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SUBJECT: Discusses preliminary accident sequence precursor analysis response to NRC request for licensee peer review.

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Donald C. Cook Nuclear Plant Unit 2
Docket No. 50-316
License No. DPR-74
PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS
RESPONSE TO NRC REQUEST FOR LICENSEE PEER REVIEW

U. S. Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attn: W. T. Russell

June 28, 1994

Dear Mr. Russell:

In a letter dated May 19, 1994, the NRC requested a licensee peer review of a preliminary Accident Sequence Precursor Program analysis. This analysis was based on Licensee Event Report Number 316/93-007, which summarizes an August 2, 1993, trip of Donald C. Cook Nuclear Plant Unit 2.

On August 2, 1993, Cook Nuclear Plant Unit 2 reactor tripped as a result of a main turbine exhaust hood high temperature trip. Two abnormalities occurred following the trip, which led to the evaluation of the reactor trip as a potential severe core damage accident precursor. First, the feedwater valves from the east motor driven auxiliary feed pump throttled further than expected after receiving a flow retention signal, requiring operator action to establish correct flow rates. Second, the main steam isolation valves started drifting closed, requiring operator action to reopen the valves and maintain normal steam flow. Both of these abnormalities contributed to a core damage probability estimate of 6.4×10^{-6} in the preliminary Accident Sequence Precursor Program analysis. Comments on the analysis of these two abnormalities are provided below.

The feedwater valves from the east motor driven auxiliary feedwater pump throttled further than expected, reducing the flow to the two steam generators that it supplies. The flow was corrected by the reactor operator as required by the reactor trip response procedure.

In the Accident Sequence Precursor analysis, this reduced flow situation was appropriately treated as a pump failure, since the pump could not provide its required flow without intervention. A

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general non-recovery factor was then applied to the modified failure rate of the auxiliary feedwater system, which includes all three auxiliary feedwater pumps. However, since the specific failure mechanism for the auxiliary feedwater pump is known, a non-recovery factor appropriate to this failure mechanism should be used. A non-recovery factor of .04 should be applied to the specific auxiliary feedwater pump failure, ("ASP Models," Appendix A to NUREG /CR-4674, Volume 17, page A-5), since the recovery action occurs in the control room and is directed by the reactor trip response procedure (2-OHP 4023-E-0). Recovery of this pump was successful in the actual trip event. This can be easily accomplished by modifying the random auxiliary feedwater pump failure probability (.02) to be the random probability (.02), plus the product of the event specific failure probability (1.0) and the event specific non-recovery factor for this pump (.04), for a result of .06. This maintains the original random failures and non-recovery factors appropriate to the sequence while adding the event specific auxiliary feedwater failure and appropriate non-recovery factor.

The second abnormal occurrence was the main steam isolation valves drifting off their backseats. If all four valves were to eventually shut, the steam supply to the main feed pumps would be lost, requiring additional actions to restart the main feed pumps should all auxiliary feedwater fail.

In the preliminary Accident Sequence Precursor Program analysis, this was assessed as a penalty against the non-recovery factor for the main feedwater system.

Although not noted in the Licensee Event Report, only two of the four main steam isolation valves were drifting off their backseats. Had both of these eventually closed, steam still would have been available to supply the main feedwater pumps. Therefore, no penalty against the non-recovery factor is appropriate.

Given the comments provided above, we expect the new core damage probability will be 4.5×10^{-7}. Other than the two comments provided above, the preliminary Accident Sequence Precursor Program analysis accurately describes the reactor trip event.

Sincerely,

for W. G. Smith

E. E. Fitzpatrick
Vice President

cad

cc: A. A. Blind
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