October 6, 2000

MEMORANDUM TO: Richard L. Emch., Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: David E. LaBarge, Senior Project Manager, Section 1 ***/RA by***
Project Directorate II ***Richard L. Emch Acting for/***
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: MEETING WITH DUKE ENERGY CORPORATION ON THE OCONEE NUCLEAR STATION EMERGENCY FEEDWATER SYSTEM (TAC NOS. MA9294, MA9295 AND MA9296)

DATE & TIME: October 25, 2000
9:00 a.m.

LOCATION: U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, Maryland
Conference Room O-9 B4

PURPOSE: To discuss information needed by the staff for its review of the June 21, 2000, submittal proposing changes to the Oconee Nuclear Station Updated Final Safety Analysis Report section of the Emergency Feedwater System

PARTICIPANTS*:

NRC	Duke Energy
J. Hannon, NRR	W. Foster
J. Tatum, NRR	L. Nicholson
R. Emch, NRR	E. Burchfield
D. LaBarge, NRR	N. Clarkson
R. Eckenrode, NRR	A. Park
S. Rhow, NRR	

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

Attachment: Summary of Potential Discussion Items

cc w/atts: See next page

CONTACT: D. LaBarge, NRR
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*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, petitioners, interveners, or other parties to attend as observers pursuant to "Commission Policy Statement on Staff Meetings Open to the Public" 59 *Federal Register* 48340, 9/20/94.

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<u>DISCUSSION QUESTIONS RELATED TO</u>
<u>UFSAR PROPOSED AMENDMENT DATED JUNE 21, 2000</u>
<u>OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 EFW SYSTEM</u>

A. The Oconee Emergency Feedwater (EFW) system was originally reviewed and approved based on information that was provided and assertions that were made by the licensee largely in response to Three Mile Island (TMI) Action Plan items. It is the staff's view that, following the TMI event, the Oconee EFW system was approved based on information that was submitted by the licensee indicating that the EFW system was in compliance (or would be modified to be in compliance) with the staff's review criteria (e.g., safety-related, seismic Category I, Class IE, environmentally qualified (EQ), able to mitigate all design-basis events including main feedwater line break (MFLB), able to withstand a single active failure and loss of offsite power (LOOP) concurrent with postulated events, fully automatic in order to prevent steam generator dryout and loss of reactor coolant system (RCS) inventory, and able to cool the plant down to decay heat removal (DHR) entry conditions within its established mission time). Essentially, a standard safety-related system was approved and very few exceptions to the staff's criteria were noted. For efficiency and to facilitate the staff's review, the licensee is requested to identify each specific case where the staff's review criteria for EFW are not currently satisfied (including design attributes, accident mitigation capability, and TMI Action Plan requirements). The following information should be provided for each specific case that is identified (as applicable):

1. For exceptions to the staff's criteria that were previously approved by the NRC, provide reference to the document(s) where the staff granted its approval for the exception.

2. Provide justification as to why the exception should be allowed, including a description of how the applicable EFW function will be satisfied and a discussion of the risk implications.

3. For each non-safety related component that is relied upon, describe how the availability and reliability of the component will be assured, including a discussion of any 10 CFR Part 50, Appendix B, quality requirements that will not be satisfied; a discussion of Technical Specification (TS) requirements that will be imposed; administrative controls, maintenance and surveillance requirements that will be established; and consideration of any other provisions that may be pertinent such as Maintenance Rule (Title 10 of the *Code of Federal Regulations* [10 CFR] Section 50.65) requirements.

4. For each non-safety related component that is relied upon, provide the revised Updated Final Safety Analysis Report (UFSAR) pages that accurately reflect the applicable changes in the design basis for these components.

5. Provide a complete listing of all manual actions that are relied upon and indicate which ones were previously approved by the staff, identifying the reference(s) that provided the staff's approval. For those manual actions that were not previously approved by the staff, provide the following information:

    a. operator actions or manipulations required,
    b. environmental conditions (temperature, humidity, noise level, lighting, radiation) where the actions are taken,
    c. ingress and egress paths and accessibility to the required equipment,
    d. procedural guidance necessary to complete the actions,
    e. required training, including operator qualifications,

f. information required and instrumentation relied upon to determine the need for action and to verify action completion, including quality requirements,
g. time available to recognize the need for action and time to complete the action after the need is recognized, including time estimates and the basis for these estimates,
h. ability to recover from credible errors in performance,
i. staff available to complete the required actions, how assigned, and other duties and responsibilities of the position, and
j. risk significance of the proposed actions.

