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SUBJECT: Forwards nonproprietary responses to NRC 940127 RAI re TR DPC-NE-3003, "Mass & Energy Release & Containment Response Methodology."

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June 1, 1994

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Duke Power Company
Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Topical Report DPC-NE-3003; "Mass and Energy Release and Containment Response Methodology"; Non-proprietary Version of Responses to RAI

By letter dated February 16, 1994, responses were provided to the NRC staff's request for additional information regarding the subject topical report. These responses contained information that Duke considers proprietary, and therefore it was requested that the information be withheld from public disclosure pursuant to 10 CFR 2.790. In accordance with the provisions of ¶ 2.790, attached please find a non-proprietary version of the responses provided by the February 16, 1994 letter.

If there are any questions, please call Scott Gewehr at (704) 382-7581.

Very truly yours,

M. S. Tuckman

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Duke Power Responses To The NRC Request For
Additional Information Dated January 27, 1994

Oconee Topical Report DPC-NE-3003-P

Question 1

Figure 2.4-2 (page 2-52) of the topical report presents comparative results of the FATHOMS and CONTEMPT large break LOCA analysis. Please clarify whether the CONTEMPT analysis is a new confirmatory analysis performed using the same input data and assumptions as the FATHOMS analysis, or is the older FSAR analysis dicsussed in Section 2.4.3.

Response

The CONTEMPT analysis shown in Figure 2.4-2 is the old FSAR analysis discussed in Section 2.4.3. The CONTEMPT-generated pressure response is taken from Figure 15-56 in the Oconee FSAR. The FATHOMS result is plotted on the FSAR figure to facilitate comparison.

Question 2

As described in Section 4.4.2 (page 4-15), high point vents are opened at 7200 seconds into the 0.005 ft^2 SB-LOCA analysis. Is the HPV effluent included or accounted for in the break flow for purposes of containment mass and energy release analysis?

Response

The HPV effluent for the 0.005 ft^2 SBLOCA case described in Section 4.4.2 is included in the mass and energy release results. However, since primary-to-secondary heat transfer is recovered for this case, the long-term mass and energy release will be non-limiting. Consequently, the containment response is not specifically analyzed for this case (see pp. 4-16 and 6-28).

Question 3

ANS-56.4-1983, paragraph 3.3, states that "a spectrum of break areas shall be analyzed to assure that the highest primary containment peak pressure and temperature have been determined." Your analyses encompass a single break size. Provide a rationale for concluding that the 34-inch MSL-DEGB containment response is limiting.

Response

Given that the double-ended guillotine break results in the highest break flow rate and there is no liquid carryout credited in this analysis, this break size provides the peak containment pressure. Sensitivity studies with smaller break sizes were performed to determine the limiting case with respect to peak containment temperature. For these smaller break sizes, reactor trip is delayed due to the less severe blowdown. The amount of superheat greatly impacts containment temperature and is a function of hot leg temperature and steam generator pressure. The sensitivity studies show that although the smaller break sizes result in higher hot leg temperatures, break enthalpy is lower than that seen during the first few seconds of the double-ended guillotine break. For a short period of time after initiation of the double-ended guillotine break, the combination of hot leg temperatures near their full power values and rapidly decreasing steam generator pressure results in much higher break enthalpies than the smaller break cases. Thus, the 34 inch double-ended guillotine break is the limiting break size with respect to both peak containment pressure and temperature.

Question 4

Referring to Section 5.3 "SG Pressure" (page 5-5), the lower OTSG pressure is non-conservative for break flow and enthalpy. Have any sensitivity studies been performed to examine its effect, or any compensating bias applied to the results? Explain the reason for the modeling problem.

Response

When initializing a RETRAN model to the desired initial conditions, the user must choose the parameters which are considered fixed and allow other parameters to vary in order to obtain a converged steady-state solution. This approach permits introducing conservative allowances in many initial conditions, but also is constrained by physical reality in that an overall energy balance must exist at time zero. The fixed parameters chosen are usually the ones which are most significant for the analysis (e.g. primary coolant average temperature, coolant flow rate, etc.). Steam generator pressure is one of the parameters that is allowed to vary during initialization in order to achieve the required steady-state energy balance. The value resulting from this initialization process for steam generator initial pressure is 910 psig, which is near the realistic pressure. Therefore, the initial secondary system conditions are consistent with the heat transfer required at this power level. Additional conservatism in this parameter cannot by accommodated by RETRAN and would be non-physical.

Question 5

Referring to Section 5.3 "Steam Generator Operating Level" (page 5-5), explain how the numbers add-up to [ ] lbm and why the numbers are inconsistent with FSAR 15.13.4.

