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SUBJECT: Forwards description of Oconee QA-1 licensing basis,suppl response to Subpart 1 of Section 2.2.1 of GL 83-28 & generic criteria for classifying QA-1 structures,sys & components.

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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

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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.

<u>History of Development of Oconee's QA (QA-1) Program</u>

<u>Pre-PSAR: AEC/DPC Dialogue</u>

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.

<u>Early Duke Design and Piping classification</u>

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.

PSAR

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".

<u>Oconee QA-1 SSCs Added to the Original Licensing Basis</u>

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.

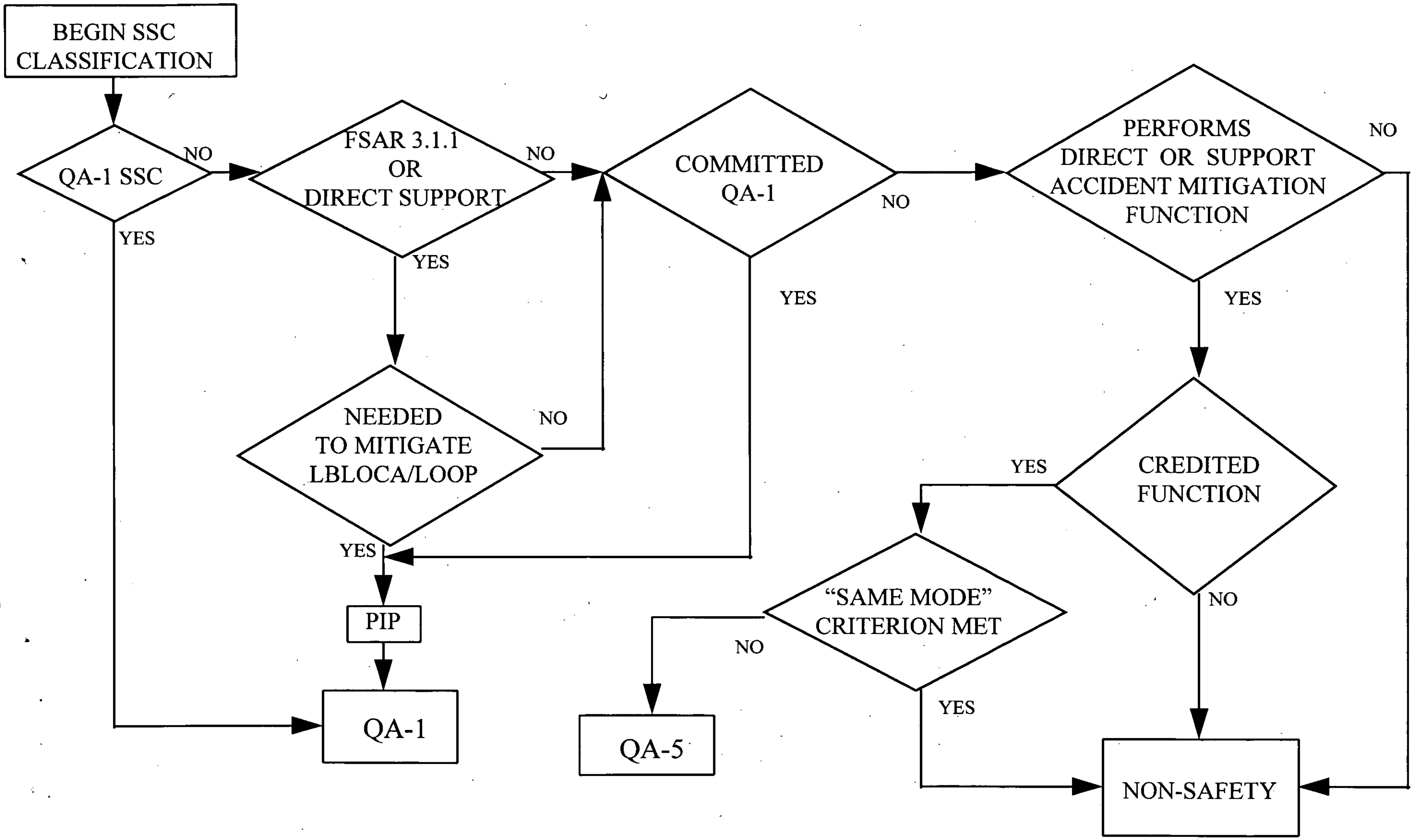
9. If the SSC performs a direct or support accident mitigation function it will be reviewed to determine if it is taken

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.