDST level and pressure instrumentation

Apparent Violation 97-08-04

- Discussion:
  - » Root cause of mislabeled instrument valves was a failure to properly verify configuration when valve labels were added
  - » Duke believes LDST level and pressure instrumentation was properly classified

Apparent Violation 97-08-04

- Completed Corrective Actions:
  - » Corrected valve labeling deficiencies on all three units
  - » Assessed the need for a more thorough review of plant labeling for instrument valves
  - » Completed LDST modifications on all three units
    - Added a separate reference leg for each LDST level transmitter
    - Added a redundant LDST pressure transmitter
  - » Removed continuous fill line on all three units

Apparent Violation 97-08-04

- Remaining Corrective Actions
  - » Root valve position verification program will be expanded to include critical root valves outside containment and/or where position is not self-revealing
  - » Continuing management emphasis on proper control of instrument valve position
  - » Based on lessons learned from this event, Duke is taking the conservative step to voluntarily upgrade the LDST instrumentation to QA-1
    - – Scope of upgrade may be impacted by HPI Reliability Study and HPI/LPI Design Basis Inspection

Apparent Violation 97-08-05

- Failure to make a report within the time required by 10CFR 50.72 (b)
  - » Duke analysis determined on May 5, 1997 that HPI system had been inoperable between March 6, 1997 and May 2, 1997
  - » 50.72 notification made on May 6, 1997

Apparent Violation 97-08-05

- Discussion:
  - » Event was reported under 50.72 (a) on May 3, 1997 and communicated
    - 2 of 3 HPI pumps were inoperable
    - We suspected LDST level indications were erroneous
    - Fluctuations in discharge pressure and pump amps
  - » Reportability was further evaluated by Oconee on May 5, 1997
    - 10CFR 50.72 states events reported under 50.72 (a) do not need to be reported again under 50.72 (b)
    - New information on May 5, 1997 was the duration that the HPI System had been inoperable
  - » New information did not appear to meet supplemental reporting requirements of 50.72 (c) since it did not affect changes in plant conditions
  - » Additional ENS notification made on May 6, 1997 to provide information on past inoperability of the system

Apparent Violation 97-08-05

- Discussion:
  - » LER on past inoperability of HPI System submitted on June 2, 1997 per 10CFR 50.73
  - » Duke procedures and NUREG-1022 do not provide clear guidance on supplemental notifications
  - » Duke would appreciate further staff input to assure we properly update 50.72 reports

Safety Significance

- Duke analysis concludes past inoperability of HPI System results in the event being a precursor
- LER 287/97-03 thoroughly addresses safety significance
  - » LDST error resulted in Unit 3 HPI System operating outside the design basis
- Unit 3 HPI System degradation during May 3, 1997 cooldown
  - » 3A and 3B HPI pumps rendered inoperable during the event
  - » Unit 3 safely brought to cold shutdown
  - » Little risk of core damage during the actual Unit 3 shutdown

Closing Remarks

May 3, 1997

1:00am	LDST H_2 vented. RCS temperature is ~300°F
1:19am	1540 gallons of coolant added to LDST (1 add)
2:40am	RCS <250°F and 350 psig, 3B HPI pump running, 3A HPI pump in standby
3:00am	RCS temperature is ~240°F. Pressurizer level stabilized at ~100 inches.
7:00am	RCS pressure 272 psig, temperature 236°F
7:45am	LDST level indicates 55.9 inches and is no longer decreasing - this indication exists for ~1.75 hours.
8:00am	RCS temperature is approximately 225°F. Cooldown has resumed following shift turnover.
9:00am	RCS temperature is ~205°F
9:07am	Operators secure 3A2 RCP (One RCP left running)
9:12am	The licensee estimates that the LDST and HPI pump suction piping were empty at this time.
9:13am	HPIP DISCHARGE HEADER PRESSURE LOW statalarm received and cleared. Reactor Operator notes RCS makeup flow normal. ***Alarm response guides do not direct entry into AP for loss of HPI Makeup/Letdown until letdown is isolated at 0932.***
9:14am	HPIP DISCHARGE HEADER PRESSURE LOW statalarm was received a second time; 3HP-31 observed to be cycling and RCP TOTAL SEAL FLOW LOW statalarm received. Pump discharge pressure fluctuating between normal and low pressure. ***Supplemental RO begins entering several ARGs, including ARG for this statalarm.***
9:15am	HPI pump 3A auto starts from a low seal injection flow signal. Operator shuts HPI pump 3A down as seal injection flow appears to be normal. ***Supplemental RO enters AP for abnormal RCP operation, based on ARG direction, between ~0915 and ~0918.***
9:16am	Operator places HPI pump 3A in auto and pump restarts. Operators observed high amps on 3A HPI pump motor and low amps on 3B HPI pump motor. HPIP DISCHARGE HEADER PRESSURE LOW AND RCP TOTAL SEAL INJECTION FLOW LOW alarms are both received.
9:17am	Operator shuts down 3B HPI pump. 3A HPI pump amps, at 70-120, were running higher that 3B HPI pump amps, at 10 (50 is normal). The operators thought that the 3A HPI pump was pumping water, while 3B HPI pump was not. Valve 3HP-31 was shut by operators.
~9:18am	***OSM enters control room, observes that supplemental RO is using an AP.***

07/22/97

MEMO TO FILE

Re: Oconee Nuclear Station
QA Classification of LDST Instrumentation
File No: OS-114

In May 1997, Unit 3 was proceeding to cold shutdown to inspect HPI nozzles. A partially drained reference leg on the Letdown Storage Tank (LDST) level instrument caused the tank to appear full when in reality it was almost empty. As a result, as the RCS was cooling down, the level dropped in the tank, starving the HPI pump suction. Two HPI Pumps were rendered inoperable. Based on the lessons learned from this event, Oconee intends to upgrade the pressure and level instrumentation to QA-1. The scope of the upgrade may be impacted by the HPI Reliability Study and HPI/LPI Design Basis Inspection, both of which will be completed later this year.

The NRC Augmented Inspection Team (AIT) reviewed the event and concluded that because the LDST level instrument was important to ensuring the readiness of the HPI Pumps, this instrumentation should have been designated as safety related. However, it is clear that the current classification of the instrumentation as non safety-related is consistent with Oconee's licensing basis.

The issue of QA classification of structures, systems and components (SSCs) has been previously addressed in correspondence between Duke and the NRC. The fact that Oconee was licensed prior to the issuance of many current day standards that define QA classification methods has primarily led to differences in interpretation. In order to clarify our position and resolve any misunderstandings, Duke submitted a letter to the NRC dated April 12, 1995 describing what we believed to be our licensing basis for classification of QA1 SSCs (Attachment I). On August 3, 1995 the NRC responded with an SER concurring with our position (Attachment II). Also attached is an excerpt from an internal Duke document, OSC-6100, which gives a historical narrative and interpretation of the licensing basis relative to QA classification (Attachment III). Two important points are concluded from these documents. First, the licensed QA program at Oconee is unique in that it is <u>not function based</u>, and second, the SSCs licensed to be QA1 are addressed in UFSAR 3.1.1.

Specifically how this classification relates to the LDST instrument is as follows: UFSAR 3.1.1 states that quality standards are applied to, "...systems, structures, and components (SSCs) essential to accident prevention and to mitigation of accident consequences..." Those essential systems are then listed. The LDST instrumentation is used to establish the proper initial conditions in the tank. It is not used to prevent a LOCA nor to mitigate its effects. Additionally, the classification of the LDST level instrument was previously accepted by the NRC in our response to RG 1.97. In this response we stated that the "... tank is not required to be utilized during an accident..." nor did we categorize this instrument as a Type A variable.

Although the QA classification of SSCs is unique at Oconee, we believe the classification of this instrument is not. It is not uncommon for other plants to rely on non-safety instrumentation to monitor plant conditions, ensuring that they remain within the bounds of safety analyses and/or Technical Specifications. EPRI has issued a guidance document, EPRI NP-6895s, "Guidelines for the Safety Classification of Systems, Components, and Parts Used in Nuclear Power Plant Applications" which states that safety classification is based on an item's function in mitigation of a design basis event, not on its "...availability and readiness... for the accomplishment of safety-related functions."

Therefore, it is our position that the LDST instrumentation was properly categorized as non-safety related with no QA requirements. However, to further upgrade the reliability of this instrumentation, Oconee is taking the steps to upgrade the instrumentation. The scope of the upgrade will consider the results of the reliability study by the PRA Group and an independent audit of our HPI System.

S. L. Nader

S.L.Nader
Engineering Supervisor II

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DUKE POWER

April 12, 1995

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

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Oconee QA-1 Licensing Basis and Generic Letter 83-28, Section 2.2.1, Subpart 1 Supplemental Response

Dear Sir:

Please find attached, as discussed in the February 6, 1995 "Oconee Safety-Related Classification Issues" Meeting with the NRC in Atlanta, a description of the Oconee QA-1 licensing basis. Attachment 1 provides a detailed description of the history of Oconee's QA-1 licensing basis. Attachment 2 provides the Oconee licensing position on Generic Letter (GL) 83-28. Attachment 3 provides a supplemental response to Subpart 1 of Section 2.2.1 of GL 83-28 and the general criteria for classifying QA-1 SSCs. Attachment 4 provides Oconee's position on Non Oconee QA-1 structures, systems, and components (SSCs) which are used to mitigate accidents. Attachment 5 defines terms used in this document and should be reviewed first. Other attachments are provided as necessary to support points discussed in these attachments.

Duke requests NRC review and approval for Attachment 3.

If there are any questions regarding this document, David Nix can be contacted at (803) 885-3634.

Very truly yours,

BWPeele for
J. W. Hampton

Attachments

9504180256 38pp.

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QVD - EC12A
NSRB - EC12A
OS-815.01
ELL - EC050

ATTACHMENT 1
HISTORY OF OCONEE QA-1 LICENSING BASIS

A description of the historical development of Oconee's QA program is provided below. Attachment 1 also provides an historical perspective of why non QA-1 SSCs are credited for accident mitigation in the Oconee safety analyses.

History of Development of Oconee's QA (QA-1) Program

Pre-PSAR: AEC/DPC Dialogue

Before the issuance of NRC guidance documents, there is very little information available regarding the definition of safety-related SSCs. Consistent with available Atomic Energy Commission (AEC) guidance at the time, the nuclear industry developed its own definition of safety-related. The AEC guidance consisted primarily of correspondence between the AEC and the utilities building nuclear power plants. Formal guidance such as Regulatory Guides did not exist during the early Oconee construction era. The design and construction of early commercial nuclear power plants was performed using available mechanical, civil, and electrical codes. SSCs were classified at ONS consistent with this code-based approach.

Early Duke Design and Piping classification

During the design and procurement of materials for Oconee mechanical systems, the Design Engineering Department of Duke Power Company defined the term safety related as 1) nuclear piping per USAS B31.7 nuclear piping code or 2) specifically identified large break LOCA (LBLOCA) mitigation (e.g., Low Pressure Service Water) non-nuclear, seismically qualified piping per USAS B31.1 piping code. It appears that USAS B31.7 originated the term nuclear safety-related. The Duke piping classification system essentially reflected the USAS B31.7 definition. This code-derived definition of safety-related does not correlate well with the functionally based definition eventually developed by the NRC. It should be noted that the NRC definition of safety-related was not issued prior to construction of ONS. Based on conversations with available engineering personnel involved with the construction of Oconee, electrical power supplies to components within these Duke piping systems appeared to be classified consistent with the piping classification and functional requirements. Oconee Electrical Design relied heavily on Mechanical Design during the Oconee construction to provide information on whether a component required IEEE class 1E power. This appears to be why several B31.1 systems not required for LBLOCA/LOOP mitigation (such as Main Steam) had non QA-1 power supplies to valves.

