UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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.

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

2) <u>Second category</u> - 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.