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DESCRIPTION	ENCLOSURE
Ltr. re our 11/23/76 ltr....trans the following: (1-P) PLANT NAME: Cook 1 & 2	Furnishes requested additional information regarding Exxon Nuclear Company Report XN-76-51, concerning information supporting the operation of the Cook Plant Unit 1 for fuel Cycle 2. (9-P) DO NOT REMOVE ACKNOWLEDGED

SAFETY — FOR ACTION/INFORMATION — ENVIRO 12/9/76 RJL

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NRC FORM 195 (2-76)

EXXON NUCLEAR COMPANY, Inc.
2101 Horn Rapids Road, Richland, Washington 99352
PHONE: *(509) 946-9621*

50-315

50-316

Regulatory Docket File

November 30, 1976

Mr. Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dennis:

In your letter to Mr. John Tillinghast dated November 23, 1976 you requested additional information be supplied regarding Exxon Nuclear Company Report XN-76-51. This report provided information supporting the operation of the D. C. Cook Nuclear Plant Unit #1 for fuel Cycle 2.

This letter transmits responses to this request for additional information for your review. One copy of these responses is being transmitted via telecopier; forty (40) copies are being transmitted under separate cover.

Very truly yours,

Gerald Owsley

G. F. Owsley, Manager
Reload Licensing

GFO:gf
Attachments
As above

CC: Mr. John Tillinghast

12424

ADDITIONAL INFORMATION FOR D. C. COOK

QUESTION 1

The nodalization diagram shown on Figure 5.1 of XN-76-36, for the RELAP reflood calculation uses an axially split downcomer with accumulator and Safety Injection System (SIS) input to the downcomer regions. Provide additional description and justification for the use of this nodalization and water injection assumption relative to the previous ENC-WREM reflood model.

RESPONSE

As discussed in XN-76-36, the RELAP4-EM/FLOOD model used for the D. C. Cook analysis used two volumes to represent the downcomer region. This nodalization is a more realistic representation of the actual reactor system than the previously used single-volume downcomer, in that, a significant cross-sectional area change occurs in the downcomer region. The lower downcomer node includes the regions on both sides of the thermal shield, the region between the core barrel and the core baffle, and the core bypass. The total flow area for this region is 49.53 ft^2. The upper downcomer region includes the volume between the reactor vessel and the core barrel with a flow area of 32.41 ft^2. Since the liquid height in the downcomer equals the liquid volume divided by the horizontal flow area, a change in flow area will alter the liquid height; and hence, the driving head for reflood. The two-volume downcomer permits the area change to be modeled for the D. C. Cook reactor.

The Safety Injection System (SIS) modeling for the D. C. Cook reflood calculations is described in XN-76-36. Initially SIS flow and accumulator flow are modeled as a fill system injecting into the lower downcomer region with fluid conditions near saturation (saturation at lowest containment pressure). This model is consistent with the approved ENC-WREM PWR model. Pressure drop penalties are applied at the intact loop junction to the pressure vessel to account for interaction effects of ECC fluid and superheated steam in the cold leg pipes. This model is in accordance with the approved ENC-WREM PWR model as described in XN-75-41, Supplement 5, Revision 1.

When steam flow is established in the intact loops, the SIS flow is switched from the lower downcomer to the upper downcomer region and is input at the actual fluid temperature of the SIS water. At the time steam flow is established in the intact loops, the steam flow and enthalpy is sufficient to heat the SIS fluid to saturation by condensing some of the intact loop steam flow. Since SIS fluid and intact loop steam must flow through the same pipes, mixing of these fluids is expected. Thus, the assumption of homogeneous equilibrium inherent in the RELAP4-EM/FLOOD program realistically represents the expected conditions and conservatively minimizes the steam flow from the upper downcomer to the containment. This results in a minimum pressure drop at the steam slip flow junctions (to the containment) and a conservative reflood system pressure.

QUESTION 2

A fluid temperature in the upper head region equal to the hot leg temperature shall be used unless a lesser temperature is justified by actual measurements in a similar plant or unless a lesser temperature results in a higher peak cladding temperature than the hot leg temperature.

RESPONSE

The upper head fluid temperature sensitivity study has been completed. The base case consisted of the 1.0 DECLS reported in XN-76-51, the limiting break for the D. C. Cook Nuclear Unit 1 with the upper head temperature set to the hot leg temperature. A second calculation was performed, identical to the first, except the upper head temperature was set equal to the average of the hot leg and cold leg temperatures. This results in a decrease in cladding temperature at EOBY of 52°F and decrease in volume averaged temperature at EOBY of 47°F. This results in an approximate 24°F decrease in peak cladding temperature if the lower value of upper head temperature was used. Thus, the use of the hot leg temperature for the upper head is conservative.

QUESTION 3

Describe and justify the phase separation model assumed for the upper head during blowdown:

RESPONSE

A phase separation model was input to the upper head region using the available RELAP4 bubble rise model. The use of the phase separation model is justified in that the upper head region is relatively stagnant and thus phase separation is expected to occur. Also, the flow path from the upper head to the upper plenum occurs at the top of the control rod guide structure and the possibility exists that when the mixture level falls below this level, only steam will flow to the upper plenum with the remaining upper head liquid being held up in the upper head region and thus unavailable for core cooling during blowdown. A phase separation model is necessary to consider these effects.

Modeling the upper head with phase separation model is a more conservative representation than a homogeneous model and is also a more realistic representation of this region. The parameter values used for the phase separation model are derived empirically for blowdowns from nearly stagnant vessels similar to the upper head region. A sensitivity study was performed for D. C. Cook system which confirmed that the separated upper head model resulted in a higher PCT than a comparable calculation using a homogeneous model.

