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SUBJECT: Responds to NRC 981105 ltr re violations noted in insp repts 50-269/98-08,50-270/98-08 & 50-287/98-08.Corrective actions: component database was 80% complete as of 980731 & is scheduled to be completed 981216.

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November 16, 1998

U. S. Nuclear Regulatory Commission
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Subject: Oconee Nuclear Station
Docket Nos. 50-269, 270, 287
Response to NRC Inspection Report 98-08

This letter provides Duke Energy Corporation (Duke) comments related to certain aspects of NRC Inspection Report 50-269, 270, 287/98-08 (IR 98-08) issued on October 5, 1998. IR 98-08 contains a detailed review of the Oconee Safety Related Designation Clarification (OSRDC) Project, comments on certain design characteristics of many systems at the plant, and probabilistic risk insights regarding certain Emergency Feedwater (EFW) System issues. The attachment to this letter provides Duke's perspective regarding specific findings in IR 98-08 related to the above topics. It is Duke's hope that sharing our perspective on these issues will help to foster further discussions with the NRC with a goal to achieve a common understanding on these matters.

A key objective of the OSRDC Project was to assure that the design of Oconee was within its licensing basis. The OSRDC Project clarifies those structures, systems, and components (SSCs) that are QA-1 (included in the scope of 10 CFR 50 Appendix B) within the Oconee licensing basis. Duke performed extensive research to clarify those SSCs that are QA-1 and submitted the results of this effort to the staff in 1995. Duke's definition of QA-1 SSCs was approved by the staff in a safety evaluation report dated August 3, 1995. Thus, the scope of Oconee's SSCs should be compared against the scope approved in the August 3, 1995 safety evaluation report. However, certain language in IR 98-08 could be interpreted as comparing the Oconee design and licensing basis to typical requirements for a newer facility, which, in our view, would be inappropriate. Specific examples are addressed in the attachment to this letter.

Duke recognizes that the NRC's engineering review in IR 98-08 used risk insights from the Oconee PRA, which are beneficial in focusing both NRC and Duke resources on issues of potential significance. Duke believes that it is important to maintain a macroscopic review of overall plant risk when assessing individual issues. Duke has reviewed the inspection report related to risk-insights and has performed additional analyses to address specific comments and conclusions in the inspection report. As a result, Duke believes that issues related to the EFW System are not as significant from a risk and reliability perspective as may be implied in IR 98-08. A brief summary of these risk analyses is provided in the attachment to this letter. The intent of these comments is to share our perspective of the overall significance of some of the inspection report statements with the NRC.

Duke is committed to the safe, reliable operation of Oconee Nuclear Station and continues to encourage a low threshold for the identification and resolution of problems. Duke is submitting its perspective on these issues to further increase the dialogue to achieve a common understanding on certain issues identified in the report. Duke wishes to further discuss these comments as part of the normal inspection process as both organizations continue to work to resolve open items in the Management Oversight Group charter.

Sincerely,

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Response to IR 98-08

The following are statements contained in IR 98-08 which Duke believes need clarification.

1. **Section E1.1.b, "The inspectors noted that some of the level of detail in the data was good in that related indication and associated components were included. However, some equipment was notably missing such as electrical power supplies (e.g. breakers and relays). Licensee engineers stated that they planned to add electrical components to the database."**

These statements refer to the Oconee Safety Related Designation Clarification (OSRDC) Project.

The OSRDC Project included within its plans identification of electrical power supplies such as breakers and relays. Identification of these components was scheduled to be completed after the time of the inspection. The initial OSRDC endeavor was to complete the component database for 23 of 27 events by the end of 1998 and complete the remaining four events during 1999. The component database for this phase was 80% complete as of July 31, 1998, and is not scheduled to be completed until December 16, 1998. OSC-6100, Instructions For Preparing The Event Mitigation Database, Revision 3, Attachment C defines the process used to develop a component database of all components required to support ONS events. The process defines the systems needed to mitigate specified events then identify the mechanical hardware, and finally all the support components which includes breakers, relays, switches, etc. Section 6.3 of OSC-6100, Attachment C defines the equipment required for electrical support.