An August 14, 1972 internal letter by Mr. Robert E. Miller illustrates the relationship between safety-related systems and their piping classification:

> "... system (piping) classifications for the Oconee Nuclear Station were conceived and established in early 1968, materials were procured in 1968 and erection requirements were established and used in 1969. Duke's system classification has been defined in the FSAR and all safety-related system diagrams, as found in Sections 6 and 9 of the FSAR, describe in detail the applicable system classifications, all of which has been reviewed by the AEC/ACRS.
>
> The AEC A-D (eventually in Regulatory Guide 1.26) system classification did not emerge until after Oconee was well underway and has never been imposed on the Oconee design. In fact, the AEC made known its A-D system classification to Duke only on the McGuire Nuclear Station in a letter received by Duke on April 14, 1971, from Dr. Peter A. Morris (AEC)."

The Regulatory Guide 1.26 system listing was consistent with the functional definition of safety-related provided by the NRC in Regulatory Guide 1.29 and Generic Letter 83-28. However, Duke's piping classification was not consistent with the Reg Guide 1.26 listing since the Reg Guide 1.26 listing was more functionally based than the Duke classifications. Duke developed the Duke Piping Classification System by use of available ASME codes.

A description of the design and classification of piping is provided to demonstrate its relationship to the original quality assurance program. The application of the original quality assurance program to Oconee systems was consistent with the original piping classification philosophy.

<u>PSAR</u>

The PSAR was developed based on pre-construction permit dialogue between Duke, the AEC Staff, and the Advisory Committee for Reactor Safeguards (ACRS). As a result of the dialogue between Duke, the AEC, and the ACRS, the scope of equipment considered as safety-related by Duke first appeared in Section 1.4.1 of the PSAR. Duke referred to this equipment as "essential to accident prevention and to mitigation of accident consequences". The PSAR was submitted in late 1966/early 1967 as part of the construction permit application for Oconee Nuclear Station. Section 1.4.1 of the PSAR provides a reply to Criterion 1 "Quality Standards" as proposed by the AEC in a November 22, 1965 press release H-252 in its "General Design

Criteria for Nuclear Power Plant Construction Permits". Section 1.4.1 of the PSAR states that:

> "... the integrity of SSCs essential to accident prevention and to mitigation of accident consequences has been considered in the design evaluations. These SSCs are:
>
> 1. Fuel Assemblies
> 2. Reactor Coolant System
> 3. Reactor instrumentation, controls, and protective systems
> 4. Engineered safeguards systems
> 5. Radioactive materials handling systems
> 6. Reactor building
> 7. Electric power sources "

These systems clearly do not encompass all presently postulated accidents. However, this list of systems does envelop the majority of SSCs required for mitigation of the postulated large break loss of coolant accident/loss of offsite power (LBLOCA/LOOP) accident. There is also other equipment in this list whose integrity was considered necessary to prevent offsite dose to the public (i.e., B31.7 code-designed nuclear piping and radioactive materials handling systems).

The AEC and ACRS correspondence throughout the pre-construction permit period indicates that the regulatory focus was on LBLOCA mitigation. This is apparent by the repeated reference to "LOCA" and the number of questions tied to systems needed for LBLOCA mitigation.

For example, on June 13, 1967, the AEC issued a Safety Analysis for Instrumentation, Control, and Power. The section on the Engineered Safety Feature Protection System states, in part, that "the Engineered Safety Feature Protection System automatically performs the following functions to mitigate the effects of a serious accident: a) initiates operation of the core emergency injection system upon detection of a low reactor coolant pressure. b) initiates operation of the Reactor Building Cooling System upon detection of an abnormally high Reactor Building pressure. c) initiates containment isolation upon detection of an abnormally high Reactor Building pressure." All of these conditions are indicative of a LBLOCA.

In addition, dialogue between the AEC and Duke focused on the quality assurance of primary systems rather than secondary systems. For example, a July 11, 1967 AEC letter from N. J. Palladino (ACRS) to the Honorable Glenn T. Seaborg (AEC Chairman), states in part:

> "The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system as well as inspection during service life, and recommends that the applicant (Duke; for Construction Permit) implement these

improvements in primary system quality that are practical with current technology."

Pre-FSAR/Post PSAR Duke/AEC Dialogue

AEC approval was received in November 1967 to begin construction of Oconee Nuclear Station. The systems listed in the Design Criterion 1 Reply of the PSAR were constructed to the quality assurance requirements of that period under direction of Duke Power's Design Engineering and Construction Departments. A letter dated February 13, 1974 from A. C. Thies (DPC) to Mr. Voss A. Moore (AEC) provides a perspective of the development of the early Duke QA program:

> "The attachment (to the subject letter) does not describe the QA program which existed during the early design and construction of Oconee Nuclear Station. It does, however, present the corporate QA organization which, during this period (approximately late 1973/early 1974), has evolved due to experience gained in construction of Oconee and issuance of recent AEC guidelines."

The AEC guidance regarding the scope of a nuclear power plant quality assurance program was also changing during the construction of Oconee Nuclear Station as can be observed in the development of GDC-1 "Quality Standards".

During this period, the ACRS/Duke correspondence clearly focused on the primary system, containment, ES-actuated systems, power sources needed to directly mitigate a LBLOCA, and prevention of the release of radiation to the public. LBLOCA mitigation was the subject of most of the correspondence during this period of FSAR development.

FSAR

While construction on Oconee Nuclear Station continued, Duke developed the applications for operating licenses through correspondence with the AEC. The Final Safety Analysis Report (FSAR) was submitted to the AEC in October 1971. This submittal for the Oconee operating licenses addressed Duke Power Company's finalized version of the original quality assurance program. This is delineated in the original FSAR Appendix 1A, Criterion 1 "Quality Standards". This is now Section 3.1.1 of the FSAR. The Oconee quality assurance program was provided in reply to General Design Criterion 1 of the seventy "General Design Criteria (GDC) for Nuclear Power Plant Construction Permits". These GDC were proposed by the AEC in a proposed rulemaking for 10CFR 50 published in the Federal Register on July 11, 1967. The Oconee FSAR states that:

> "the integrity of SSCs essential to accident prevention and to

mitigation of accident consequences has been included in the reactor design evaluations. These SSCs are:

1. Reactor Coolant System
2. Reactor Vessel Internals
3. Reactor Building
4. Engineered Safeguards System
5. Electric Emergency Power Sources"

This list of SSCs comprised the original Oconee QA-1 SSCs to which 10CFR50 Appendix B would be applicable. These systems mitigate a LBLOCA/LOOP. While there are differences between this list and the list provided in the PSAR, the items deleted were only those SSCs not required for mitigation of a LBLOCA/LOOP and the additional SSCs already covered by Appendix 1B of the FSAR.

An important point is that the scope of the Oconee QA-1 program was not required to encompass all SSCs requiring seismic design criteria or single failure design criteria. The scope and applicability of seismic and single failure design criteria are described under different design criteria. There were many SSCs that were seismically designed which did not fall under the scope of the original Oconee QA-1 program. FSAR Section 3.2.2 gives some examples such as the CCW intake structure, CCW pumps, upper surge tanks, and emergency feedwater pumps. Although this is not consistent with current NRC guidance, it is clear that some seismically designed, single failure proof systems were not classified as QA-1 when Oconee received its license. However, all SSCs that fell within the original Oconee QA-1 program met both single-failure and seismic design criteria.

The original FSAR list of SSCs that made up Oconee's QA-1 program is further supported by FSAR Appendix 1B, which at that time provided a more specific list of SSCs which Oconee intended to treat Oconee QA-1. The SSC list was in Appendix 1B of the FSAR at the time the operating licenses were granted for Units 1,2, and 3 (in 1973), and was recognized and approved by the AEC in a Division of Reactor Licensing report dated July 24, 1970, to the ACRS. Based on review and on site inspection, this report concluded the "quality assurance program, as described in the FSAR will assure an acceptable level of quality of the safety-related systems, equipment, and structures incorporated in Oconee Station Units 1, 2, and 3".

The FSAR Appendix 1B list of SSCs is consistent with the Duke philosophy regarding application of the quality assurance program at Oconee during construction and licensing. The SSCs shown in this list, with few exceptions, are items which were:

1) Necessary to mitigate a LBLOCA/LOOP design basis accident, or

2) Pressure boundary to prevent release of radioactive fluids which if released could present a danger to the public (as determined by dose levels), or,

3) Electrical/Instrumentation items designed per draft IEEE 279 Class 1E.

There are some examples of SSCs which did not appear on the Appendix 1B list which are required for mitigating a LBLOCA/LOOP, such as portions of the Condenser Circulating Water (CCW) System. However, at the time of construction of Oconee, the engineers recognized that some features of these non-nuclear, USAS B31.1.0 systems permitted their exclusion from the quality assurance program. These features were:

* redundancy and diversity,
* passive mitigation functions,
* seismic design, and
* constant use of these systems in normal operation of the plant.

Post-FSAR Correspondence

Until early 1974, the only detailed list of Oconee QA-1 SSCs was contained within FSAR Appendix 1B. To establish the component level boundaries for testing and maintenance, Appendix 1B of the FSAR was replaced by the Duke QA Topical Report in Revision 36 of the FSAR on July 21, 1975. However, the Duke QA Topical Report, now Chapter 17 of the FSAR, did not provide a specific list of structures, systems, and components that are considered Oconee QA-1. This left only the FSAR reply to Criterion 1, along with any additional commitments on docket since development of the FSAR, to provide the licensed scope of the Oconee QA-1 program.

In a letter dated April 27, 1973 from A. C. Thies (DPC) to Mr. R. C. Young (USAEC), Duke states that "Personnel are presently in the process of specifying those structures, systems, and components which are considered safety-related and must be addressed by the operational quality assurance program. This development should be completed no later than December 31, 1973." The development of a detailed Oconee QA-1 SSC list resulted from; 1) the issuance of the Facility Operating License for Units 1, 2, and 3 in 1973-1974, which required inclusion of all Oconee QA-1 SSCs into the ISI/IST programs and 2) the replacement of FSAR Appendix 1B by the Duke QA Topical Report. This list of Oconee QA-1 SSCs became known as the "Safety-Related Structures, Systems, and Components (SRSSC) Manual". The SRSSC Manual was not submitted to the NRC for review and approval. The SRSSC Manual effectively replaced the Appendix 1B list of SSCs in the FSAR, although it was not presented as the Oconee licensing basis nor intended to conflict with the FSAR. The SRSSC Manual list of SSCs was not included in the FSAR.

Generic Letter 83-28

On July 8, 1983, the NRC issued Generic Letter (GL) 83-28, titled "Required Actions Based on Generic Implications of Salem ATWS Events". Section 2.2.1 of this letter entitled "Equipment Classification and Vendor Interface" required licensees to submit, for NRC review, a "... description of their programs for safety-related equipment classification".

Duke responded to GL 83-28 Section 2.2.1 in letters dated November 4, 1983, January 17, 1984, and June 9, 1987. In these letters, Duke described the scope of the Oconee operational QA program. The NRC audited this program in late July 1985. A Safety Evaluation Report dated November 4, 1987 was issued which approved the scope of the Oconee operational QA program.

Expansion Beyond the Original Oconee QA-1 Scope

Since the time of the original Oconee design, additional postulated accidents and safety concerns have been addressed through correspondence between Duke and the NRC. In response to regulatory issues Duke upgraded many SSCs to QA-1. Although not all-inclusive, the following examples are provided:

Pipe Break Concerns - In 1972-1973, Duke responded to the AEC's concerns regarding the impact of postulated pipe breaks on safety-related equipment. In a December 29, 1972 letter to the NRC, Duke commits to treat the Duke Piping Class F portions of Main Steam and Emergency Feedwater as Oconee QA-1. This was the first time that Class F piping, other than portions of the Low Pressure Service Water System, were placed in the Oconee QA-1 program.

TMI Concerns - In the NRC Commissioner's meeting of April 25, 1979, Duke committed to provide improvements to the Emergency Feedwater System (EFW) at Oconee. Several modifications were made to the EFW System as a result of this commitment. One of these modifications installed two Oconee QA-1 motor driven EFW pumps for each Oconee Unit. In addition, all pipes and valves associated with this motor driven EFW pump modification were classified Oconee QA-1.

SSF - During the period 1978-1982, Duke described the conceptual design of the SSF. In a letter dated January 28, 1982, Duke summarized the SSF design and listed the QA-1 portions of the SSF.