Upper head nodalization studies have been completed. The first case was reported in XN-76-36, as the 4-loop ice condenser sample problem and was done with a homogeneous model and with the upper head temperature set to the hot leg temperature. A second calculation has been performed identical to the first, except the upper head was modeled as a separated volume. This results in less water available for core cooling during the blowdown phase since some of the upper head water is trapped in the upper head, below the top of RCC assembly guide tubes.

The reduced cooling causes an increase in cladding temperature at EOBY of 142°F and an increase in fuel averaged temperature at EOBY of 90°F. This difference results in an approximate 45°F increase PCT due to the use of a separated rather than a homogeneous volume.

<u>*QUESTION 4*</u>

Provide a calculated effect on peak cladding temperature for each of the following model changes incorporated in the ENC-WREM-II model:

A. *Flow Blockage*

B. *FLECHT/ENC3 multipliers*

C. *Hot wall delay*

D. *Steam cooling*

E. *Reflood model-downcomer nodalization (#1 above)*

RESPONSE

As requested by the NRC Staff, ENC is providing the change in PCT resulting from each of the model changes comprising ENC-WREM-II, as well as the total change in going from ENC-WREM-I to ENC-WREM-II. These were originally provided in Reference 1, but the model used for these calculations has been modified, resulting from NRC Staff review, thus, a new sensitivity study was performed. The sensitivity studies are based on the 1.0 DECLS for the D. C. Cook Unit 1 nuclear plant reported in Reference 2 except the maximum LHGR is 12.14 kW/ft rather than 13.68 kW/ft. It is important to recognize that if the sensitivity study had been performed for a plant without an ice condenser containment system (H. B. Robinson or Palisades), the only significant effect of the new model (ENC-WREM-II) would result from the improved hot wall delay model and the ΔPCT would be 45 to 60°F.

Table 1 summarizes the results of the requested sensitivity studies. It is to be noted that the ENC-WREM-I is a conservative model which was never intended to be used for plants with extended period of reflood rates less than 1.0 inch/sec. Note that when ENC-WREM-II is applied to this calculation, rupture is not calculated to occur. In order to make a realistic comparison between the two models, the rupture temperature was reduced about 50°F. This causes rupture to occur and the code to switch to the steam cooling model. By forcing the code to switch to the steam cooling model, representative comparisons can be made between the old and new models. However, the data for the calculations in which rupture was not forced is included for completeness.

The reduction in PCT due to ENC-WREM-II is about 540°F for the D. C. Cook plant. As indicated by the table, the temperature decrease is due mainly to the improved calculation of flow around the blockage and to the FLECHT/ENC3 multipliers. This is not unexpected since the ENC-WREM-I model assumed an 80% flow area reduction (maximum calculated flow area reduction) near the blockage; but only a 23% area reduction is calculated to occur. ENC-WREM-II takes into account the actual amount of blackage* which is calculated to occur; thus, ENC-WREM-II calculates a more realistic flow at and downstream of the blockage. The improved low flood rate FLECHT heat transfer is based on the expanded data base available in Reference 2 which was not available during ENC-WREM-I development. This data showed improved heat transfer during the initial 50 seconds of the transient relative to the original data.

Although the ENC-WREM-II steam cooling model with the ENC3 multipliers conservatively predicts 100% of the low flood rate low pressure data between the 4 and 8 foot elevations, the steam cooling model change led to an increase in cladding temperature as compared to the previous model.

The 90°F reduction in PCT due to the hot wall delay model change is caused by the reduction in the ECC water spilled during the refill period. This water becomes available for cooling the core and for filling the downcomer.

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* This flow area reduction due to ballooning calculated with the ENC flow blockage model approved in ENC-WREM-I.

The axially split downcomer properly accounts for the change in the cross-sectional area of the downcomer at the top of the thermal shield. This allows for an accurate calculation of the liquid height in the downcomer. Since the liquid velocity in the downcomer is small, the additional inertial and frictional terms are negligible; thus, it only affects the calculation through the determination of the liquid height. For the D. C. Cook case, the downcomer is filled by the accumulators very early in the transient and the downcomer remains filled for the remainder of the transient. Therefore, this change has no effect on PCT for D. C. Cook.

TABLE 1

ENC-WREM-II MODEL CHANGE SENSITIVITY STUDIES

(ICE CONTAINMENT REACTOR)

	ENC-WREM-1	Hot Wall Delay	ENC3 Multipliers		Blockage	SC Model	ENC-WREM-II	
PCT (°F)	2453.	2363.	1727.[1]	2250.[2]	1999.	2629.	1723[1]	1905.[2]
Time of PCT (sec)	344.	332.	342.	350.	274.	366.	336.	282.
Location of PCT (ft)	7.1	7.1	8.9	7.4	7.1	7.1	8.9	7.4
Time of Rupture (sec)	120.	129.	----	131.	120.	120.	----	143.
Location of Rupture (ft)	6.9	6.9	----	7.1	6.9	6.9	----	7.1
Max. ZrO_2 (%)	11.5	9.4	1.40	6.9	3.7	15.9	1.36	2.6
Location Max. ZrO_2 (ft)	7.1	7.1	7.9	7.4	7.1	7.1	7.9	7.4
ΔPCT	----	-90.	-726.	-203.	-454.	+176.	-730.	-548

1 No rupture.
2 Forced rupture.

REFERENCES

1. Worley, L. C., Rowe, D. S., and Galbraith, K. P., "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC-WREM-II", XN-76-27, July, 1976.

2. "Donald C. Cook Unit 1 LOCA Analysis Using the ENC WREM-Based PWR ECCS Evaluation Model (ENC-WREM-II)", XN-76-51, October 1976.