Duke Energy.

PR 50
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OFFICE OF SECRETARY
RULEMAKINGS AND
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December 17, 2007

Secretary, U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
ATTN: Rulemakings and Adjudications Staff

Subject: Comments on Proposed Rule 10 CFR 50, Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, RIN 3150-AI01

Duke Power Company LLC d/b/a Duke Energy Carolinas, LLC (Duke) offers the attached comments relative to the solicitation for public comments regarding proposed rule to provide updated requirements for protection against pressurized thermal shock (PTS) events for pressurized water reactor pressure vessels. This proposed rule was published in the *Federal Register* October 3, 2007, RIN 3150-AI01.

Please address any questions to R. L. Gill at (704) 382-3339.

Duke appreciates the opportunity to provide these comments.

Sincerely,

Thomas P Harrall Jr

Thomas P. Harrall, Jr.

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Duke Energy Carolinas, LLC
Comments on Proposed Rule
Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events
Federal Register October 3, 2007, RIN 3150-AI01

The NRC has completed a series of reports that describe the results of the research project that was undertaken to develop a technical basis to support a risk-informed revision of the Pressurized Thermal Shock (PTS) rule (10 CFR 50.61).

The reports that have been reviewed and that form the basis of the comments (a subset of the reports listed in FR Vol. 71, No. 156, p. 46523, August 14, 2006) are the following:

1) NUREG-1806 Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report, August 2007

2) Letter Report (Un-numbered) Oconee Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA), March 3, 2005

3) NUREG/CR-6858 RELAP5 Thermal Hydraulic Analysis to Support PTS Evaluations for the Oconee-1, Beaver Valley-1, and Palisades Nuclear Power Plants, September 2004

These comments are related to reactor trip events with subsequent main feedwater overfeed in B&W-designed reactors. A review of the above documents indicates that these event sequences have been considered in the PTS PRA report, but the significance of these events with respect to PTS has been missed in the overall integrated methodology. It is unclear in the reports as to how this situation occurred, but a re-evaluation of the significance of these events should be performed to determine any impact on the underlying technical basis for the proposed revisions to 10 CFR 50.61.

Table 3.3 of Reference 2 (the PRA report) has an initiator category labeled "Excessive MFW." The comments in this table indicate the author's expectation that this event sequence is not significant to PTS. It is correctly noted that the event is likely to be more severe at hot zero power. This would be true due to both low decay heat and low main feedwater temperature. This event sequence is carried forward in the PRA report as a type of reactor trip event (Figures 4.18 and 4.49). Section 4.4 of Reference 2 describes the concept of grouping all of the PRA sequences into thermal-hydraulic bins with similar temperature and pressure transient responses. The report states that judgment was used in this binning process when a RELAP5 analyses for a particular event sequence did not exist. Reference 3 does not indicate that any RELAP5 analyses were performed for the Oconee MFW overfeed sequence. The final set of bins is shown in Table 4.27 of Reference 2. From the information provided in Table 4.27 it is not apparent to which bin the reactor trip with main feedwater overfeed event sequence was assigned. Section 6.2.5 correctly describes the overcooling potential of the MFW System. Table 7.7 describes the human failure event (HFE #7 - fail to terminate overfeed) associated with this event sequence. Section 9.3 of Reference 2 summarizes the most significant cut sets for Oconee 1, the reference plant for the B&W design, and the reactor trip with main feedwater overfeed is not among these cut sets. Therefore, the PRA methodology, which did not have the

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benefit of any thermal-hydraulic analysis results for this event sequence, does not identify this event sequence as significant.

Reference 1, the summary report, in Section 6.5.2, describes the reactor trip with main feedwater overfeed event sequence. The report states the following:

> "Another set of failures is overfeeding of the steam generators. As with other cases, the initiating event is the reactor/turbine trip. These cases will result in an overcooling event. The failure could be anything from equipment/component failure to control failure or operator error. Cases have been run where a single steam generator is filled to the top, and the water level is maintained at that level. There are cases where multiple steam generators are filled to the top. Cases were run where the steam generator was filled to the top, then feedwater was stopped and the steam generator was allowed to boil dry."

Section 6.7.2.6.1.1 of the summary report states the following:

> "For secondary-side events, the RCS is rapidly cooled by overcooling to the steam generators but the RCS remains at high pressure and, often, forced flow of coolant through the RCS loops continues. The RCS fluid cools, but the extent of the cooldown is limited because the ultimate heat sink temperature is the saturation temperature at atmospheric pressure, which represents the final state in the secondary coolant system."

The above statements in the summary report, along with the apparent assignment of these event sequences to an insignificant bin in the PRA report, prompt the following concerns:

1) Some main feedwater overfeed cases were run, but there is no indication that any RELAP5 overfeed analyses were performed for the B&W design. The B&W design will overcool more rapidly than other PWR designs because of the once-through steam generators. The initial secondary water inventory is low, and the overfeed will immediately influence the rate of heat transfer. The event progresses to a counter flow water-solid heat exchange process, and the temperature of the primary side cold leg water returning from a steam generator will approach the main feedwater temperature. This low cold leg water temperature along with the cold safety injection water has the potential to severely overcool the reactor vessel. Insights based on overfeed analyses for PWR designs with U-tube steam generators are not applicable to the B&W design.

2) The overfeed events that were analyzed are described as only filling to the top of the steam generator. Perhaps this assumption of a limited duration overfeed is supported by the plant design and/or by operator recovery actions credited by the PRA. A continued overfeed would be more severe relative to PTS.

3) The PRA report considers a zero power (low decay heat) initial plant condition. That initial condition is much more severe for main feedwater overfeed events. Thermal-hydraulic analyses of main feedwater overfeed events should consider this initial condition.

4) This above statement in Reference 1 ". . . the extent of the cooldown is limited because the ultimate heat sink temperature is the saturation temperature at atmospheric pressure " is not correct for a B&W design. The extent of the cooldown for a main feedwater

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overfeed is related to the main feedwater temperature, which will be low at zero power with no preheating, and the primary cooldown will be enhanced by the cold safety injection water.

The significance of the above comments in the overall integrated risk due to PTS for B&W-design plants is not known, but additional consideration of the issues summarized above is warranted. There is a possibility that the conclusions drawn in the references may be incomplete.

From: "Jones, Luellen B" <lbjones@duke-energy.com>
To: <SECY@nrc.gov>
Date: Mon, Dec 17, 2007 8:24 AM
Subject: Duke Comments on Proposed Rule 10 CFR 50 RIN 3150-AI01

Please see attached comments from Duke Energy Carolinas on the above proposed rule.

<<Duke Comments RIN 3150 AI01.pdf>>

CC: "Swindlehurst, Gregg B" <GBSwindlehurst@duke-energy.com>, "Gill, Robert L Jr" <rlgill@duke-energy.com>

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