2. **Paragraph E8.1.c, "The inspectors found that many components that were relied upon to mitigate design basis accidents were not in a QA program. Almost half (15 of 31) of the components reviewed were not fully QA."**

The fact that all of the components relied upon to mitigate DBAs are not within the scope of 10 CFR 50 Appendix B is consistent with Oconee's licensing basis. Appendices 1A and 1B of the original FSAR define the QA program at the time of licensing. The NRC Safety Evaluation to Supplemental Response to GL 83-28, dated August 3, 1995, acknowledges this fact. That SER also contains a description of the ONS QA program at the time of licensing. It states that: "The FSAR Appendix 1B list of 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, or,

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

There are some examples of SSCs that do not appear on the Appendix 1B list that are required for mitigating a LBLOCA/LOOP, such as portions of the CCW System. … it was recognized that some features of these non-nuclear, USAS B31.1.0 systems permitted their exclusion from the QA 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."

It was acknowledged in the ONS licensing basis that not all SSCs necessary to mitigate the design basis accident, LBLOCA/LOOP, were required to be or would be in the QA program.

Duke voluntarily established the OSRDC program in 1995 to specifically address the issue of non-QA SSCs mitigating design basis events. The OSRDC initiative resulted in the development of an Augmented QA program. These non-QA-1 components will be assigned a QA category called QA-5 and will be maintained and tested in a manner similar to QA-1. Duke submitted a description of OSRDC and the QA-5 program to the NRC on April 12, 1995. The August 3, 1995, SER clearly defines what is QA-1 at ONS and that non-QA SSCs are used to mitigate design basis events.

In addition to the original licensing basis QA program, Duke had committed to include additional SSCs in its QA program. The additional SSCs are listed in Section 3.1.1 of the UFSAR as Second Category-SSCs Added To The Original Licensing Basis.

3. **Section E8.1.b, Single Failure, "The EFW system was described in the UFSAR as safety-related and QA-1."**

This statement is not correct. UFSAR Section 10.4.7, Emergency Feedwater System, does not describe the EFW System as being safety-related and QA-1. UFSAR Section 3.1.1.1, QA-1 SSCs Added To The Original Licensing Basis, Item 1, states that the following portions of the emergency feedwater (EFW) systems are QA-1 .

- the motor driven (MD) EFW pumps
- the piping from the MD EFW pumps to the steam generators
- the EFW flow control valves (excluding the operators)
- the power supply to the MD EFW pumps and controls
- piping from the upper surge tanks (USTs) to the MD pumps
- UST level monitoring circuitry and associated valves
- EFW flow transmitters upstream of the flow control valves
- MD and turbine-driven EFW pump initiation signals

Thus, only portions of the EFW system are classified as QA-1.

4. **Section E8.1.b, Single Failure, "The caustic pump and valve were not QA, were not seismic, were not in a harsh environment, were not in the IST program, and were not in the TS or another administrative control program. At newer plants, this function was typically included in the automatic safety related systems."**

The caustic pump and valve are not required to be QA or seismic by the licensing basis as defined in Appendices 1A and 1B of the FSAR. They neither mitigate a LBLOCA/LOOP, are part of a RCS pressure boundary, nor meet the requirements of Draft IEEE 279, 1E. They will, however, be considered for inclusion in the QA-5 program as part of the OSRDC project and are covered under the Maintenance Rule.

5. **Section E8.1.b, Single Failure, "Thirteen of the 30 components were not seismically designed. … Seven of the 13 were not included in the SQUG program."**

These statements appear to suggest in the context of the report that these components should be seismic and/or part of the SQUG program.

The list of 30 components was a snapshot of selected QA-1 and non-QA-1 equipment used to mitigate events. This list was generated from the results of the OSRDC project as an aid for the inspector's review. The list was generated using an event based approach (consistent with current day definitions of event mitigation).

