UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

January 29, 2004

EA-04-018

Duke Energy Corporation
ATTN: Mr. Ronald A. Jones
Vice President
Oconee Site
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - NRC INSPECTION REPORT NO. 05000269/2004007, 05000270/2004007, AND 05000287/2004007

Dear Mr. Jones:

On January 21, 2004 the U.S. Nuclear Regulatory Commission (NRC) completed an in-office and site inspection to resolve a previously identified issue at your Oconee Nuclear Station. The purpose of the inspection was to review Unresolved Item 05000269,270,287/2003003-002, High Energy Line Break (HELB) Accident Scenario Review that was identified in an inspection performed during the week of June 16 - 20, 2003. The enclosed report documents the results of this inspection which were discussed on January 27, 2004, with your staff.

This inspection was an examination of activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examinations of records, documents, discussion with NRC staff, and interviews with licensee personnel.

This report documents an apparent violation which is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600. The current Enforcement Policy is included on the NRC's Web site at www.nrc.gov; select **What We Do, Enforcement**, then **Enforcement Policy**. The apparent violation involved the application of 10 CFR 50.59 for a May 2001 revision to the Oconee Updated Final Safety Analysis Report for the HELB analysis in which the licensee failed to identify Unreviewed Safety Questions and changes involving more than minimal increase in risk which required prior NRC approval. Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the number and characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

An open predecisional enforcement conference to discuss this apparent violation will be scheduled at a future date. The NRC will contact you regarding this date. The decision to hold a predecisional enforcement conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. This conference is being held to obtain information to assist the NRC in making an enforcement decision. This may include information to determine whether a violation occurred, information to determine the significance of a violation, information related to the identification of a violation, and information related to any corrective actions taken or planned. The conference will provide an opportunity for you to provide your perspective on these matters and any other information that you believe the NRC should take into consideration in making an enforcement decision.

You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding the apparent violation is required at this time.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

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Charles A. Casto, Director
Division of Reactor Safety

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

Enclosure: Inspection Report No. 05000269/2004007, 05000270/2004007, and 05000287/2004007

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-269, 50-270, 50-287
License Nos:	DPR-38, DPR-47, DPR-55
Report No:	05000269/2004007, 05000270/2004007 and 05000287/2004007
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	June 20, 2003 - January 21, 2004
Inspectors:	M. Scott, Senior Reactor Inspector
Approved by:	Mark S. Lesser, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000269/2004-007, 05000270/2004-007, 05000287/2004-007; 06/20/2003 - 01/21/2004; Oconee Nuclear Station, Other Activities.

The inspection was conducted by a senior reactor inspector. An apparent violation was identified for making a change to the facility in 2001 that involved Unreviewed Safety Questions and more than a minimal increase in risk, without prior NRC approval pursuant to 10 CFR 50.59. With completion of the Updated Final Safety Analysis Report a (UFSAR) change, the licensee accepted the attendant, non-conforming safety issues contained therein. Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

NRC Identified Findings

Cornerstone: Mitigating Systems

Apparent Violation: The inspectors identified an apparent violation of 10 CFR 50.59 (a)(1) (1999 version of 10 CFR) which states, in part, that the licensee may make changes in the facility as described in the safety analysis report without prior Commission approval, provided the proposed change does not involve an unreviewed safety question (USQ). 10 CFR 50.59 (a)(2) states, in part, that a proposed change involves an USQ if the probability of occurrence or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased, or if it may create an accident different from any previously evaluated.

On May 17, 2001, the licensee made a change to the facility, as described in the Updated Final Safety Analysis Report, Section 3.6.1.3, associated with the High Energy Line Break (HELB) analysis, which involved unreviewed safety questions, and failed to obtain prior NRC approval. The UFSAR Section was changed to increase the maximum initiation time following HELB of Emergency Feedwater from 15 to 30 minutes and of High Pressure Injection from 1 hour to 8 hours (based on referenced reports and analysis). The analysis discussed an increased cycling of pressurizer Safety Relief Valves on steam and water, boiler condenser mode of decay heat removal, and an unapproved computer code for application to HELB, but failed to recognize that such changes may increase the probability of occurrence or the consequences of a malfunction of equipment important to safety or may create an accident different from any previously evaluated. In addition, the change resulted in more than a minimal increase in risk. (Section 4OA5)

This is an Apparent Violation pending results of an enforcement conference.