Portions of CCW System - In a February 6, 1995 management meeting with the NRC, Duke committed to upgrade portions of the CCW System to Oconee QA-1.

The case-by-case nature in which these SSCs were reclassified as Oconee QA-1 makes it difficult to define a consistent functional dividing line between QA-1 and non QA-1 equipment for accidents other than the Large Break LOCA/LOOP.

Oconee Non QA-1 SSCs Credited for Accident Mitigation

Duke accident analyses take credit for some SSCs not originally licensed as Oconee QA-1 to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100. Duke addressed various NRC concerns through licensing correspondence subsequent to the original licensing of the station. In many cases, no commitment was made to classify these SSCs as Oconee QA-1. Often the decision not to reclassify SSCs required for accident mitigation as QA-1 was made because it was clearly recognized that these systems were not originally procured per existing QA-1 (10 CFR50 Appendix B) requirements nor was there a requirement to incorporate these SSCs into the Oconee QA-1 program. In addition, many of these SSCs met the features of redundancy and diversity, passive mitigation, seismic design, and constant operation. These features diminished any added value that would have been provided by placement under a quality assurance program.

Summary

During the February 6, 1995 management meeting in Atlanta with the NRC, Oconee management provided a brief presentation of this history of Oconee's QA-1 licensing basis. The conclusions drawn were:

* The licensed QA program at Oconee was unique.
* QA-1 at Oconee was originally intended to cover items listed in Section 3.1.1 of the FSAR.
* The reply to General Design Criterion 1 in Section 3.1.1 of the FSAR was intended to include in the QA program those SSCs essential to mitigation of the LBLOCA/LOOP and other primary systems whose releases could result in danger to the public due to radioactive release.
* With an understanding of how Oconee is different, we can move on to how we propose to resolve issues with the QA program.

REFERENCES

1) USAS B31.7 2/68 ed. and 6/68 Errata, "Draft USA Standard Code for Pressure Piping - Nuclear Power Piping", pub. by American Society of Mechanical Engineers.

2) USAS B31.1.0 7/67 ed., "USA Standard Code for Pressure Piping - Power Piping", pub. by American Society of Mechanical Engineers.

3) Conversations with R. E. Miller, K. S. Canady, J. O. Barbour, D. Jamil, S. L. Nader; dtd July 1994 - April 1995.

4) Duke Internal Memo dated 08/14/72 by R. E. Miller (Senior Engineer) documenting information of early Duke piping classification and design of LPSW System and associated dialogue with AEC.

5) NRC Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants", Revision 3, February 1976.

6) NRC Regulatory Guide 1.29, "Seismic Design Classification", Revision 3, September 1978.

7) Oconee Nuclear Station Preliminary Safety Analysis Report (PSAR), Section 1.4.1.

8) AEC Safety Analysis for Instrumentation, Control, and Power dated 06/13/67 to C. G. Long (AEC)

9) Letter dated 07/11/67 from N. J. Palladino (ACRS) to Honorable G. T. Seaborg (AEC Chairman) regarding "Report on Oconee Nuclear Station Units 1,2, and 3".

10) Letter dated 02/13/74 from A. C. Thies (DPC) to V. A. Moore (NRC) regarding the Duke Power Quality Assurance program.

11) Oconee Nuclear Station Final Safety Analysis Report (FSAR), through Revision 37 of 06/03/76, Appendix 1.A.1

12) Oconee Nuclear Station Final Safety Analysis Report (FSAR), 1993 Update, Sections 3.1.1, 3.2.2, 5.2.3.3.6.

13) Oconee Nuclear Station Final Safety Analysis Report (FSAR), through Revision 35 of 09/30/74, Appendix 1B.

14) AEC Report to the Advisory Committee for Reactor Safeguards dated 07/24/70 regarding "Duke Power Company Oconee Nuclear Station Operating License".

15) Draft IEEE 279, 8/68 ed., "Criteria for Nuclear Power Plant Protection Systems"

16) Letter dated 04/27/73 from A. C. Thies (DPC) to R. C. DeYoung (AEC) regarding "Description of the ONS Operational Quality Assurance Program".

17) Duke Power Company Quality Assurance Topical Report, Amendment 18, 12/15/94.

18) Oconee "Safety-Related Structures, Systems, and Components Manual", Revision 3, 09/30/76.

19) NRC Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events", dated 07/08/83.

20) Letter dated 11/04/83 from H. B. Tucker (DPC) to H. R. Denton (NRC) regarding response to Generic Letter 83-28.

21) Letter dated 01/17/84 from H. B. Tucker (DPC) to H. R. Denton (NRC) regarding response to Generic Letter 83-28 item 2.2.1-5.

22) Letter dated 06/09/87 from H. B. Tucker (DPC) to NRC addressing request for additional information on Generic Letter 83-28 items 2.1 and 2.2.

23) NRC Inspection Report 85-21 from V. L. Brownlee (NRC) to H. B. Tucker (DPC), dated 09/13/85.

24) NRC Safety Evaluation Report for Generic Letter 83-28 Item 2.2 Part 1, from H. N. Pastis (NRC) to H. B. Tucker (DPC), dated 11/04/87.

25) Letter dated 12/29/72 from A. C. Thies (DPC) to A. Giambusso (AEC) addressing request for additional information on consequences of postulated pipe failure of the Main Steam and Feedwater piping outside the Reactor Buildings for Oconee Units 1,2, and 3.

26) Letter dated 04/25/79 from W. O. Parker, Jr (DPC) to H. R. Denton (NRC) on Improvements to the Oconee Emergency Feedwater System.

27) NRC Safety Evaluation Report for "Environmental Qualification of Safety-Related Electrical Equipment at Oconee Nuclear Station" dated 05/22/81.

28) Letter dated 01/28/82 from W. O. Parker, Jr (DPC) to H. R. Denton (NRC) on Seismic Qualification of the Oconee Emergency Feedwater System.

29) Letter dated 02/23/95 from A. Gibson (NRC) to J. W. Hampton (DPC) summarizing 02/06/95 Management Meeting in Atlanta regarding safety-related equipment classification.

30) Code of Federal Regulations, Title 10, Part 50, Appendix B, 01/01/94.

ATTACHMENT 2
OCONEE LICENSING POSITION ON GENERIC LETTER 83-28

On July 8, 1983, the NRC issued Generic Letter (GL) 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events". Section 2.2.1 of this letter, entitled "Equipment Classification and Vendor Interface", required licensees to submit, for NRC review, a "... description of their programs for safety-related equipment classification". Subpart 1 of Section 2.2.1 required a discussion of "the criteria for identifying components as safety-related within systems currently classified as safety-related".

In a letter dated November 4, 1983, Duke referenced the "Oconee Nuclear Station (ONS) Safety-Related Structures, Systems and Components Manual (SRSSC)", as the document which provides the criteria for identifying components as safety-related. Duke's response to GL 83-28 Section 2.2.1 was audited in an Inspection Report dated September 13, 1985, and approved in an NRC Safety Evaluation Report dated November 4, 1987.

The Oconee response to Subpart 1 of Section 2.2.1 of GL 83-28 was intended to assure components are treated with the same quality standards as their parent system. This is consistent with the clarification provided in Subpart 1 of Section 2.2.1 of GL 83-28 which states that "...This (criteria to be provided for identifying components as safety-related) shall not be interpreted to require changes in safety classification at the systems level". The Duke response assured that components whose safety classification was unclear would be handled in a conservative manner. The response was intended to apply to components within SSCs already identified as safety-related by the ONS SRSSC Manual. The response was not intended to be construed as a reclassification of the entire scope of the Oconee SSCs to the the functional definition of safety-related provided in GL 83-28.

Duke Power Company recognizes that the criteria contained in our response to Subpart 1 of Section 2.2.1 needs to be revised to reflect the QA-1 licensing basis for Oconee Nuclear Station. Attachment 3 to this submittal provides the general criteria for identifying components as QA-1 at Oconee Nuclear Station. Attachment 3 supersedes earlier submittals related to Subpart 1 of Section 2.2.1 of GL 83-28.

ATTACHMENT 3

SUPPLEMENTAL RESPONSE TO SUBPART 1 OF SECTION 2.2.1 OF GL 83-28

GENERAL CRITERIA FOR CLASSIFYING QA-1 SSCS

This attachment supersedes previous Oconee submittals related to Subpart 1 of Section 2.2.1 of GL 83-28. The general criteria used to determine if a SSC is QA-1 are delineated in this attachment. These general criteria are divided into two categories. The first category provides general QA-1 criteria based on the original licensing basis of ONS. The second category provides general criteria for SSCs that were added to the QA-1 licensing basis after issuance of the original operating licenses for Oconee Nuclear Station. Following NRC review and approval, Duke Power will revise Section 3.1.1 of the FSAR to include the general criteria provided in this attachment. A more detailed QA-1 checklist is being developed to further clarify the general criteria in this attachment.

Original Oconee QA-1 Licensing Basis (FSAR Section 3.1.1)

The integrity of SSCs essential to prevention and mitigation of the Large Break LOCA coincident with loss of offsite power has been included in the reactor design evaluations. These SSCs are:

1. Reactor Coolant System

   From a quality assurance perspective, the Reactor Coolant System consists of all connecting piping, valve bodies, pump casings, heat exchangers, or vessels out to and including the first isolation valve. The integrity of the pressure boundary of the connecting piping, valve bodies, pump casings, heat exchangers, or vessels is the function which determines applicability of the quality assurance program.

2. Reactor Vessel Internals

   The Reactor Vessel internals consist of the plenum assembly and the core support assembly. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance holder tubes. The plenum assembly consists of the upper grid plate, the control rod guide assemblies, and a turnaround baffle for the outlet flow.

   Reactor vessel internals do not include fuel assemblies, control rod assemblies, surveillance specimen assemblies, or incore instrumentation.

3. Reactor Building

The Reactor Building consists of the following:

* The structure, which consists of a post-tensioned reinforced concrete cyclinder and dome connected to and supported by a massive reinforced concrete foundation slab.

* The entire interior surface of the structure (a steel plate liner)

* Welded steel penetrations through which numerous mechanical and electrical systems pass into the Reactor Building.

* Access openings to the Reactor Building

4. Engineered Safeguards System

The Engineered Safeguards System consists of structures, systems, or components necessary to:

* Provide emergency cooling to assure structural integrity of the core:

    High Pressure Injection System

    Low Pressure Injection System

    Core Flooding System

* Maintain the integrity of the Reactor Building

    Reactor Building Spray System

    Reactor Building Cooling System

    Reactor Building Isolation System (this includes all piping penetration isolation paths)

* To collect and filter potential Reactor Building penetration leakage

    Penetration Room Ventilation System

* In addition, support systems necessary to ensure that the above systems can perform their intended safety functions are considered QA-1. These systems are:

Low Pressure Service Water portions necessary to supply cooling water to:

Reactor Building Cooling Units

Decay Heat Removal Coolers

Keowee emergency start, load shed, and emergency power switching logic

Analog and Digital ES Channels and DC Power to support operability of these channels

5. Emergency Electric Power Sources

The following power sources and distribution systems serve QA-1 functions:

* Keowee Hydroelectric Units 1 and 2, including;

  Keowee Hydro-Generator and Emergency Start Circuits,
  Keowee 600/208/120 VAC Auxiliary Power System, and
  Keowee 125 VDC Power System.
  The following mechanical Keowee SSCs:

  - Governor Oil System
  - Governor Air System
  - Guide Bearing Oil System
  - Turbine Sump System
  - Cooling Water System

* Underground Emergency Power Path, including;

  Underground Cable,
  Transformer CT4, and
  Standby Busses.

* Overhead Emergency Power Path, including;

  Keowee Main Step-Up Transformer,
  Associated Transmission and 230 KV Switchyard Components (e.g., transmission lines and power circuit breakers),
  230KV Switchyard Yellow Bus,
  230 KV Switchyard 125 VDC Power System, and
  Unit start-up transformers (CT1, CT2, and CT3).