Section 3.1.1 of the UFSAR defines the list of SSCs which are "essential". This became the basis of the ONS QA-1 program scope. Section 3.2.2 defines the list of SSCs that were designed to be seismic. It is important to note that each of these lists were NOT created based on an event mitigation approach. In addition, support systems were generally not included. This is consistent with the classification methodology that was generally in use in the late '60s and early '70s.

The SQUG methodology is event based. It generated a list of equipment required to maintain the plant in a safe shutdown condition for 72 hours after an earthquake and coincident with a LOOP. No other accident or transient is postulated (i.e., no LOCA). The scope of SQUG is documented in ONS Calculation OSC-5710 which was generated in accordance with accepted industry guidance. The ONS SQUG program has been reviewed and approved by an industry peer group. The NRC accepted the ONS SQUG Program in an SER dated May 22, 1992.

Since the UFSAR list of seismic equipment, the OSRDC project, and the SQUG lists were each generated using a different methodology and a different set of input assumptions, it is expected that they have different results. In summary, the systems described in the inspection report have appropriate seismic classifications in accordance with the Oconee design and licensing basis.

6. **Section E8.1.b, TS and SLC, The report stated that the atmospheric dump valves were not included in either the TS or SLC administrative control programs.**

Duke would like to clarify that SLC Section 16.9.9, Auxiliary Service Water Main Steam System And Atmospheric Dump Valve Operability Requirements, states that the Auxiliary Service Water (ASW) System shall be OPERABLE with the ASW pump and the associated piping and valves. It further states that the Main Steam Atmospheric Dump Valves are included as part of the associated valves. In addition, the SLC specifies that the Atmospheric Dump Valves will be stroked at each refueling outage.

7. There were several comments in the inspection report which related to PRA vs. UFSAR issues. These comments were primarily concerned with differences between the PRA and the UFSAR.

In Duke's view, a risk-informed approach should maintain a high level perspective of how systems, components, and operator actions contribute to plant risk and ultimately to public health risk. The EFW system is an important contributor to plant risk; however, the importance primarily lies in turbine driven pump failures during blackout sequences and UST failure during tornado events. System models are intended to support the high level risk perspective and are not to be used as stand alone analyses.

The PRA is well suited for identifying important aspects of plant operation and design, but does not necessarily lend itself well to identifying all potential discrepancies in the UFSAR. This stems from the fact that PRAs have been developed from a completely different set of goals, assumptions, and ground rules. The focus here would be on the more significant failure modes of the system. Attempting to compare and verify modeling details and results between the PRA and the UFSAR leads to inconsistencies and contradictions. The UFSAR assumptions and criteria are based on deterministic considerations and at this time are not fully supported by a risk-informed rationale. The PRA is of value in identifying the relative safety significance of the system, the risk significant failure modes and providing perspective on the more important versus less important aspects of system performance. Therefore, a high level risk perspective should be maintained in identification and resolution of differences between the PRA and the UFSAR.

Specific Report Comments

Below a number of comments and clarifications are provided concerning several PRA related issues.

a. Potential Failures of Condensate Valve C-187

Section E8.1 b, Single Failure, "However, the PRA apparently contradicted the UFSAR when it stated that a single failure of C-187 coincident with a main feedwater line break would cause a loss of all EFW."

Response:

Validating or supporting all UFSAR statements is not one of the main objectives of the Oconee PRA or any of the other probabilistic risk and reliability studies conducted at Oconee. Probabilistic analyses do not postulate "single failures" in a licensing sense, but instead consider the probability of functional failure of systems or components that could lead to core damage. The frequency of accident initiating events are combined with the probabilities of functional failure of accident mitigation systems and equipment to derive an annual plant risk of core damage. The failure of an individual system (such as EFW) does not constitute an accident sequence and should be kept within the context of how this system failure contributes to a core damage sequence and thus to Core Damage Frequency (CDF).