B. The questions that follow help illustrate the sort of information that is needed for completing this review. The answers to these questions should be included with the discussion of the specific cases referred to above, or (when a specific case is not really applicable) the answer should be provided separately.

1. Risk conclusions are included without supporting analysis or other explanation. A discussion or explanation of the risk assessment should be included in support of the risk conclusions contained in the submittal. For example, the Discussion of Proposed Changes, p. 9, states that the postulated high energy line break (HELB) scenario that damages the 4KV switchgear results in a core damage frequency (CDF) of approximately 4E-7. It further states that, based on this low risk, further modifications to the facility are not justified. Further explanation of the analysis is needed.

2. The submittal requests approval to rely on other safety-related and nonsafety-related equipment (e.g., auxiliary service water (ASW), EFW cross-connect valves) without indicating the measures that will be taken to assure the equipment will be available and reliable (controls, configuration management, equipment upgrade, periodic testing, maintenance, etc.). A discussion of the measures that will be taken in this regard is required.

3. Identify all equipment that is relied upon to perform a safety-related function that is not classified as Oconee Quality Assurance (QA)-1, and provide justification for why this equipment should not be re-classified as QA-1.

4. Define the specific criteria that will be established for relying on non-safety equipment for mitigating design basis events. Describe additional measures that will be taken (e.g., inspection, testing, periodic maintenance, equipment qualification) to support reliance on less than safety-related equipment.

5. Describe changes that will be made to the licensing basis, the UFSAR, and the TS related to the other safety-related and non-safety related systems and equipment (including systems and equipment from the other units) that is relied upon to support

the EFW function of a given unit. For example, the submittal addresses only proposed changes to UFSAR Section 10.4.7, Emergency Feedwater System. In addition to reduced design criteria for the EFW system, the submittal proposes new or expanded safety functions/design criteria for the Standby Shutdown Facility (SSF) ASW system; EFW unit crosstie valves; station ASW system and steam generator (SG) atmospheric relief valves; and high pressure injection (HPI) system feed and bleed.

6. UFSAR Rewrite, p. 10-22, states that the TS required 72,000 gallons of EFW water supply is sufficient to cool down with the reactor coolant pumps (RCPs) and the secondary plant available. Does the proposed design basis change take into consideration the availability of a sufficient quantity of water for a cooldown with a LOOP concurrent with a steam generator tube rupture (SGTR) or a small break loss of coolant accident (SBLOCA)? Explain.

7. UFSAR Rewrite, p. 10-23, states that long-term inventory can be provided to the SGs by SSF ASW, station ASW, or the EFW cross connect from another unit. Provide additional explanation to address the following concerns:

    a. The station ASW (and SG atmospheric relief valves) are not safety-related and may not be available when the upper surge tank (UST) becomes empty. Station ASW cooling takes about one hour to get started, it is somewhat low in reliability (about 10 percent probability of failure), and has never been fully tested.

    b. The EFW unit cross connect is not fully safety related and may not supply sufficient water to the affected unit. An operating unit may have only about 30,000 gallons (one hour) of extra feedwater available in its condensate storage tank (CST). There is no evidence that ESW flow testing between all units has been performed, and there may not be an engineering analysis or operating procedures to support this method.

    c. The SSF is safety related, but it is notably unreliable (about 30 percent probability of failure). The SSF EDG may not meet current standards for emergency power systems reliability and testing.

8. UFSAR Rewrite, p. 10-24, states that the HPI system can remove decay heat via RCS feed and bleed. If this method is relied upon to mitigate design basis events, should the pressurizer power operated relief valve (PORV) be safety-related, seismic, EQ, and be included in the TS to ensure operability?

9. UFSAR Rewrite, p. 10-24, states that the condenser hotwell is designed to withstand an maximum hypothetical earthquake (MHE) with a nominal available capacity of 120,000 gallons. Previous licensee statements indicated that the piping attached to the hotwell was not seismically designed and, if that piping broke, it could empty the hotwell water onto the turbine building floor. How does the Oconee design protect the hotwell water during a seismic event? If this is not important, explain why.

10. UFSAR Rewrite, p. 10-24, states that the piping from the hotwell to the motor-driven emergency feedwater pumps (MDEFWPs) is seismically qualified. Is this the same as seismically designed? Previous licensee statements indicated that the piping from the hotwell to the EFWPs was not seismically designed. How and when did it become seismically designed? Explain the difference.

11. UFSAR Rewrite, p. 10-25, states that manually operated valves that provide a seismic boundary are normally closed. However, this statement is contradicted on p. 10-26 by the following statement: "Each motor-driven EFW pump recirculation line is provided with a normally open manual valve as its seismic to non-seismic boundary." How does this meet the seismic design requirements? This seems to be contrary to statements in the licensee's letter of May 7, 1986, that described the seismic qualification of the EFW system.