* Unit Main Feeder Busses

* 4160 VAC Safety Auxiliary Power System
* 600/208 VAC Safety Auxiliary Power System
* 120 VAC Vital I&C Power System
* 125 VDC Vital I&C Power System

6. Reactor Protective System

   Note: The Reactor Protection System (RPS) is not covered by the equipment categories identified in FSAR Section 3.1.1. However, the RPS was listed in Section 1.4.1 of the PSAR and subsequently in FSAR Appendix 1B. The RPS is required for LBLOCA/LOOP mitigation and has always been QA-1. Therefore, Duke believes that it warrants inclusion into the category of "original QA-1 licensing basis".

Oconee QA-1 SSCs Added to the Original Licensing Basis

Any SSC committed to the NRC as being classified Oconee QA-1 per any correspondence subsequent to the original Oconee QA-1 licensing basis will be identified in this section. As was discussed at the February 6, 1995 management meeting with the NRC, this list of additional Oconee QA-1 SSCs will be developed through the Oconee Safety-Related Designation Clarification (OSRD) Project. This list of additional Oconee QA-1 SSCs is scheduled to be completed by July 10, 1995. Upon completion of this list, a supplement to Attachment 3 will be submitted to the NRC.

Some examples are:

Duke Class F portions of Main Steam Piping,
Duke Class F portions of Emergency Feedwater Piping and components,
Portions of Low Pressure Service Water System serving the following items:
- **High Pressure Injection Pump motor bearing coolers**
- **Motor Driven Emergency Feedwater Pump motor air coolers**
- **Turbine Driven Emergency Feedwater Pump cooling water jacket,**

Reactor Vessel Level Indication System,
Portions of the Condensor Circulating Water System,
Regulatory Guide 1.97 Instrumentation,
Standby Shutdown Facility,
Post LOCA Hydrogen Control Equipment.

ATTACHMENT 4
OCONEE LICENSING POSITION ON NON QA-1 SSCs WHICH ARE USED TO MITIGATE ACCIDENTS

It is clear in the Oconee licensing basis that there are some non QA-1 SSCs at Oconee for which credit is taken to mitigate accidents. Duke believes these SSCs warrant coverage under an augmented quality assurance program. These SSCs fall outside the licensed quality assurance program for Oconee Nuclear Station as delineated in Attachment 3. Therefore, Duke has proposed voluntary application of selected 10CFR50 Appendix B Criteria to these SSCs.

A new QA classification (QA-5) is being developed such that Duke can identify those SSCs for testing and maintenance under selected Appendix B criteria without procuring the SSCs per Appendix B. The primary tasks which must be completed to establish the QA-5 program are:

1) A list of accidents/events in the Oconee licensing-basis must be established. Accidents/events not requiring safety-related functions and accidents/events which are design criteria only must be filtered from this list. The balance of these accidents/events will be the "QA-5 Accident/Event List". This list of QA-5 accidents/events is provided in Attachment 4a.

2) For each QA-5 accident/event in Attachment 4a, an accident chart will be created which will identify primary critical safety functions and primary supporting functions. Some of the equipment performing these functions might not be QA-1. If a non-QA-1 SSC performs one of these identified functions, it will be included in the QA-5 program. Attachment 4b provides a general summary and flowchart of the process which determines Oconee SSC classification.

   Note: An accident chart will also be created for LBLOCA/LOOP. SSCs from this chart will also be classified per Attachment 4b.

3) It is necessary to determine which of the 18 criteria of 10 CFR50 Appendix B will be applied to the SSC once it is identified as QA-5 and to what extent each criterion will be applied. The extent to which the QA-5 program will invoke the 18 criteria of 10 CFR50 Appendix B is under development and will be provided to the NRC in the near future.

Implementing a QA-5 program will enhance the current practices at Oconee by identifying additional SSCs which can be maintained and tested in an augmented quality program using selected 10 CFR 50 Appendix B criteria. Procurement will not be in accordance with 10 CFR 50 Appendix B. Parts will be procured "equal or better in quality" based on engineering judgement. This procurement process will be the same as the current practices for procurement of non QA-1 parts at Oconee. The use of the "equal or better in quality" philosophy for procurement requirements will maintain the as-built material condition of the applicable SSCs.

This new classification will more clearly delineate between safety-related (QA-1) and non-safety equipment. This clearer line of division will assist both Duke Power and the NRC in review and implementation of the QA-1 program at Oconee in accordance with Appendix B of 10CFR50.

Duke believes that application of the QA-5 program to non QA-1 SSCs credited for accident mitigation will provide greater assurance of equipment quality and reliability.

ATTACHMENT 4a
OCONEE QA-5 ACCIDENT/EVENT LIST

The accidents/events addressed in the Oconee licensing basis were evaluated to determine which of those accidents/events were appropriate for the application of an augmented Quality Assurance Program. The LBLOCA/LOOP is the design basis accident for Oconee's QA-1 Program. The SSCs for LBLOCA/LOOP will also be evaluated along with the QA-5 accident/event SSCs as addressed in Attachment 4b.

From the accidents/events addressed in the Oconee licensing basis, the following are those Duke has determined to be appropriate for the application of an augmented QA Program.

Accident/Event	Reference
Loss of Lake	FSAR 2.4.11.6
Loss of Intake Structure	FSAR 2.4.11.6
Tornado	FSAR 3.2, 3.3, 3.5.1.3
Loss of Control Room Habitability	FSAR 3.11.4, 6.4 3.1.11
LTOP	FSAR 5.2.3.7
Loss of Decay Heat Removal	FSAR 5.2.3.7
Loss of Offsite Power(LOOP)	FSAR 8.2, GL 88-17
Turbine Trip	FSAR 8.2.1.3, 10.3.3, 10.4.6, 10.4.7.1.5
Loss of Main Feedwater	FSAR 10.4.7
Uncompensated Operating Reactivity Changes	FSAR 15.1
Start-up Accident	FSAR 15.2
Rod Withdrawal Accident	FSAR 15.3
Moderator Dilution Accident	FSAR 15.4
Cold Water Accident	FSAR 15.5

Loss of Coolant Flow	FSAR 15.6
Control Rod Misalignment	FSAR 15.7
Loss of Electric Load	FSAR 15.8
SG Tube Rupture	FSAR 15.9
Waste Gas Tank Rupture	FSAR 15.10
Fuel Handling Accident (To include Spent Fuel Pool Accidents)	FSAR 15.11, 9.1.2
Rod Ejection	FSAR 15.12
Steam Line Break (To include EQ Response Containment Cooling following MSLB)	FSAR 15.13
SB LOCA (To include EQ Response Containment Cooling following LOCA)	FSAR 15.14
Maximum Hypothetical Accident	FSAR 15.15
Post Accident Hydrogen Control	FSAR 15.16

The remaining Oconee accidents/events identified in the Oconee licensing basis are not included in the augumented QA Program. The basis for excluding these accidents is described below:

The NRC has specifically approved the use of non safety-related SSCs to mitigate the following accidents/events:

Accident/Event	Reference(s)
Loss of Instrument Air	DPC Resp to GL 88-14 dtd 5/8/89, NRC ltr dtd 1/10/92, FSAR 3.1.26
Loss of All AC Power (Station Blackout)	DPC SBO Rule Resp in ltrs dtd 4/17/89 and 4/4/90, NRC SER dtd 3/10/92
ATWS	DPC ATWS Mods ltr dtd 8/80/89, NRC SER on ATWS Mods dtd 11/29/89

Fire (App. R)	DPC Resp to NRC IR 77-9 in ltrs dtd 8/4/77 and 8/29/77, NRC Resp to DPC Resp in ltr dtd 10/21/77

The following events are considered in the design of SSCs. In general, SSCs are designed to withstand these events and still perform their intended safety function. Thus, these events serve as design criteria which provide a high level of confidence that the equipment needed to safely shut down the plant will remain functional. It should be noted that transient analyses are not typically performed for these events. Thus, the focus is on preventing versus mitigating a transient.

Accident/Event	Reference(s)
Earthquake	FSAR 3.1.2, 3.2.1 3.2.2, 3.9.2, 3.9.3
Snow and Ice	FSAR 3.1.2, 3.8.1.3.5
Ground Water and External Flood	FSAR 2.4.2, 3.1.2, 3.4
Sabotage	FSAR 13.6
Wind and Hurricane	FSAR 3.1.2, 3.2.1.1.1 3.3
Pipe Rupture	FSAR 3.6, ONS Pipe Break Report to NRC dtd 4/25/73
Turbine Missile	FSAR 3.1.40, 3.5.1.2
Missile Generated Inside Containment	FSAR 3.1.40, 3.5.1.1
High Energy Line Break Outside Containment	ONS Pipe Break Report to NRC dtd 4/25/73
Internal Flood	FSAR 3.4, FSAR Supp 13, DPC ltr dtd 4/21/77
The SG Overfill/Dryout	Generic Letter 89-19, Response to USI A47.
Pressurized Thermal Shock (PTS)	FSAR 5.2.3.3.6

ATTACHMENT 4b
QA-5 CRITERIA

The following is the process to be used to classify "QA-1 Accident" and "QA-5 Accident/Event" SSCs. A flowchart is provided to illustrate this process.

1. If the SSC is currently classified as QA-1, this classification will be assumed correct and no further action is needed.

2. If the SSC is not classified as QA-1 it will be reviewed to determine if it is a component of the essential systems listed in FSAR Section 3.1.1 as described in Attachment 3 of this letter or is a component of a system performing a support function needed for any of the Essential Systems to function.

3. If the non QA-1 SSC is not a component of the essential systems listed in FSAR Section 3.1.1 or of a system performing a support function needed for any of the essential systems, it will be reviewed to determine if the component has been committed by Duke to be classified as QA-1.

4. If the non QA-1 SSC is a component of the essential systems or performs a support function for any of the essential systems it will be reviewed to determine if it is needed to mitigate a LBLOCA/LOOP accident.

5. If the SSC is needed to mitigate a LBLOCA/LOOP a PIP will be generated to identify this situation. The PIP process will be used to evaluate the operability and reportability of this condition. The PIP process will also be used to identify the necessary actions to resolve this discrepancy and have it included in Oconee's QA-1 Program.

6. If the SSC is not used to mitigate a LBLOCA/LOOP it will be reviewed to determine if the component has been committed by Duke to be classified as QA-1.

7. If the SSC has been previously committed by Duke to be QA-1, but is not classified as QA-1, a PIP will be written. The process described in item 5 will be followed and the item will either be included into Oconee's QA-1 Program or a separate Duke submittal will be made to show the previous commitment was unnecessary.

8. If the SSC has not been committed to be QA-1 it will be reviewed to determine if it performs a direct or support accident mitigation function.