REPORT DETAILS

Other Activities

4OA5 Other Activities

(Closed) URI 50-269, 270, 287/2003003-02, HELB Accident Scenario Review

Introduction. An apparent violation was identified for making a change to the facility in 2001 that involved Unreviewed Safety Questions and more than a minimal increase in risk, without prior NRC approval pursuant to 10 CFR 50.59.

Description. URI 05000269,270,287/2003003-002, HELB Accident Scenario Review, was identified concerning Oconee Updated Final Safety Analysis Report (UFSAR) Section 3.6.1.3 that was changed on May 17, 2001. The change was made prior to licensee implementation of the revised 10CRF50.59 rule, and therefore was evaluated as to whether or not it involved an unreviewed safety question (USQ). The inspectors determined that the change involved some USQs.

The UFSAR change was associated with the licensee's high energy line break (HELB) analysis for main feedwater piping. The escaping water/steam is assumed to disable the 4160 Volt breakers for at least the motor driven emergency feedwater (EFW) pumps and for the high pressure injection (HPI) pumps, and the automatic initiation of the turbine driven EFW pump. The change was to evaluate a delay in the time allowed for manual restoration of EFW from 15 minutes to 30 minutes and HPI from 1 hour to 8 hours. The EFW time to initiation was extended when it was recognized through operator action validation timelines, that it took longer than 15 minutes to provide EFW from other manual sources. The time extension for HPI injection appeared to be related to the licensee's desire to reduce site manning established to support HPI cable switching to an alternate power source. The resulting effect on plant equipment is that the pressurizer safety relief valves (SRV) are required to cycle an additional 15 minutes for up to 30 minutes until EFW is started and will change from relieving steam to relieving water, reactor coolant system (RCS) subcooling is lost, significant RCS voiding occurs, natural circulation cooling is lost, and decay heat is removed by the boiler condensing mode (BCM) for up to eight hours. Delaying EFW initiation may effect the thermally induced compressive and tensile stresses on the Once Through Steam Generator (OTSG) tubes resulting from refilling a dryed out steam generator. Extended reliance on BCM has several implications which have not been reviewed by the NRC, including computer modeling assumptions, cyclical stresses on the RCS resulting from pressure spikes and vibration, and a possible reactivity excursion upon reinitiating of RCS flow due to diluted boron concentration in the cold leg. The inspectors determined that delaying the initiation of safety systems beyond the existing licensing basis time requirements may increase the probability or consequences of a malfunction of equipment (SRV, RCS boundary) and may create the possibility of an accident or malfunction of a different type than previously evaluated and therefore is a USQ that requires NRC approval.

Safety Relief Valve Concern

The licensee's calculation, OSC-7299, revision 0, and referenced report (Duke Power MDS Report No. OS-73.2) was the basis for the previous recovery times for EFW and HPI (15 minutes and 1 hour, respectively). Revision 1 to the calculation introduced the increased recovery times. The 10 CFR 50.59 evaluation indicated that one of the key assumptions was that the SRVs would lift and reseat with water passing through the valves. Originally, the SRVs would be challenged and would undergo 5 cycles on steam in 15 minutes. With the longer recovery times associated with the May 2001 change, the licensee's calculation indicted that SRVs would undergo 6 challenges on steam and 8 additional challenges with water in 30 minutes. The licensee's safety evaluation stated, "Increasing the time allowed to re-establish these flows has no significant effect on the operation of the associated pumps and fluid systems." It additionally stated, "As described in the Safety Review, this activity only allows additional time before equipment is required to be placed in operation following a postulated HELB. The equipment will be operated in the same manner as it previously was, so there are no new alignments associated with this change. Since the equipment is utilized in the same way as it previously was, the consequences of a malfunction of equipment will not increase."

The inspectors were concerned that the licensee's assumption that the SRVs successfully reseat each time is critical to recovery without core damage. With a stuck open valve and no safety injection, core damage would result. The inspectors determined that the licensee's assumption did not have adequate basis. The SRVs were not designed to function in this manner and have not been adequately demonstrated to perform in this manner. The licensee's evaluation did not address that the increased number of cycles of these valves on steam and the change in medium to water may increase the probability of a malfunction (i.e., sticking open or closed) and create the possibility of an accident of a different type (loss of coolant or overpressure). The evaluation added no more technical content to the discussion on malfunctions.