12. UFSAR Rewrite, p. 10-26, states that if the EFW flow control valve for the unaffected SG failed to open during an accident, the flow path could be realigned to bypass the failed valve and reach the SG through the (non-safety) main feedwater startup path. Other plants operate with EFW flow control valves normally open to eliminate this potential EFW flow control valve failure mode. Explain why Oconee's flow control valves are normally closed.

13. UFSAR Rewrite, p. 10-26, states that, for a steam or feedwater line break, if the EFW flow control valve on the unaffected SG fails open, . . . both SGs must be isolated within 10 minutes. Except in those cases where the break makes these valves inaccessible, an operator could manually (locally) adjust either valve. What actions are specified if the valve is inaccessible? In addition, p. 10-33, states that in the unlikely event that the EFW flow control valves fail open, an operator could manually adjust either one of the valves. Explain whether this action is performed inside or outside of the control room, and what is involved in performing this adjustment.

14. UFSAR Rewrite, p. 10-29, states that condenser vacuum is broken by the opening of a single vacuum breaker valve (V-186), and that this valve is normally operated from the control room. However, shifting EFW pump suction to the hotwell would most likely be required during a LOOP event to operate the hotwell pumps (which are used to pump the hotwell water up to the UST when power is available). Since this suction flow path for the EFW pumps requires that condenser vacuum be broken, is V-186 and power to it safety related and how would V-186 be operated from the control room during a LOOP?

15. UFSAR Rewrite, p. 10-32, states that long term secondary cooling is discussed in Section 10.4.7.3.8. Shouldn't this be Section 10.4.7.3.9?

16. UFSAR Rewrite, p. 10-33, states that, for feedwater or main steam line breaks causing loss of SG pressure boundary, the operator is required to manually terminate EFW flow to the faulted SG. It does not state that this is a time-critical action that must be completed within 10 minutes for containment overpressure protection. It also does not state that a manual operator action of throttling EFW flow to the faulted SG must be completed within three minutes for EFW pump runout protection. Explain why this information was not included and what provisions have been made to ensure that these actions will be completed within the required time limits.

17. Discussion of Proposed Changes, p. 7, indicates that the MDS Report No. OS-73.2, dated April 25, 1973, on the effects of high energy line breaks outside of the containment, was approved by the Atomic Energy Commission on July 6, 1973, and continues to be the licensing basis for HELB considerations at Oconee. Oconee had concluded in 1973 that, with a station ASW pump providing water to a SG, or by using a feed-and-bleed approach, the reactor core would be safe for an extended period of time (many hours). However, since that time a concern has been identified that a

HELB in the turbine building could cause a loss of switchgear important for RCP seal cooling, which could result in RCP seal failure and a LOCA. Explain whether or not this is a valid concern, and whether the HELB analysis is still valid.

18. Discussion of Proposed Changes, p. 8, states that the licensing basis as it relates to HELBs provides exception to the single failure criterion for those HELBs that can cause a complete loss of main and emergency feedwater on the affected unit when coupled with a single active failure in the EFW system. It further states that this exception is justified considering the low CDF significance of such a postulated pipe break. This position seems to be premature since the CDF significance for a HELB that disables the three safety related 4160V safety related busses is currently the subject of an NRC unresolved item, an Accident Sequence Precursor review, and Oconee has committed to review of this issue and plans to complete its analysis by September 30, 2000. Explain.

19. Discussion of Proposed Changes, p. 9, states that restoration of feedwater within 30 minutes is sufficient. The NRC post-TMI action item that was approved for Oconee required emergency feedwater to be automatic, to prevent SG dryout and loss of RCS inventory. Without automatic EFW, Oconee SGs can be expected to go dry in about three to five minutes. The NRC has not reviewed/approved Oconee's analysis in support of an EFW system that is not fully automatic. For the 1973 HELB analysis, the NRC approved restoration of EFW within 15 minutes. However, that approval may have inappropriately relied upon an assumed very low CDF and no RCP seal LOCA. Further discussion about this issue and justification for the 30 minute delay in establishing feedwater is required.

20. Discussion of Proposed Changes, p. 21, states that the non-safety alternate flow path through the main feedwater (MFW) startup control valves is tested under Oconee's Appendix B test program. "Safety-related" at Oconee means that Appendix B applies. However, the MFW startup flow path is non-safety related.

    a. How does Appendix B apply and how is it being implemented for these valves?

    b. The Discussion of Proposed Changes, p. 22, states that the Appendix B program encompasses pumps and valves not included in the American Society of Mechanical Engineers program which are active in certain non-design basis events. However, it seems that the MFW startup flow path is being relied upon in design basis events. Explain the discrepancy.