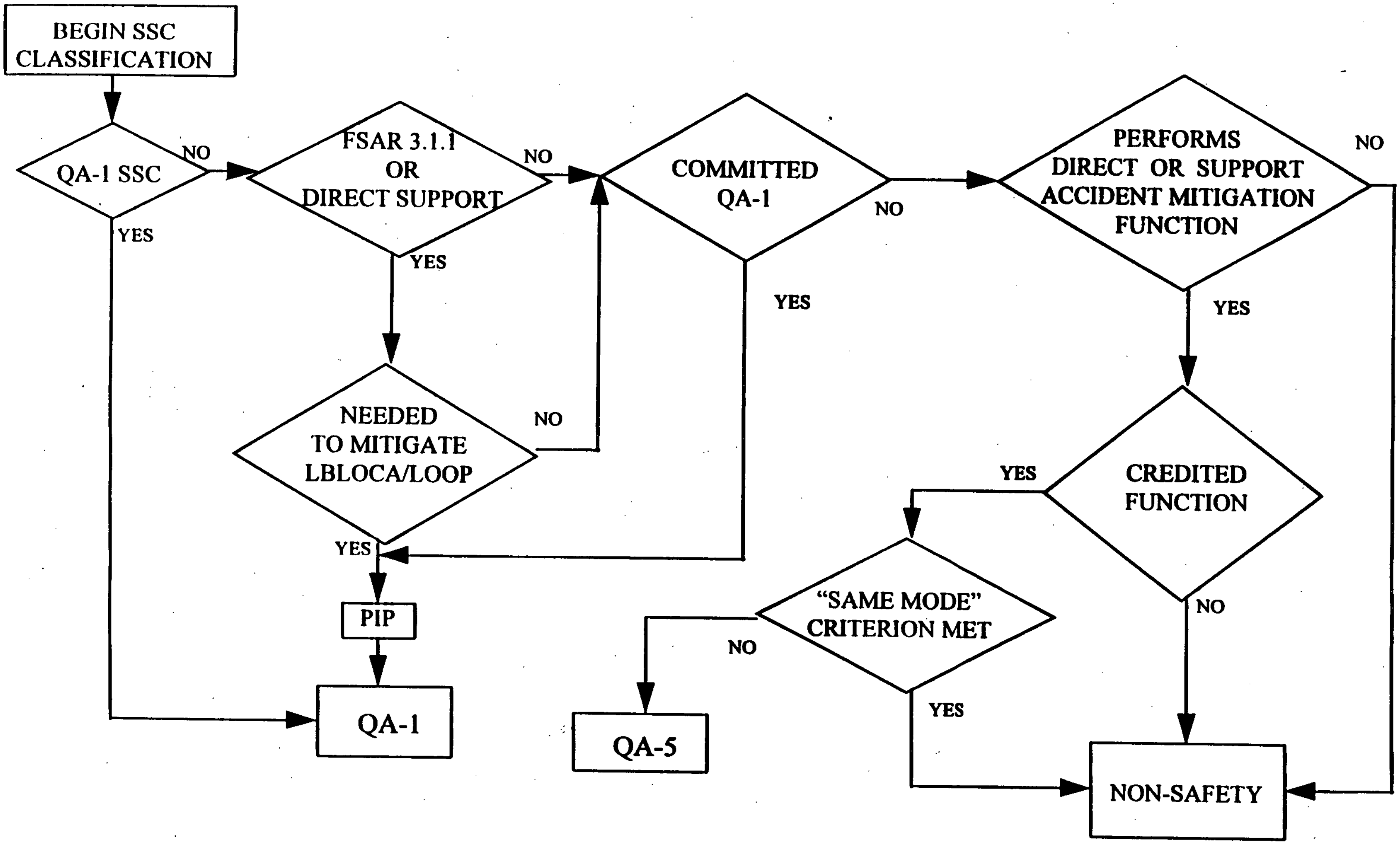
9. If the SSC performs a direct or support accident mitigation function it will be reviewed to determine if it is taken

credit for in the accident analysis, calculation, or licensing bases as the required success path to fulfill this function.

10. If the SSC serves as the primary success path to fulfill the function it will be reviewed to determine if the NRC has previously approved the use of non-safety SSCs to perform this function.

11. If the SSC routinely operates during normal plant operation in the same mode that it would function during an accident, as determined by engineering based on available design documentation, then the SSC will be classified as non-safety-related. This engineering determination must conclude that the limiting operational and design parameters under normal operating conditions bound the limiting operational and design parameters under accident conditions. If the SSC does not operate during plant operation in the same mode that it would function during an accident, as determined by engineering based on available design documentation, then the SSC will be classified as QA-5.

SSC CLASSIFICATION FLOWCHART



ATTACHMENT 5
DEFINITIONS

Augmented Quality Assurance Program - A quality assurance program voluntarily applied to selected SSCs.

Oconee QA-1 - SSCs at Oconee Nuclear Site which fall under the 10CFR50 Appendix B Quality Assurance requirements.

QA-5 - The Augmented Quality Assurance Program that Duke is proposing to apply to SSCs (that do not fall under the scope of the Oconee QA-1 program) which are required for mitigation of QA-5 accidents/events. This program will implement portions of 10CFR50 Appendix B in part to SSCs which are classified as QA-5.

QA-5 Accidents/Events - Those accidents/events whose mitigating SSCs should be considered for the QA-5 program if they are not already QA-1. QA-5 accidents/events for Oconee are contained in Attachment 4a.

Safety-Related - The definition for this term has two aspects: 1) scope of application and 2) compliance applicability.

> The scope of application is to all SSCs required to mitigate consequences of accidents, maintain RCS integrity, or achieve safe shutdown, as defined by the NRC. For Oconee, this is simply all SSCs denoted as Oconee QA-1.
>
> The compliance applicability pertains to what regulations must be applied to the SSC once it is labelled as safety-related. If an SSC is labelled safety-related, then 10CFR 50 Appendix B applies in full to that SSC.

SSCs - Structures, systems, or components. For the purpose of this submittal, any item subject to classification under a quality assurance program.

Attachment 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 3, 1995

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

SUBJECT: GENERIC LETTER 83-28 SUPPLEMENTAL RESPONSE - OCONEE UNITS 1, 2, AND 3 (TAC NOS. M92023, M92024, AND M92025)

Dear Mr. Hampton:

By letter dated April 12, 1995, you requested NRC review and approval of a supplemental revised response to Subpart 1 of Section 2.2.1 of Generic Letter (GL) 83-28, Equipment Classification Program for all Safety-Related Components. In addition, you provided a clarification of what Duke Power Company understands to be the basis for the original licensing criteria for designating systems, structures and components (SSCs) as safety-related.

We have completed our review of your submittal. Our safety evaluation is enclosed. We find your revised response to GL 83-28, Section 2.2.1, Subpart 1, to be acceptable. In addition, we found no basis to contradict your understanding of the basis for the original licensing criteria for designating SSCs as safety-related. If you have questions regarding this matter, contact me at (301) 415-1495.

Sincerely,

Leonard A. Wiens, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270
and 50-287

Enclosure:
Safety Evaluation

cc w/enclosure:
See next page

~~9508090103~~ 2pp.

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Duke Power Company

Oconee Nuclear Station

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPLEMENTAL RESPONSE TO SUBPART 1 OF SECTION 2.2.1 OF GENERIC LETTER 83-28

GENERAL CRITERIA FOR CLASSIFYING QA-1 FOR STRUCTURES, SYSTEMS, AND COMPONENTS

DUKE POWER COMPANY

OCONEE NUCLEAR POWER STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated April 12, 1995 (Ref. 1), Duke Power Company (DPC) requested NRC review and approval for a supplemental response to Subpart 1 of Section 2.2.1 of Generic Letter (GL) 83-28 on structures, systems and components (SSC) for the Oconee Nuclear Station (ONS) Units 1, 2, and 3. The letter provided the following attachments:

Attachment 1:	Detailed description of the history of Oconee's QA-1 licensing basis,
Attachment 2:	Oconee licensing position on GL 83-28,
Attachment 3:	Supplemental response to Subpart 1 of Section 2.2.1 of GL 83-28 and the general criteria for classifying QA-1 SSCs,
Attachment 4:	Oconee's position on Non-Oconee QA-1 SSCs which are used to mitigate accidents,
Attachment 4a:	Oconee QA-5 accident/event list,
Attachment 4b:	QA-5 criteria, and
Attachment 5:	Defined terms used in the document.

DPC requested the review and approval of their supplemental response to Subpart 1 of Section 2.2.1 of GL 83-28 provided in Attachment 3.

2.0 BACKGROUND

The following statements are provided based on information submitted by DPC to explain the licensing basis of the ONS. The NRC evaluations of these items is in Section 3.0.

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Attachment 1 of Reference 1, presented DPC's historical development of the Oconee QA (QA-1) program. As provided in Attachment 1, DPC indicated that before the issuance of NRC guidance documents, there was very little information available regarding the definition of safety-related SSCs. Consistent with available Atomic Energy Commission (AEC) guidance at the time, the nuclear industry developed its own definition of safety-related. The AEC guidance consisted primarily of correspondence between the AEC and the utilities building nuclear power plants. The staff determined that although formal guidance such as Regulatory Guides was limited during the early Oconee construction era, some safety guides had been issued. DPC indicated that the design and construction of early commercial nuclear power plants was performed using available mechanical, civil, and electrical codes. SSCs were classified at ONS consistent with this code-based approach.

DPC indicated that at the time the Oconee plant was being designed and constructed, DPC defined the term safety-related as 1) nuclear piping per USAS B31.7 nuclear piping code (February 1968 and Addenda), or 2) specifically identified large break loss-of-coolant accident (LBLOCA) mitigation (e.g., Low Pressure Service Water) non-nuclear, seismically qualified piping per USAS B 31.1 piping code.

In a phone call between NRC and DPC on July 17, 1995, to get clarification on item 2 of the above definition of safety-related, the following was explained by DPC. In addition to piping designed to nuclear safety-related per USAS B31.7, there was some non-nuclear piping, designed to USAS B31.1, which was required to mitigate a LBLOCA and was designated as safety-related and designed to be seismically qualified.

DPC indicated that the code-derived definition of safety-related does not correlate well with the functionally based definition eventually developed by the NRC. Oconee Electrical Design relied heavily on Mechanical Design during the Oconee construction to provide information on whether a component required IEEE Class 1E power. This appears to be why several B31.1 systems not required for LBLOCA/Loss Of Offsite Power (LOOP) mitigation (such as Main Steam) had non QA-1 (10 CFR Part 50 Appendix B) power supplies to valves. The USAS B31.7 nuclear piping code used by ONS was in effect for only a short time period. It was superseded for later nuclear plant designs by the ASME Section III Code for nuclear pressure vessel and piping design. There has been difficulty between DPC and the NRC in maintaining a consistent understanding of the definition of "safety-related" (for Oconee).

The system (piping) classifications for the ONS were conceived and established in early 1968, materials were procured in 1968 and erection requirements were established and used in 1969. The AEC piping classification system of A through D (eventually in Regulatory Guide 1.26) did not emerge until after Oconee was well underway and has never been imposed on the Oconee design.

The Preliminary Safety Analysis Report (PSAR) for ONS in Section 1.4.1, related to quality standards, indicated that: "... the integrity of SSCs essential to accident prevention and to mitigation of accident consequences has been considered in the design evaluations. These SSCs are:

1) Fuel Assemblies,
2) Reactor Coolant System,
3) Reactor instrumentation, controls, and protective systems,
4) Engineered safeguards systems,
5) Radioactive materials handling systems,
6) Reactor building, and
7) Electric power sources."

DPC indicated that while these systems do not encompass all presently postulated accidents, the list of systems does envelop the majority of SSCs required for mitigation of the postulated LBLOCA/LOOP accident. There is other equipment on the list whose integrity is considered necessary to prevent offsite dose to the public (i.e., B31.7 code-designed nuclear piping and radioactive materials handling systems). The AEC and ACRS correspondence throughout the pre-construction permit period indicates that the regulatory focus was on LBLOCA mitigation.

The original Final Safety Analysis Report (FSAR) for ONS in Appendix 1A, related to quality standards, indicates that: "the integrity of SSCs essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluation. These SSCs are:

1) Reactor Coolant System,
2) Reactor Vessel Internals,
3) Reactor Building,
4) Engineered Safeguards System, and
5) Emergency Electric Power Sources."

DPC indicated that this list of SSCs comprised the original ONS QA-1 SSCs to which 10 CFR Part 50 Appendix B would be applicable. These systems mitigate a LBLOCA/LOOP.

There were many SSCs described that were seismically designed which did not fall under the scope of the original ONS QA-1 program. FSAR Section 3.2.2 gives some examples such as the CCW intake structure, CCW pumps, upper surge tanks, and emergency feedwater pumps. It is clear that some seismically designed single failure proof systems were not classified as QA-1 when ONS received its license. However, all SSCs that fell within the original ONS QA-1 program met both single-failure and seismic design criteria.

The FSAR Appendix 1B list of SSCs is consistent with the Duke philosophy regarding application of the quality assurance program at ONS during construction licensing. The SSCs provided on this list, with few exceptions, are items which were:

1) Necessary to mitigate a LBLOCA/LOOP design basis accident, or
2) Pressure boundary to prevent release of radioactive fluids which if released could present a danger to the public (as determined by dose levels), or,
3) Electrical/Instrumentation items designed per draft IEEE 279 Class 1E.

DPC indicated that there are some examples of SSCs that did not appear on the Appendix 1B list that are required for mitigating a LBLOCA/LOOP, such as portions of the Condenser Circulation Water (CCW) System. However, it was recognized that some features of these non-nuclear, USAS B31.1.0 systems permitted their exclusion from the quality assurance program. These features were: 1) redundancy and diversity, 2) passive mitigation functions, 3) seismic design, and 4) constant use of these systems in normal operation of the plant.