In the OSC-7299 calculation the two actual SRVs are modeled as one valve. The calculation does not address the impact on the risk importance of this modeling. NRC review of the risk indicates that each SRV lift, and the change of medium from steam to water, has a certain probability of failure. With the increased number of challenges, the overall probability that the SRV(s) will fail open during the event increases. With a failed open SRV, and no HPI, core damage will result. Given an increased SRV failure probability, the delta core damage frequency for the applicable core damage accident sequences stemming from the HELB initiating event was evaluated by the NRC as greater than 1E-6. NEI Guidance 96-07 (paragraph 4.3.1, Example three) indicates that a change to the plant that causes an increase in event frequency exceeding 1E-6 per year would require NRC approval prior to making the change.

The licensee provided additional information to support their position. The licensee considered the SRVs adequate based upon NRC acceptance of the Electric Power Research Institute (EPRI) test program on SRVs, which was completed as part of a Three Mile Island Action Plan item, and a letter from Dresser, the SRV vendor. NRC

Safety Evaluation (SER) Report for NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves for Oconee Unit 1, 2, and 3 (TAC 44600, 44601, 44602) dated July 19, 1989, accepted SRV performance for UFSAR Chapter 15 accident and transient scenarios. Reference 8, (EPRI document NP-2628-SR, PWR and Relief Valve Test Program: Safety and Relief Valve Test Report, December 1982) is the performance test report on power operated relief valves and SRVs. The report states that: “The objective of the ... Program was to perform full scale operability tests on a set of primary system relief and safety valves representative of the those utilized in or planned for use in PWRs ..... The test conditions were selected to be representative of those expected in participating PWRs based on consideration of limiting UFSAR, cold overpressrization, and extended high pressure liquid injection events.”

The EPRI study was conducted in 1981 and included a Dresser model 31739A SRV found at Oconee. The Oconee equivalent SRV was tested with several (4) water lifts and did reseat. It is noted that the valves in the EPRI test had not been exposed to a containment harsh environment for some portion of a fuel cycle prior to being tested, which could influence real valve lifts. Although the Oconee similar valve was tested with consecutive lifts, it had been rebuilt prior to the water lifts. NRC staff opinion is that the testing was very limited and not sufficient to demonstrate the valves would successively re-close with multiple actuations of the subject postulated event beyond the limited lifts addressed in the above-mentioned SER. In the EPRI study, the number of valves tested is not sufficient from a quality assurance standard to demonstrate high reliability. The small EPRI sample size could not reasonably provide justification for valve confidence during consecutive challenges as modeled in the loss of 4160 Volt HELB scenario unique to Oconee.

The inspectors determined that the SRV performance during the HELB scenario is not bounded by the EPRI test program, nor by the SER. The SER accepts the Dresser 31739A safety valve blowdown results because it “does not impede natural circulation due to hot leg voiding.” Limiting transients are identified as Loss of Main Feedwater (LOFW) and Feedwater Line Break (FWLB) (Reference 22 of the SER). Generic plant transients are listed but do not include the loss of 4160 Volt power and TDEFW in a HELB. The limiting transients assumptions indicate a single failure of one emergency feedwater pump, a Loss of Offsite Power (LOOP), no credit for PORV actuation among other things, but does not assume loss of 4160 Volt power, therefore the original scenarios did not address the unique Oconee vulnerability - loss of 4160 Volt power resulting in the loss of HPI. With a LOOP, the standard emergency power is sequenced on and pump injection occurs within a very short time. With these limiting transients, the RCS/pressurizer fills within several minutes and decay heat is being effectively removed a short time thereafter. Natural circulation loss is not jeopardized by the scenarios. With these limiting transients, there is a very unlikely chance of core damage, because even with an SRV failure, HPI and EFW is promptly available. During these scenarios, the SRVs lift 3 to 4 times as opposed to the Oconee unique UFSAR change of May 2001, which challenges the SRVs 14 times, eight of which are with water. The change to Section 3.6.1.3 transient contrasts sharply with above generic limiting scenarios. The SER does not accept SRV performance as the licensee proposed in the HELB evaluation.

The licensee provided a valve vendor memorandum (dated 9-19-03). The licensee did not have this information available for the creation of the May 2001 UFSAR change. Although the memo did provide some information regarding testing on similar valves under various conditions, the various testing information did not represent the revision 1 HELB transient. The new vendor information did not provide a failure rate on the subject SRVs and did not address probabilistic reliability of the valves under water lifts. The vendor information did indicate the valve was capable under two phase lifts (saturated steam, not saturated water). However, the information stated that there has been no cycle testing for the Oconee type valve, outside of the EPRI testing. The inspectors determined the changes in performance required by the SRVs represent a USQ because the probability of occurrence or a malfunction may be increased, or an accident of a different type (loss of coolant) may be created.