21. Discussion of Proposed Changes, p. 22, states that, assuming offsite power is available, the MFW startup flow path can be aligned for EFW within 20 minutes. If offsite power is not available, how long would it take? It appears that the NRC approved reliance on the MFW startup flow path assuming that it could be quickly aligned from the control room. Oconee's letter of April 3, 1981, implies that it could be done from the control room. Also, the Discussion of Proposed Changes, p. 24, states that failure of the EFW flow control valve to open can be mitigated from the control room. Explain how this is accomplished from the control room without offsite power available. Is the method proceduralized?

22. Discussion of Proposed Changes, p. 29, states that the hotwell and demineralized water systems are monitored under the Maintenance Rule and implies that the CST is

not. Since the TS contains CST requirements, why is it and the CST pump is not covered by the Maintenance Rule?

23. Discussion of Proposed Changes, p. 30, states that the ability to refill the UST on the unaffected units from various sources makes the cross connect available to supply the affected SGs for the long term. This assumes that the demineralized water pumps, their switchgear, and offsite power are available. But if that is the case, the affected unit can refill its own UST and may not need the cross connect. Without the demineralized water pumps, where will the unaffected unit get enough water to supply the affected unit?

24. Describe the sequence of steps that would be followed to add water to the UST for each of the three sources (demineralized water, condensate storage tank, hotwell). Valves, location, method of operation, and assumed accessability during the worst case event that would require EFW operation. (Ref.: proposed revision to UFSAR Section 10.4.7.1.3). By procedure, what is the order of preference for refilling the UST during an emergency? This is important since the backup supply to the EFW pumps when the UST is no longer available is the hotwell, which means that water should not be pumped from the hotwell to the UST.

25. Discussion of Proposed Changes, p. 30, states that the station ASW pump motor is non-QA. Since the station ASW pump is relied on to mitigate certain events in place of the EFW system (e.g., to mitigate a tornado or a HELB that disables the three trains of 4KV safety-related switchgear), explain why is it not QA-1 or QA-5.

26. Discussion of Proposed Changes, p. 30, states that the EFW system is not designed to cool down to DHR conditions following any design bases event assuming a single active failure relying solely on the UST and hotwell of the affected unit. Explain how this is true for a MFLB with no active failures.

27. Discussion of Proposed Changes, p. 30, states that Duke will assure that the portions of the SSF systems, necessary for event mitigation, are fully qualified. How are they not fully qualified? What qualifications are referred to?

28. Observation: Discussion of Proposed Changes, p. 36, the last full ‘sentence’ on page 36 is not a sentence since it has no verb.

29. Discussion of Proposed Changes, p. 39, states that the following words from the existing UFSAR are incomplete and are being deleted: “all non-safety instrumentation and controls are designed such that any failure will not cause failure of any safety-related function.” Why is this design criteria being deleted? What are the non-safety instrumentation and controls that could cause failure of a safety-related function?

30. 10 CFR Part 50, Appendix A, Criterion 5, “Sharing of structures, systems, and components,” states that components important to safety cannot be shared among units unless it can be shown that such sharing will not significantly impair their ability to perform their safety function. Oconee UFSAR Section 3.1.4, Criterion 4, “Sharing of Systems,” states that reactor facilities shall not be shared unless it is shown that safety is not impaired by the sharing. A list of systems that share portions of the systems is included, but EFW is not on the list. Additionally, there is a statement that where there is sharing between Units 1 and 2, a separate system is provided for Unit 3. How is this

criteria satisfied for the EFW cross connects? Why is there no proposed change to this criterion or list of systems in the submittal?

31. The proposed UFSAR Section 10.4.7.1.2 contains a cooldown rate, time, and RCP temperature chart. Why are the assumed starting temperatures different for the two cooldown rates: 547°F for 100°F/hr and 480°F for 50°F/hr.?

32. The proposed UFSAR Section 10.4.7.1.4.1 states that "The EFW System is seismically qualified to the MHE level throughout the first isolation valves. Piping beyond these boundary points is not seismically qualified." What is the seismic qualification level of the piping between the first isolation valve and the SG?

33. With the UFSAR submittal and the design changes that either have been or will be implemented, have all of the 37 EFW single failure vulnerabilities that were identified in the Oconee study been addressed? Provide a summary description of how each of these single-failure vulnerabilities was addressed.

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