Attachment 2 provides comments on the DPC response to GL 83-28, "Required Actions Based on Generic Implication of Salem ATWS Events." Section 2.2.1 of this letter entitled "Equipment Classification and Vendor Interface," required licensees to submit, for NRC review, a "... description of their programs for safety-related equipment classification." Subpart 1 of Section 2.2.1 required a discussion of "the criteria for identifying components as safety-related within systems currently classified as safety-related". DPC indicated that their response was not intended to be construed as a reclassification of the entire scope of the ONS SSCs to the functional definition of safety-related provided in GL 83-28.

Attachment 3 provides comments on DPC's response to Subpart 1 of Section 2.2.1 of GL 83-28. It provides the general criteria for identifying components as QA-1 at ONS and supersedes earlier submittals. The safety evaluation by the staff is concerned with Attachment 3. The other attachments are to provide supplementary and background information.

ONS is a relatively early plant design where the AEC/NRC regulatory guidelines were not in effect as for later plants, and is therefore more unique in its design guidelines. Enhancements are planned (Ref. 2) for Oconee to (1) have a clear "functional" dividing line between "Oconee QA-1" and "Non-Safety SSCs which perform a function Important to Safety", and (2) have application of a graded QA program to these "Non-Safety SSCs which perform a function Important to Safety."

Attachment 4 indicates that there are some non QA-1 SSCs at Oconee for which credit is taken to mitigate accidents. A new QA classification (QA-5) is being developed such that DPC can identify those SSCs for testing and maintenance under selected Appendix B criteria without procuring the SSCs per Appendix B.

Attachment 5 presents DPC definitions that are used in the ONS quality assurance program, some of which are presented below:

> Augmented Quality Assurance Program - A quality assurance program voluntarily applied to selected SSCs.
>
> Oconee QA-1 - SSCs at Oconee Nuclear Site which fall under the 10 CFR 50 Appendix B Quality Assurance requirements.

QA-5 - The Augmented Quality Assurance Program that Duke is proposing to apply to SSCs (that do not fall under the scope of the Oconee QA-1 program) which are required for mitigation of QA-5 accidents/events. This program will implement portions of 10 CFR Part 50 Appendix B in part to SSCs which are classified as QA-5.

QA-5 Accidents/Events - Those accidents/events whose mitigating SSCs should be considered for the QA-5 program if they are not already QA-1. QA-5 accidents/events for Oconee are contained in Attachment 4a.

Safety-Related - The definition for this term has two aspects: 1) scope of application, and 2) compliance applicability.

> The scope of application is to all SSCs required to mitigate consequences of accidents, maintain RCS integrity, or achieve safe shutdown, as defined by the NRC. For Oconee, this is simply all SSCs denoted as Oconee QA-1.
>
> The compliance applicability pertains to what regulations must be applied to the SSC once it is labelled as safety-related. If an SSC is labelled safety-related, then 10 CFR 50 Appendix B applies in full to that SSC.

In a phone call between NRC and DPC on July 17, 1955 to get clarification on "the scope of application" in the above DPC definition of safety-related, the following was explained by DPC. The NRC definition is taken to be "all SSCs required to mitigate consequences of accidents, maintain RCS integrity, or achieve safe shutdown." The DPC definition for Oconee is taken to be "all SSCs denoted as Ocone QA-1." This was explained to mean that these SSCs are in accordance with FSAR Section 3.1.1, Criterion 1- Quality Standards (Category A), plus all additional SSCs that DPC has committed to be QA-1 for Oconee.

3.0 EVALUATION

The letter of April 12, 1995 (Ref. 1) from DPC presented the attachments as identified in Section 1.0, Introduction. The main attachment for review was Attachment 3 regarding the supplemental response to Subpart 1 of Section 2.2.1 of GL 83-28 and the general criteria for classifying QA-1 SSCs. The remaining attachments provide background and supplemental information. In the same manner the staff evaluation also covers some background information, including a review of the Final Safety Analysis Report (FSAR) sections that relate to the review of Attachment 3.

A nuclear operating license is issued based on the finding that there is a reasonable assurance that the authorized activities can be conducted without endangering the health and safety of the public. As discussed above, the ONS is an early nuclear plant design whose nuclear safety guideline requirements has some differences from current design requirements. The first formalized approach to nuclear safety was the use of the Maximum Credible Accident (MCA) which is defined as the postulated credible accident which poses a potential hazard greater than any other accident which is also considered to be credible. This was to be designed against. It was also necessary to demonstrate that a plant met the guidelines set forth in the Code of Federal Regulations, Title 10, Part 100 (10 CFR 100). Another approach that evolved was a method known as the Design Base Accident (DBA) approach, which is the approach used at Oconee. The DBA approach principally consists of explicitly identifying low frequency high consequence accidents which must be designed against.

DPC indicates in ONS FSAR Appendix 17, Quality Assurance, that the quality assurance program conforms to the applicable regulatory requirements of 10 CFR Part 50, Appendix B.

Our evaluation includes a review of ONS information in an effort to identify what DPC has specified as safety-related SSCs and in particular to identify which design basis accidents (DBAs) were used as the criteria under which the ONS was originally licensed to identify the SSCs that should be included in the ONS QA-1 program.

The material reviewed below was from information provided by DPC in their FSAR (update of December 31, 1993), and from various other references. The following background information includes a discussion on sections of the FSAR that relate to the design criteria for quality assurance, the reactor coolant system and connected systems, the Chapter 15 Accident Analyses, and Attachment 4 on non QA-1 SSCs. Attachment 3, the supplemental response to GL 83-28, is discussed in Section 3.6.

3.1 FSAR Section 3.1, Conformance with NRC General Design Criteria

In the ONS FSAR Section 3.1, Conformance with NRC General Design Criteria, DPC indicates that the principal design criteria for Oconee 1, 2, and 3 were developed in consideration of the seventy General Design Criteria for Nuclear Power Plant Construction Permits <u>proposed</u> by the AEC in a proposed rule-making published for 10 CFR Part 50 in the Federal Register of July 11, 1967. Three pertinent FSAR Sections are <u>Section 3.1.1, Criterion 1</u> - Quality Standards (Category A), <u>Section 3.1.2, Criterion 2</u> - Performance Standards (Category A) and <u>Section 3.1.44, Criterion 44</u> - Emergency Core Cooling Systems Capability (Category A). The DPC discussions for Criterion 1, 2, and 44 are presented below:

1) FSAR Section 3.1.1, Criterion 1 - Quality Standards (Category A).

For Criterion 1 DPC states that those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

The DPC discussion on Essential Systems and Components states that the integrity of systems, structures, and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. The system, structure, and components (SSC) are: a) Reactor Coolant System, b) Reactor Vessel Internals, c) Reactor Building, d) Engineered Safeguards System, and e) Emergency Electric Power Sources.

A table in FSAR Section 3.1.1 references applicable sections in the FSAR where codes, quality controls, and tests are included. Section 3.1.1 states that the Quality Assurance Program is discussed in detail in Chapter 17, "Quality Assurance".

2) FSAR Section 3.1.2, Criterion 2 - Performance Standards (Category A)

For Criterion 2 DPC states that those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and, b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

The essential systems and components for FSAR Section 3.1.2 are the same SSCs that were listed above for FSAR Section 3.1.1.

Seismic forces (earthquakes) and other natural phenomena are considered. There is no indication in Section 3.1.2 of these forces being considered in combination with a loss of offsite power. DPC stated in Section 2.0, the DBA is a LBLOCA/LOOP combination.

3) FSAR Section 3.1.44, Criterion 44 - Emergency Core Cooling Systems Capability (Category A)

For Criterion 44, DPC states that at least two Emergency Core Cooling Systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each Emergency Core Cooling System (ECCS) and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each ECCS shall be evaluated conservatively in each area of uncertainty. The system shall not share active components and shall not share other features or components unless it can be demonstrated that: a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

The discussion section for Criterion 44 states that the emergency core cooling is provided by pumped injection and pressurized core flooding tanks. Pumped injection is subdivided in such a way that there are two separate and independent strings, each including both high pressure and low pressure coolant injection, and each capable of providing 100 percent of the necessary core injection with the core flooding tanks. There is no sharing of active components between the two subsystems in the post-accident operating mode. The core flooding tanks are passive components which are needed for only a short period of time after the accident, thereby assuring 100 percent availability when needed. This equipment prevents clad melting for the entire spectrum of Reactor Coolant System (RCS) failure ranging from the smallest leak to the complete severance of the largest reactor coolant pipe (Section 15.14, "Loss of Coolant Accidents" on page 15-48).

Although Criterion 44 mentions RCS failure in sizes from small leaks to the complete severance of the largest reactor coolant pipe, there is no mention of doing analyses for a spectrum of sizes to find the limiting size for large and small pipe breaks. However, Chapter 15 of the FSAR states that a spectrum of break sizes were used in the analyses to determine the worst break.

3.2 FSAR Section 3.2 - Classification of Structures, Components, and Systems

Section 3.2 provides classification of the SSCs as described below.

1) Section 3.2.1 addresses Seismic Classification

Section 3.2.1.2 - Components and Systems, states that capability is provided to shutdown safely all three units in the event of a maximum hypothetical earthquake. Equipment and portions of systems that can withstand the maximum hypothetical earthquake are identified in Section 3.2.2.

2) Section 3.2.2 addresses System Quality Group Classification

This section defines the design criteria used with respect to the loss-of-coolant accident (LOCA), natural phenomena, and also describes the division of components and piping into classifications related to components or systems. It is stated that a maximum hypothetical earthquake will not result in a LOCA and the simultaneous occurrence of these events will not result in loss of function to vital safety related components or systems.

The following design objectives result from consideration of the design criteria:

Loss-of-Coolant Accident

Capability is provided to assure necessary protective actions, including reactor trip and operation of the ECCS, to protect the public during LOCA, even in the event of a simultaneously occurring maximum hypothetical earthquake. (It is noted that for accident analysis it is not assumed that a LOCA and seismic event occur simultaneously).

The following equipment and portions of systems are stated to be able to withstand the maximum hypothetical earthquake:

a. Reactor Coolant System,

b. Borated water storage tank and piping to high pressure and low pressure injection pumps and Reactor Building spray pumps,

c. HP injection pumps and piping to Reactor Coolant System,

d. LP injection pumps, LP injection coolers and piping to both Reactor Coolant and Reactor Building spray pumps,

e. Core flood tanks and piping to spray header, and the spray headers,

f. Reactor Building spray pumps, piping to spray headers, and the spray header,

g. Reactor Building Coolers,

h. Low pressure service water (LPSW) pumps, LPSW piping to LP injection coolers and Reactor Building coolers and LPSW piping from these coolers to the condenser circulating water (CCW) discharge,

i. CCW intake structure, CCW pumps, pump motors, CCW intake piping to the LPSW pumps, also through the condenser and emergency CCW discharge piping and CCW and CCW discharge piping,

j. Upper surge tanks and piping to the emergency feedwater pump,

k. Emergency feedwater pump and turbine and auxiliary feedwater piping to the steam generators,

l. Main steam lines to and including turbine stop valves. Turbine bypass system up through Main Steam System isolation valves, and steam supply lines to the emergency feedwater pump turbine,

m. Penetration Room Ventilation System,

n. Reactor Building penetrations and piping through isolation valves, and

o. Electric power for the above.

Section 3.2.2.1 - System Classifications

This section states that plant piping systems or portions of systems are classified according to their function in meeting design objectives. The systems are further segregated depending on the nature of the contained fluid. For those systems which normally contain radioactive fluids or gas, the Nuclear Power Piping Code, USAS B31.7 and Power Piping Code USAS B31.1 are used to define material, fabrication, and inspection requirements. Fabrication and erection of piping, fittings, and valves are in accordance with the rules for their respective classes. Welds between classes of systems (Class I to II, I to III, or II to III) are performed and inspected in accordance with the rules for the higher class.

Sections 3.2.2.2, 3.2.2.3, and 3.2.2.4 provide further guidance on system piping, system valve, and system component classification respectively.