Delayed EFW Initiation Concern on Steam Generator Tubes

The inspectors questioned the delayed EFW initiation effects on the once through steam generators (OTSG) under the HELB scenario. The long straight tubes in a Babcock & Wilcox OTSG have existing axial loading analysis, described in the UFSAR (such as Section 3.9), due to temperature differences between the OTSG components (tubes to shell), when refilling a dryed out OTSG. The Oconee Emergency Operating Procedures have limits on temperature changes (stated to be +100 and -60 degree F differential temperatures, tensile and compression, respectively for Units 2 & 3). These limits are assumed to be manageable with immediate injection capability, but with delayed injection timing could be critical to prevent exceeding the limits. In the loss of 4160 Volt HELB scenario, EFW flow would be delayed for up to 30 minutes instead of 15 minutes and HPI injection is further delayed. With delayed injection, cold to hot leg temperature differences are driving this divergence combined with other cooling effects. Thus, the differential temperature across the OTSGs would be a potential concern/consideration in the scenario’s evaluation. Excessive stresses could cause OTSG tubes to crack or rupture. The inspectors were concerned that these issues had not been addressed in the licensee’s evaluative documents. From this the inspectors concluded that this is a USQ because of the possibility that an accident different from any previously evaluated may be created, such as a steam generator tube rupture or excessive accident induced tube leakage concurrent with a HELB.

Boiler Condenser Mode Concern

The introduction of BCM was a new mode of long term (8 hours) decay heat removal added to the UFSAR via the May 2001 change. Per Revision 1 to OSC-7299 calculation, enough volume will be lost from the RCS via the SRVs and Reactor Coolant Pump (RCP) seals leakage such that, significant voiding will occur, natural circulation and subcooling is lost, and BCM is relied upon after 30 minutes into scenario. BCM cooling comes from core decay heat and steam rising up the hot leg and condensing in the OTSG tubes. This condition or mode could persist up to 8 hours under the scenario until RCS is under some form of pressure and temperature control. The licensee used RELAP5 software to model the RCS response, which was different than the model previously used. This computer code has not been approved for use in HELB by the

NRC, nor has the NRC been afforded the opportunity to review the licensee's assumptions. The inspectors were unable to conclude that the RELAP5 modeling reasonably reflected actual conditions, timeline/chronology, and magnitude of changes in the plant during the scenario. The calculation generally indicated that results were very sensitive to EFW initiation timeline and other model/scenario facets.

Within the scenario, the reactor coolant significantly boils and "chugs" as steam is condensed in the reactor coolant components under saturated RCS and increased voided volume conditions, thus creating potential component vibration and multiple pressure spikes that constitute mechanical cycles. There is no practical industry experience with this mode of cooling or impact discussion of the possible induced stress and fatigue to plant equipment. The UFSAR (chapters 3 and 5) addresses dynamic loading and fatigue analysis of primary piping and components. The inspectors were concerned that the licensee's evaluation did not address the challenges to the mechanical components of the plant with many more dynamic cycles present for up to eight hours. For example, UFSAR Sections 3.9.1.1 and Table 5.2 indicate cycle transients that the RCS has been designed to withstand and the HELB/BCM is not indicated. Additionally, Section 3.9.3 discusses design and dynamic loads to meet Section 3.1.33 RCS pressure boundary capability criterion. The 10 CFR 50.59 review of the UFSAR and support documentation did not recognize this potential problem. The inspectors could not determine the effects of operation in the BCM for an extended period on plant equipment. From this the inspectors concluded that this is a USQ because of the possibility that an accident different from any previously evaluated (structural failures and loss of coolant) may be created.