Response

FSAR Section 15.13.4 states that an initial steam generator inventory of 62,600 lbm was used to evaluate the steam line break mass and energy release. However, more current information based on plant operating experience indicates that this assumption is overly conservative. The steam generator inventory of [ ] lbm is based on the assumption that the steam generator downcomer is filled with saturated liquid. Given the volume of the downcomer and the density of saturated liquid at the nominal full power steam generator pressure, a downcomer full of saturated liquid would contain [ ] lbm of water. The mass of water and steam in the tube region of a clean generator at power operating conditions is about [ ] lbm. This would increase the total inventory to about [ ] lbm. However, steam generator fouling will also increase the required tube region inventory. Thus, the assumption of [ ] lbm includes an allowance of [ ] lbm to account for the impact of steam generator fouling on the overall primary-to-secondary heat transfer coefficient.

Steam generator fouling impacts inventory in two ways. First, steam generator fouling increases the frictional pressure drop in the tube region of the steam generator. Thus, a higher level in the downcomer is needed to offset this increased frictional pressure drop. The assumption of a downcomer full of saturated liquid clearly bounds the impact of steam generator fouling on the downcomer inventory. Second, steam generator fouling decreases the overall primary-to-secondary heat transfer coefficient. In order to remove the same amount of energy, the heat transfer surface area, or boiling length, must be increased. An increase in the boiling length increases the tube region inventory. Discussions with B&W steam generator experts indicate that this effect does not appreciably impact the total steam generator inventory. Thus, the total inventory of [ ] lbm should be adequate to conservatively account for the impact of steam generator fouling.

Question 6

Regarding "Fission Heat," in Section 5.4 (page 5-7), are *all* or *n minus 1* rods assumed to insert?

Response

The most reactive control rod is assumed to remain in the fully withdrawn position to maximize the return to power due to the cooldown of the RCS (i.e., n-1 rods are assumed to insert on reactor trip).

Question 7

Regarding "Limiting Single Failure," in Section 5.4 (page 5-11), identify what other single-failures were considered. Indicate whether the proposed plant modifications intended to eliminate the operator action requirement to terminate FW addition involve or could create new single-failure concerns.

Response

The two cases presented in DPC-NE-3003-P assume that the ICS is in automatic control with the MFW control valve functioning properly. These cases are presented primarily to demonstrate the analysis methodology. In order to alleviate the reliance on operator action, a plant modification will be made to automatically initiate feedwater isolation during a steam line break. The design of the feedwater isolation system has not yet been completed. Thus, at the present time the limiting single failure cannot be identified, nor is it known if any new single failure concerns will be created by the feedwater isolation system. A single failure analysis will be performed and these concerns will be addressed as part of the design of the feedwater isolation system.

Question 8

Regarding "ICS" (page 5-11) in Section 5.4, please provide additional justification or rationale for neglecting the effect of rod motion.

Response

Following larger steam line breaks but prior to the resulting reactor trip signal, the combined effect of decreasing turbine header pressure and T-ave results in the ICS increasing reactor demand to the high limit of about 103% FP. Since reactor trip occurs within the first five seconds of the accident and the control rods move at a constant speed of 30 inches per minute, the rods would at most move about 3%. This amount of rod movement would result in a negligible reactivity addition. Thus, it is reasonable to assume that the control rods are in manual control.

For the sensitivity studies with smaller break sizes, the combined effect of decreasing turbine header pressure and T-ave would result in an increase in reactor demand to the high limit of about 103% FP. The nuclear instrumentation (NI) flux error is the difference between actual and indicated power level due to the effect of the reactor vessel downcomer temperature on the excore NI flux detectors. When downcomer temperature decreases, the indicated power level will be less than the actual power level. Examination of the actual power level response and NI flux error for the smaller break cases shows that the indicated power level would increase well above 103% FP due to the cooldown of the Reactor Coolant System. Thus, control rod insertion would occur with the ICS in automatic. Therefore, it is conservative to assume that the control rods are in manual. The NI flux error is conservatively accounted for in the Reactor Protective System by increasing the high power and flux/flow trip setpoints.

Question 9

Regarding MSLB containment analyses, explain if and how revaporization is considered?

Response

The standard interfacial mass and heat transfer equations for the superheated vapor phase in FATHOMS are used to determine the amount of condensation remaining in the droplet phase (that is, remaining in the vapor region as droplets). The heat transfer rates are dependent on drop concentrations, relative velocities between the phases, and several other factors. These equations are given on p. 34 of NUREG/CR-3262, Vol. 1 (Reference 2-20 of DPC-NE-3003-P), which is the COBRA-NC code equations manual. The COBRA-NC equations manual serves as the basis for the FATHOMS code, and therefore these heat transfer equations are the same as in FATHOMS. As the liquid film thickness on the surfaces of the containment walls and heat structures increases, some of this film may be vaporized by the superheated atmosphere. This is all calculated in FATHOMS using the equations referenced above. No additional algorithms to model this heat transfer are applied in any of the FATHOMS analyses.