3.3 FSAR Section 5, Reactor Coolant Systems and Connected Systems

The Reactor Coolant System (RCS) consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnecting piping. The system is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator.

The RCS is analyzed for a maximum hypothetical earthquake to determine that resultant stresses do not jeopardize the safe shutdown of the RCS and removal of decay heat.

There is reliance on interconnected systems. The principal heat removal system interconnected with the RCS is the Steam and Power Conversion System. This system provides capability to remove reactor decay heat for the hypothetical case where all station power is lost. Under these conditions decay heat removal from the reactor core is provided by the natural circulation characteristics of the RCS which provides a method of energy removal from the core with transfer of energy to the secondary system through the steam generators. The turbine driven emergency feedwater pump supplies feedwater to the steam generators. Cooling water flow to the condenser is provided by the emergency discharge line which discharges to the tailrace of the Keowee Dam. The analysis for this unlikely condition of total loss of station electric power is presented in FSAR Section 8.3.2.2.4, "Station Blackout Analysis." Should the condenser not be available to receive the steam generated by decay heat, which is unlikely in view of emergency discharge line flow, the water stored in the feedwater system can be pumped to the steam generators and the resultant steam vented to atmosphere to provide required cooling.

The natural circulation cooldown mode of operation is not expected to be undertaken at ONS except for SBLOCA events which do not allow continued operation of or restart of reactor coolant pumps. In all other situations, procedures recommend that hot shutdown be maintained until those systems required for forced circulation are put back into service.

3.4 FSAR Section 15, Accident Analysis

Chapter 15 of the FSAR details the expected response of the plant to the spectrum of transients and accidents which constitute the design basis events. The analyses presented show that the plant response is either inherently limited by the characteristics of the system or is terminated by the normal function of the Reactor Protective System (RPS) and the Engineered Safeguards Protective System (ESPS).

Section 15.13, Steam Line Break Accident, Section 15.14, Loss of Coolant Accidents, and Section 15.15, Maximum Hypothetical Accident, were reviewed to examine the systems that were used.

1) Section 15.13, Steam Line Break Accident

The introduction for this section states that the worst case overcooling accident is the double-ended rupture of a 34 inch main steam line from rated power conditions with offsite power available. The analysis was done with (1) integrated control system (ICS) and operator action and (2) without the ICS and operator action, which is the worse case. For the worse case the High Pressure Injection System (HPIS) actuates at 30 seconds, core flood tanks initiate injection at approximately 60 seconds, and the Low Pressure Injection System (LPIS) actuates at 100 seconds. The return to power peaks at 8 percent rated power and returns to subcritical at 166 seconds. The DPC conclusion is that the results show that the unit can successfully mitigate the transient without taking credit for ICS or operator action, although normal ICS and operator action will significantly moderate the plant response. The peak return to power is not great enough to cause fuel damage.

2) Section 15.14, Loss of Coolant Accidents

The introduction for this section states that a failure of the RCS pressure boundary will result in a loss of primary coolant inventory and the potential for the core to uncover. These hypothetical failures are considered to occur in all piping and components up to an including a double-ended rupture of the largest pipe in the system. If the core is not rapidly reflooded and long term heat removal established, decay heat will cause the fuel cladding to fail and release the fission product inventory. The ECCS is designed to deliver sufficient coolant to provide the necessary core decay heat removal for credible LOCA's.

This section provides information on break spectrum analysis for both large and small break LOCA. The conclusion for this section stated that a complete spectrum of LOCA's have been conservatively analyzed with the B&W evaluation model which conforms to 10 CFR 50 Appendix K. The results of these analyses meet the acceptance criteria of 10 CFR 50.46. The Reactor Building and subcompartment pressure response analyses show that the structural design limits were not exceeded. The off-site environmental consequences are within the dose limits of 10 CFR Part 100. Therefore, the consequences of all design basis LOCA's have been shown to be acceptable.

The list of equipment used for protective actions for a LOCA were not presented in Section 15.14, but are given in FSAR Section 3.2.2.

3) Section 15.15, Maximum Hypothetical Accident

The introduction to this section states that the analyses in the preceding sections have demonstrated that even in the event of a LOCA accident, no significant core melting will occur. However, to demonstrate in a more conservative manner that the operation of a nuclear power plant at the proposed site does not present any undue hazard to the general public, a maximum hypothetical accident (MHA) involving a gross release of fission products is evaluated. No mechanism whereby such a release occurs is postulated, since this would require a multitude of failures in the engineered safeguards which are provided to prevent such an occurrence.

No systems are mentioned in this maximum hypothetical accident. However, this section indicates that the preceding sections have demonstrated that no significant core melting will occur from any of the other preceding accidents analyzed.

3.5 Attachment 4, Oconee Licensing Position On Non QA-1 SSCs Which Are Used to Mitigate Accidents

In Attachment 4 DPC states that there are some non QA-1 SSCs at Oconee for which credit is taken to mitigate accidents. These are not in the ONS QA-1 licensed quality assurance program for ONS. DPC has proposed voluntary application of selected 10 CFR 50 Appendix B criteria to these SSCs. In this new QA classification (QA-5) DPC indicates that they will identify those SSCs for which testing and maintenance will be performed under selected Appendix B criteria. However, the SSCs will not be procured per Appendix B requirements. Parts will be procured "equal or better in quality" based on engineering judgement. The purpose of this new QA-5 classification is to more clearly delineate between safety-related (QA-1) and non-safety equipment.

DPC presented the three primary tasks which need to be completed to establish the QA-5 program:

1) Prepare a list of accident/events in the Oconee licensing-basis and filter out the QA-5 accidents (presented in Attachment 4a of Ref.1),

2) For each QA-5 accident/event in Attachment 4a, an accident chart will be created which will identify primary critical safety functions and primary supporting functions. Some of the equipment performing these functions might not be QA-1. If a non QA-1 SSC performs one of these identified functions, it will be included in the QA-5 program. Attachment 4b was presented in Reference 1 which provides a general summary and flow chart of the process which determines Oconee SSC classification. An accident chart will also be created for LBLOCA/LOOP. SSCs from this chart will also be classified per Attachment 4b.

3) DPC will then determine which of the 18 criteria of 10 CFR 50 Appendix B will be applied to the SSC once it is identified as QA-5 and to what extent each criterion will be applied.

3.6 Attachment 3, Supplemental Response to Subpart 1 of Section 2.2.1 of GL 83-28 General Criteria for classifying QA-1 SSCs

Section 2.2.1 of GL 83-28 stipulates that licensees and applicants shall describe in detail their program for classifying all safety-related components as safety related on plant documents and in information handling systems that are used to control plant activities that may affect these components. Specifically, Subpart 1 requested the licensee's criteria for identifying components as safety-related within systems currently classified as safety-related. This was not to be interpreted to require changes in safety classification at the systems level.

GL 83-28 (Ref. 3) defined a component as safety-related if it is required to assure: (a) the integrity of the reactor coolant system pressure boundary, (b) the capability to achieve and maintain a safe shutdown, or (c) the capability to prevent or to mitigate the consequences of an accident which could result in potential offsite exposures.

The licensee's original response to GL 83-28 Section 2.2.1, Subpart 1 was reviewed and found to be acceptable as documented in the NRC's Safety Evaluation Report for Oconee Nuclear Station, Unit 1, 2, and 3 dated November 4, 1987 (Ref. 4).

Attachment 3, supplemental response to Subpart 1 of Section 2.2.1 of GL 83-28, supersedes the previous Oconee submittals related to this issue.

DPC presented the general criteria used to determine if a SSC is QA-1. This is divided into two categories:

1) First category - provides general QA-1 criteria based on the original licensing basis of ONS, and

2) Second category - provides general criteria for SSCs that were added to the QA-1 licensing basis after issuance of the original operating licenses for ONS.

DPC plans to revise Section 3.1.1 of the FSAR to include the general criteria provided in Attachment 3 following NRC review and approval. The first and second categories are presented below.

1) First Category, Original Oconee QA-1 Licensing Basis (FSAR Section 3.1.1)

This first category includes the integrity of SSCs essential to prevention and mitigation of the Large Break LOCA coincident with Loss Of Offsite Power for the following five SSCs: 1) Reactor Coolant System, 2) Reactor Vessel Internals, 3) Reactor Building, 4) Engineered Safeguards System, and 5) Emergency Electric Power Sources. In addition, 6) Reactor Protective System, another system not addressed in FSAR Section 3.1.1, was provided by DPC.

The DPC presentation for the six SSCs identified above is provided below.

1. Reactor Coolant System

From a quality assurance perspective, the Reactor Coolant System consists of all connecting piping, valve bodies, pump casings, heat exchangers, or vessels out to and including the first isolation valve. The integrity of the pressure boundary of the connecting piping, valve bodies, pump casings, heat exchangers, or vessels is the function which determines applicability of the quality assurance program.

2. Reactor Vessel Internals

The Reactor Vessel Internals consist of the plenum assembly and the core support assembly. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, thermal shield, and surveillance holder tubes. The plenum assembly consists of the upper grid plate, the control rod guide assemblies, and a turnaround baffle for the outlet flow.

Reactor vessel internals do not include fuel assemblies, control rod assemblies, surveillance specimen assemblies, or incore instrumentation.

3. Reactor Building

The Reactor Building consists of the following:

- The structure which consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a massive reinforced concreted foundation slab.

- The entire interior surface of the structure (a steel plate liner).

- Welded steel penetrations through which numerous mechanical and electrical systems pass into the Reactor Building.

- Access openings to the Reactor Building.

4. Engineered Safeguards System

The Engineered Safeguards System consists of structure, systems, or components necessary to:

- Provide emergency cooling to assure structural integrity of the core:

  High Pressure Injection System
  Low Pressure Injection System
  Core Flooding System

- Maintain the integrity of the Reactor Building

  Reactor Building Spray System
  Reactor Building Cooling System
  Reactor Building Isolation System (this includes all piping penetration isolation paths)

- In addition, support systems necessary to ensure that the above systems can perform their intended safety functions are considered QA-1. These systems are:

  Low Pressure Service Water portions necessary to supply cooling water to:

    Reactor Building Cooling Units
    Decay Heat Removal Coolers

  Keowee emergency start, load shed, and emergency power switching logic

  Analog and Digital ES Channels and DC Power to support operability of these channels

5. Emergency Electric Power Sources

The following power sources and distribution systems serve QA-1 functions:

- Keowee Hydroelectric Units 1 and 2, including:

  Keowee Hydro-Generator and Emergency Start Circuits,
  Keowee 600/208/120 VAC Auxiliary Power System, and
  Keowee 125 VDC Power System.

The following mechanical Keowee SSCs:

- Governor Oil System
- Governor Air System
- Guide Bearing Oil System
- Turbine Sump System
- Cooling Water System

- Underground Emergency Power Path, including:

  Underground cable,
  Transformer CT4, and
  Standby Busses.

- Overhead Emergency Power Path, including:

  Keowee Main Step-Up Transformer,
  Associated Tranmission and 230KV Swith\chyard Components (e.g., transmission lines and power circuit breakers),
  230 KV Switchyard Yellow Bus,
  230 KV Switchyard 125 VDC Power System, and
  Unit Start-up Transformers (CT1, CT2, and CT3).

- Unit Main Feeder Busses

- 4160 VAC Safety Auxiliary Power System

- 600/208 VAC Safety Auxiliary Power System

- 120 VAC Vital I&C Power System

- 125 VDC Vital I&C Power System

6. Reactor Protective System

The Reactor Protective System (RPS) is not covered by the equipment categories identified in FSAR Section 3.1.1. However, the RPS was listed in Section 1.41. of the PSAR and subsequently in FSAR Appendix 1B. The RPS is required for LBLOCA/LOOP mitigation and has always been QA-1. Therefore DPC believes that it warrants inclusion into the category of "original QA-1 licensing basis."

2) Second Category, Oconee QA-1 SSCs Added To The Original Licensing Basis

In this category DPC includes any SSC committed to the NRC as being classified QA-1 per any correspondence subsequent to the original Oconee QA-1 licensing basis. As discussed in a February 6, 1995 management meeting with the NRC (Ref. 2), this list of additional Oconee QA-1 SSCs will be developed through the Oconee

Safety-Related Designation Clarification (OSRD) Project. The list of additional Oconee QA-1 SSCs is scheduled to be completed by July 10, 1995. Upon completion of this list, a supplement to Attachment 3 is to be submitted to the NRC.