The inspectors questioned the potential for recriticality with BCM. Babcock and Wilcox Nuclear Technologies (BWNT) submitted a potential 10 CFR Part 21 report to the NRC on July 31, 1995. Report number 1995-186 contained a potential safety concern on post small break loss of coolant accident (SBLOCA). The document was not discussed in the 10 CFR 50.59 evaluation. The report provided preliminary information on the potential for core recriticality caused by a reactivity insertion due to moderator dilution while in BCM. The report reads in part, "During the course of a small break LOCA and any other transients that involve the partial loss of reactor coolant system (RCS) inventory, the steam generators may provide an energy sink for part or all of the core decay heat via boiler/condenser operation. In this mode, steam generated in the core through boiling is passed through the hot legs to the steam generators and condensed. The condensate is returned to the core through the cold legs. Because boron volatizes at a concentration substantially below that of the source, the concentration of boron carried with the steam is greatly reduced with the result that the boron concentration downstream from the steam generator is gradually reduced. Evaluation of the boron carryover fraction with the MULTI-Q code shows that the fraction is temperature dependent and limited to ten percent for the conditions of interest." In the report, BWNT had indicated that they would provide additional information at a later date. When asked by the inspectors, the licensee produced a Framatome document, 77-5002260-00 Evaluation of Potential Boron Dilution Following Small Break Loss-of-Coolant Accident. This September 1998 document indicated that with HPI and EFW injection from the onset of small break that a cold leg reactivity insertion on the bump of a RCP would cause a recriticality of 15 to 35 percent with some core damage possible. The document did not discuss the possible mechanical effects of cool, deborated water in the core during recriticality (e.g., potential prompt criticality with steam expansion/rapid core voiding). The inspectors were concerned that due to the timeline differences, injection pump availability variances, and the potential for natural circulation recriticality, the conditions of the 1998 document appear to be sufficiently different than those of the May 2001 HELB transient, that documentation of those differences and the exact potential for recriticality and potential damage must be evaluated. The licensee's evaluation did not address recriticality potential with the HPI return at eight hours. The licensee's stated position was "that although the occurrence of BCM can result in an addition of positive reactivity under natural circulation, the risk of a significant core power excursion leading to core damage is low." The 10 CFR 50.59 Rule in effect at the time the May 2001 evaluation was written did not allow any increase in risk in a facilty change without prior NRC approval. From this the inspectors concluded that this is a USQ because of the possibility that an accident different from any previously evaluated may be created.

Analysis. The finding is not suitable for evaluation using the SDP. The SRV concern alone was evaluated by separate probabilistic risk analysis. Given the increased SRV failure probability alone (assumed failure probability with water medium of 0.1), the delta core damage frequency for the applicable core damage accident sequences stemming from the HELB initiating event was greater than 1E-6. NEI Guidance 96-07 indicates that a change to the plant that causes an increase in event frequency exceeding 1E-6 per year would require NRC approval prior to make the change. The significance of this finding, based on the increase in risk associated with the SRV change alone is low to moderate.

Enforcement. 10 CFR 50.59 (a)(1) (1999 edition) states in part, that the licensee may make changes in the facility as described in the safety analysis report without prior Commission approval, provided the proposed change does not involve an USQ. 10 CFR 50.59 (a)(2) states, in part, that a proposed change involves a USQ if the probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased or if the possibility for an accident or malfunction of different from any previously evaluated may be created.

On May 17, 2001, the licensee made a change to the facility, as described in the Updated Final Safety Analysis Report, Section 3.6.1.3, associated with the High Energy Line Break (HELB) analysis, which involved unreviewed safety questions, and failed to obtain prior NRC approval. Specifically, calculation OSC-7299 was changed to increase the maximum initiation time following HELB of Emergency Feedwater from 15 to 30 minutes and of High Pressure Injection from 1 hour to 8 hours. The analysis discussed increased cycling of pressurizer Safety Relief Valves on steam and water and, boiler condenser mode of decay heat removal, and an unapproved computer code for application to HELB, but failed to recognize that such changes may increase the probability of occurrence or the consequences of a malfunction of equipment important to safety or may create an accident different from any previously evaluated.

The licensee has not placed this issue into the corrective action program, or acknowledged this as a performance deficiency.

This is being treated as an apparent violation (AV), 50-269, 270, 287/2004007-01, Failure to Obtain Prior NRC Approval to a Change to the Facility involving Unreviewed Safety Questions on High Energy Line Break Analysis.

4OA6 Meetings, including Exit

The NRC presented the inspection finding to Mr. Noel Clarkson on January 27, 2004 via a telephone conversation. The licensee acknowledged the finding presented. Proprietary information is not included in this inspection report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee:
N. Clarkson, Regulatory Compliance Manager
D. Garland, Operations Senior Engineer
L. Nicholson, Safety Assurance Manager, Oconee Nuclear Station
G. Swindlehurst, Nuclear Engineering Manager

NRC personnel:

NRR personnel

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-269, 270, 287/2004007-01	AV	Failure to Obtain Prior NRC Approval to a Change to the Facility Involving Unreviewed Safety Questions on High Energy Line Break Analysis

Closed

50-269, 270, 287/2003003-02	URI	HELB Accident Scenario Review