In the FATHOMS code, separate mass and momentum equations exist for the droplet and vapor phases, but only one set of energy equations exists for the liquid/droplet phases. Therefore, the droplets and continuous liquid phases are always at the same temperature within a single calculational volume. For this reason, the Oconee FATHOMS model contains a separate node for the sump region. Within the atmosphere region, very little liquid will exist in the continuous liquid phase. The temperatures of this liquid film and the droplets in the atmosphere should be very close, so this modeling technique is applied.

Question 10

Section 6.4.6.5 (page 6-44) of the topical report states that SB-LOCAs require a reduction in the containment spray actuation setpoint and opening of the boron dilution flowpath for acceptable containment response. Please explain the extent to which these requirements have been implemented.

Response

A reduction in the Technical Specification value for the spray initiation setpoint is required to keep the Reactor Building temperature within EQ requirements for some SBLOCA scenarios. This reduction is from 30 psig to 20 psig. Even though the Technical Specification setpoint is 30 spig, the actual setoint used at Oconee is 10 psig. Therefore, the impact of the reanalyses on the spray actuation setpoint does not have any real safety impact on current operations. It was decided not to formally request approval of a Technical Specification change until the methodologies in DPC-NE-3003 are approved by the NRC.

Current emergency operating procedures (EOPs) require the opening of the boron dilution flowpath within 9 hours following a large break LOCA. For conservatism, this flowpath was not taken credit for in the large break LOCA containment analyses until 24 hours. In the SBLOCA analyses opening of the boron dilution flowpath is credited at 15 hours. Current EOPs do not instruct the operator to open the boron dilution line following a SBLOCA. This guidance will be added to the EOPs following approval of DPC-NE-3003-P. In the interim it is concluded that the time available to perform this action is sufficient to enable the engineering staff in the emergency response organization to assume the responsibility for this action.

Question 11

Section 6.4.5 (page 6-35) of the report indicates a spray initiation setpoint of 20 psig (plus delay). Section 6.5.5 (page 6-47) indicates 30 psig (plus delay). Please clarify the spray initiation setpoint.

Response

As mentioned above in the above response to Question #10, the Technical Specification spray initiation setpoint must be lowered from 30 to 20 psig for some SBLOCA scenarios. In the LBLOCA and MSLB containment analyses, the existing 30 psig setpoint was assumed since this value had no unacceptable impact on the results of either analysis.

Question 12

The EQ envelope depicted in the report for MSLB (e.g., figures 6.5-2, 6.5-5) is different than that depicted for LOCAs (e.g. figures 6.3-4, 6.4-1). Explain the discrepancy. Also, indicate what "case-by-case" analyses have been performed to confirm the acceptability of the MSLB responses with respect to EQ requirements.

Response

The EQ envelopes for the MSLB and LOCA figures of Chapter 6 of DPC-NE-3003 are the same. From 0 to 100 seconds, the EQ requirement is 312°F, and from 100 to 1500 seconds, the requirement is 290° F. The may appear different due to the LOCA figures not including the first 100 seconds. During the time period from 0-100 seconds, the EQ envelope is not challenged by the LOCA and was not plotted in order to provide better resolution after 100 seconds. Figure 6.2-12 shows that the peak LOCA temperature in the 0-100 second time period is 285°F. This is well below the 312°F EQ envelope limit during that period of time.

Upon the completion of DPC-NE-3003-P, Duke Power performed an engineering calculation to demonstrate the impact of the new MSLB peak containment temperature analysis result on the safety-related equipment located inside containment. All equipment required to mitigate and/or monitor MSLB was included in this evaluation. To demonstrate the adequacy of the equipment to perform its safety function, two "worst-case" pieces of equipment were selected. The "worst-case" equipment are the Viking penetration and BIW cable. "Worst-case" is defined as the equipment exposed to the lowest test temperature for the shortest period of time during the LOCA test program. The test temperatures for these cases were plotted and compared to the containment temperatures obtained for the MSLB to demonstrate the insignificance of those time periods for which the MSLB curve is above the tested curve when compared to the temperatures and duration of the testing. The calculation utilizes the results and conclusions of an analysis performed by

Babcock & Wilcox using a two dimensional finite element model. This model showed that even for containment temperature reaching almost 500°F during MSLB, the temperature of the equipment internals was significantly lower than for a LOCA due to the brief period of time during which the equipment was exposed to elevated temperatures. These analyses were done for transmitters, motor operators, electrical penetrations, instrument enclosures, and cable jackets. Based on these engineering calculations, it was concluded that the MSLB temperature response did not have an unacceptable impact on safety-related equipment inside containment.