Some examples are:

Duke Class F portions of Main Steam Piping,
Duke Class F portions of Emergency Feedwater Piping and components,

Portion of Low Pressure Service Water System serving the following items:

- High Pressure Injection Pump motor bearing coolers
- Motor Driven Emergency Feedwater Pump motor air coolers
- Turbine Driven Emergency Feedwater Pump cooling water jacket,

Reactor Vessel Level Indication System,
Portions of the Condenser Circulation Water System,
Regulatory Guide 1.97 Instrumentation,
Standby Shutdown Facility,
Post LOCA Hydrogen Control Equipment.

Based on the staff reviews within this area, it was determined that the supplemental response to GL 83-28, Section 2.2.1, Subpart 1 contained in DPC's letter dated April 12, 1995 (Ref. 1), provides an acceptable basis for defining QA-1 equipment classification which is consistent with the original licensing basis of ONS.

3.7 Summary

In our examination of the FSAR and other references we did not find any basis to contradict the DPC assertion that the only DBA for ONS is the LBLOCA/LOOP for delineation of equipment to be defined as safety-related QA-1.

DPC provided criteria for properly selecting the systems related to the LBLOCA/LOOP DBA for the QA-1 quality category. Some additional SSCs are in the process of being added to the original licensing basis by DPC's voluntary OSRD Project. Other SSCs are being put into DPC's voluntary Augmented Quality Assurance program for upgrade. The staff agrees that these steps provide an adequate approach for safety classification. If properly implemented the augmented QA program should help ensure that SSCs important to safety will receive the appropriate operation, maintenance and testing. The augmented QA program should provide enhancement to assure that equipment important to the mitigation of accidents and transients will perform their intended function.

In our review of Attachment 3, Supplemental Response to Subpart 1 of Section 2.2.1 of GL 83-28, we found the criterion to be acceptable for SSCs of the first category. This acceptance of the proposed classification of the SSCs in the supplemental response to GL 83-28 is based on the condition that no previous SSCs that were classified as QA-1 be downgraded in classification. The evaluation of the criterion for the review of the SSCs in the second category was also found to be acceptable. DPC indicated that a list of QA-1 SSCs from the OSRD Project for the second category is scheduled for completion by July 10, 1995. Upon completion of this list, a supplement to Attachment 3 is to be submitted to NRC.

4.0 CONCLUSION

The staff has reviewed the licensee's submittal of the supplemental response to Subpart 1 of Section 2.2.1 of GL 83-28 on structures, systems and components and finds the approach to be acceptable as discussed in Section 3.0.

5.0 REFERENCES

1. Letter from J. W. Hampton, DPC, to USNRC, dated April 12, 1995.

2. Letter from E. W. Merschoff, NRC, to J. W. Hampton, DPC, dated February 23, 1995.

3. GL 83-28, "Required Action Based on Generic Implications of Salem ATWS Event," dated July 8, 1983.

4. Safety Evaluation Report, "Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-28, Generic Letter 83-28, Item 2.2.1, Equipment Classification Programs For All Safety-Related Components," November 4, 1987.

Historical Perspective of Oconee's System and Component Classification Methodology

This section provides one interpretation of current system and component classification methodology for nuclear power plants. It also explains how Oconee's methodology is different. This interpretation and explanation may not be accurate in every detail, and may not include every relevant aspect of system and component classification. However, the perspective provides a context for understanding the Event-Based Classification Methodology presented in Sections 1.0 - 3.0.

5.1 Classification Methodology in Current Use

After Oconee licensing, regulations and standards evolved to provide methodology that resulted in some consistency for identifying systems and components important to nuclear safety. This meant that the structures, systems and components (SSCs) designed to standards such as ASME and IEEE were generally the same SSCs that met single failure requirements and seismic design requirements. These SSCs were also the same ones designated "safety-related," and were the same ones included in Appendix B Quality Assurance programs. Further, as new regulatory requirements were added, these same SSCs defined the scope for ASME Section XI programs, motor-operated valve programs, environmental qualification programs, and maintenance rule programs. There are exceptions, but with current-day usage, the unifying basis for the scope of all these plant programs and for system design criteria is the designation "safety-related."

An example of current-day usage of "safety-related" is presented succinctly in the introduction of Regulatory Guide (RG) 1.29, *Seismic Design Classification.* It ties together the scope of SSCs that are:

- subject to seismic design requirements
- safety related (i.e., derived from the 10CFR100 Appendix A definition of systems and components required for the Safe Shutdown Earthquake)
- subject to 10CFR50 Appendix B quality assurance requirements

The regulatory guide states "This guide describes *an acceptable method* for identifying and classifying those plant features" that should be seismically designed. This *acceptable method* is simply provision of a comprehensive list of SSCs that can be important to nuclear safety.

Essentially the same list of SSCs found in RG 1.29 is repeated in RG 1.26, *Quality Group Classifications and Standards*, but the list is divided into three tiers. The highest tier is matched to the requirements of ASME Section III Class 1, the next tier to Class 2, and the third tier to Class 3. Thus, the same list of SSCs is also used to establish piping classifications.

The lists of SSCs provided in RG 1.26 and RG 1.29 are found again with little modification in ANSI N18.2 (1972), ANSI N 18.2a (1975) and ANSI N51.1 (1983). The typical current-day answer to the question of what is "safety-related" is to point to these lists of SSCs. The link between these lists, and the concept of certain SSCs being more important to safety than others, is the definition of "safety related." This definition is in 10CFR100 Appendix A, and is repeated in 10 CFR 50.49 (EQ), 10 CFR 50.65 (Maintenance Rule), and GL 89-10 (MOV Testing and Surveillance). Quoting from GL 89-10:

> The term "safety-related" refers to those systems and components that are relied on to remain functional during and following design-basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100.

GL 89-10 then proceeds to define the term "design-basis events" (DBEs) used in the definition (this is also identical to the definition in 10 CFR 50.49):

> Design-basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure the functions delineated [in the "safety-related" definition above].

Therefore, in a nutshell, "safety-related" SSCs are those used for DBE mitigation to ensure three things: RCS pressure boundary integrity, safe shutdown, and meeting Part 100 guidelines. It is this set of SSCs that are intended to be bounded by system lists such as that found in RG 1.29.

5.2 Comments on Current Classification Methodology

It should be noted that RG 1.29 and the related RGs and industry standards provide a short-cut method for determining all SSCs that are relied upon for RCS pressure boundary integrity, for safe shutdown, and for meeting Part 100 guidelines. This short-cut is simply a conservative, exhaustive list of all SSCs that could possibly be important in mitigating design basis events. The list has been so conservative that the industry and the NRC have agreed in principle on graded quality assurance programs that do *not* apply full QA program requirements to a significant portion of these SSCs.

It is also important to note that other alternatives exist for identifying SSCs subject to design requirements and QA programs. Oconee's license is based on an alternative methodology, established prior to RG 1.29, prior to 10 CFR 100 Appendix A, and prior to 10 CFR 50 Appendix B. The Oconee methodology is outlined next.

5.3 Original Oconee Methodology

Unlike the current-day SSC classification process, Oconee SSC classification does not hinge directly on the Three-Part Safety-Related Definition quoted earlier. Oconee's license pre-dated general use of that definition. Therefore, some of the SSCs which would now be identified within the scope of that safety-related definition are *not* in Oconee's QA-1 program, are *not* designed to seismic or single failure criteria, are *not* subject to ASME Section III, and are *not* designed in accordance with IEEE or ANSI standards. Most current nuclear standards had not been drafted at that time. Relative SSC importance at Oconee was determined using a different set of criteria, as explained in the following paragraphs.

5.3.1 General Design Criteria

Oconee design was driven heavily by proposed General Design Criteria (GDC) that eventually became final in 10 CFR 50 Appendix A. Oconee remains committed only to the proposed GDCs. GDC-1 and -2 were important for SSC classification issues. GDC-1 required the identification of systems and components essential to accident prevention and mitigation to ensure that appropriate quality standards would be applied. GDC-2 required that those same essential systems and components be designed to withstand natural phenomena loads. However, little guidance was provided in the GDCs on how to identify the essential systems and components.

It is inferred that Oconee used a "Lines of Defense" approach to identify essential systems and components. This approach is documented in ANSI N51.1 (1983), Appendix C, *Historical Background and Rationale for Equipment Classification.* This approach identifies "successive and independent lines of defense [to] protect against the release of radioactivity to the environs (i.e., fuel cladding, reactor coolant pressure boundary, emergency core cooling system, and primary containment)." And indeed, in FSAR Section 3.1.1, Oconee lists the following five systems as essential:

Reactor vessel internals	(fuel)
Reactor coolant system	(RCS pressure boundary)
Engineered safeguards systems	(ECCS + containment isolation)
Reactor building	(Containment)
Electric emergency power sources	(Supports ECCS & containment isolation)

Of course, capability for mitigation of the Large Break LOCA was considered to be of major importance, and was a primary influence in the selection of the above essential systems.

5.3.2 Piping Classes

GDC-1 and -2 identified essential systems and components, but this categorization was not sufficient for all classification purposes. ANSI B31.7, *Nuclear Power Piping,* imposed three classes of nuclear piping requirements, but left it to the Owner to choose what systems belonged to each class. Oconee's selected method for pipe classification was to use the "Multiple Fission Product Barriers" approach. Again, this methodology is also documented for historical purposes in ANSI N51.1, (1983), Appendix C, which states it was "based on a relative consequence from failure of fluid-containing barriers." Oconee applied the nuclear piping code to systems containing radioactive fluid, and classified these systems as follows:

Class 1 piping	reactor coolant system
Class 2 piping	piping containing high-temperature radioactive fluid
Class 3 piping	piping containing low-temperature radioactive fluid

Note that piping systems with important mitigation functions, but that did not contain radioactive fluid, were not included within the scope of ANSI B31.7. Piping classification for ANSI B31.7 application was not function-based.

5.3.3 System Design Criteria

Additional system criteria were imposed with respect to natural phenomena and loss of coolant accidents. This system criteria led to another category of important SSCs: those subject to seismic and tornado design criteria. Oconee FSAR Section 3.2 describes these additional important SSCs as follows:

LOCA:	Protective actions, including reactor trip and ECCS, will succeed even with a simultaneous earthquake.
Earthquake:	Safe shutdown capability is provided. The FSAR lists systems and equipment that can withstand the maximum hypothetical earthquake. Support systems for some of the equipment were not included.
Tornado	Safe shutdown capability is provided. The FSAR lists the ASW system, water sources, and power supplies for ASW. Primary-side tornado-mitigation systems are not listed.

5.3.4 QA Program Scope

Appendix B to 10 CFR 50 was first issued in 1970, after Oconee design and construction was well underway. The Appendix B requirements, eventually addressed in Oconee's QA-1 program, imposed yet another need for SSC classification. SSCs included in QA-1 scope were those deemed important for reasons such as those just described; e.g., those listed for GDC-1 and -2, those assigned to Class 1-3 piping classes, and those explicitly listed as being capable of withstanding seismic and tornado loads.

Therefore, the Oconee QA-1 scope was not directly linked to design basis event mitigation, and also does not tie directly to satisfying the current-usage Three-Part Safety-Related Definition for ensuring RCS integrity, safe shutdown, and dose within

Part 100 guidelines. Oconee's methodology determined SSC importance to safety using different criteria than plants licensed a few years later.

An April 12, 1995, letter to the NRC states that only those SSCs listed for GDC-1 (i.e., those specified in FSAR Section 3.1.1) were originally committed to the Atomic Energy Commission (AEC) to be in the QA-1 scope. Other SSCs were also originally included in the QA-1 scope by Duke, but the April 12th letter indicates their inclusion was not considered to be a commitment to do so. The letter explains that only the original commitments of FSAR 3.1.1 plus later commitments made since the original license comprise the extent of scope Oconee has committed to address under 10 CFR 50 Appendix B.