In the case of condensate valve C-187, all potential accident sequences (i.e., cut sets) involving the failure of this valve to close on demand fell below 1E-08 /Rx-yr which is a negligible contribution to risk.

b. HELB Causing Failure of all EFW and all 4160V Power

Section E8.1 b, Single Failure, "Also, a main feedwater line break analysis that described a consequential loss of all EFW and all three trains of safety-related 4160 volt switchgear was apparently not addressed by the PRA."

Oconee has undergone numerous PRA Studies beginning with NSAC/60 in 1981 up to the current Oconee PRA analysis. The early Oconee PRA studies in fact received a considerable amount of review from the nuclear industry and the NRC. The Oconee PRA has been subsequently updated twice. None of the three studies considered the HELB induced loss of power accident sequence to be a probabilistically significant sequence.

The main switchgear is located on the mezzanine level of the Turbine Building in an area that consists almost entirely of electrical equipment, cable trays, distribution panels, and electrical buswork. There are only a very limited number of lines in the "high energy" category in this area and only a handful of piping segments that are close enough to affect any of this equipment. Furthermore, the turbine driven emergency feedwater (TDEFW) pump is designed to run without electrical power and would have to be affected in some way by the failure. Given that the TDEFW pump is located on the opposite side (east to west) of the Turbine Building from the electrical switchgear equipment, a failure that impacts the manual start of the TDEFWP is extremely unlikely.

However, supposing that one segment of piping exists, the probability of piping failure can be estimated based on EPRI TR-102266 "Pipe Failure Study Update". This report estimates the probability of piping failure (>6" dia.) of PWR feedwater/condensate piping to be approximately 6E-10 /hour. This report also determined that roughly half (50%) of these breaks fell into smaller break categories. Using this failure rate (6E-10), assuming a 50% break size factor, and assuming a plant capacity factor of 90%, the probability of such a break is estimated to be 2.4E-06 /Rx yr.

What is still missing from this probability is the likelihood that the break will occur at a very specific location and angle on the piping segment such that the spray from the break is directed at a specific area to cause damage or loss of power to all three main switchgear. This consideration is likely to lower the event probability by at least another 1 to 2 orders of magnitude. If such an unlikely failure were to occur, the ASW switchgear in the Aux Bldg would still be available to power the Aux Service Water Pump (for feedwater) and one HPI pump (for RCP seal cooling and ECCS). EFW from another unit would also still be available and the plant could also activate the SSF. Therefore, the conclusion is that this type of accident sequence is probabilistically insignificant and does not warrant explicit modeling in the Oconee PRA.

c. Risk Significance of Throttling EFW Flow

Section E8.1 b, <u>Single Failure</u>, "The PRA described the operator action to immediately throttle EFW flow (using the EFW flow control valves and manual loaders) as a significant contributor to the probability for the loss of all EFW."

Response:

Based on additional analysis by Duke, this operator action is not a significant contributor to the probability for the loss of all EFW. A statement to this effect did appear in the results section of Appendix A.7 of the Oconee IPE study. In the EFW solution process this event (FEFEFW2DHE) appeared in cut sets that would have high SG pressure although the event description and other assumptions implied that the event was a potential failure mode only for sequences with a depressurized SG. The incorrect characterization of EFW pump runout failures was corrected in the Oconee PRA, Revision 2 update (although it has a new event name FEFISOLDHE). When the event is correctly tied to events with low SG pressure, it does not appear in the top EFW cut sets and not in any core melt cut sets above 1E-08/Rx yr. Thus, this operator action is not a significant contributor to the probability for the loss of all EFW.

d. 24 hr Mission Time - Discussion on Caustic Valve

"The pump and the valve were not in the PRA, since the lack of caustic addition would not have any impact on core damage probability and would have little effect on releases to atmosphere within the first 24 hours of an accident. Note: The PRA only considers the first 24 hours of an accident."

The PRA analysis of the effects of an accident on containment and potential releases to the environment (Level 2 and Level 3 analyses) are considered for a time period greater than 24 hours for estimating public health consequences.