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DCP/NRC0990
NSD-NRC-97-5274
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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: T. R. QUAY

SUBJECT: RESPONSES TO COMMENTS ON THE AP600 STANDARD SAFETY ANALYSIS
REPORT (SSAR) CHAPTER 15 ACCIDENT ANALYSES

Reference: Letter from William C. Huffman to N. J. Liparulo, "Comments on the AP600
Standard Safety Analysis Report (SSAR) Chapter 15 Accident Analyses," dated
1/21/97

Dear Mr. Quay:

Reference 1 provided NRC comments on Chapter 15 of the AP600 SSAR. Attached are responses to
four of the comments on the Chapter 15 analyses. These responses close, from a Westinghouse
perspective, the following open items:

OITS-4480, Comment 16
OITS-4481, Comment 17
OITS-4485, Comment 21
OITS-4486, Comment 22

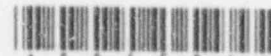Please contact Ms. Susan V. Fanto (412)374-4028, if you have any questions concerning this material.

Susan V Fanto for

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

cc: W. C. Huffman, NRC
N. J. Liparulo, Westinghouse (w/o Attachment)

E0041/1

ATTACHMENT 1
TO WESTINGHOUSE LETTER DCP/NRC0990
Comment Responses

Loss of Normal Feedwater Flow (SSAR 15.2.7)

Comment 16 - (OITS-4480)

Provide a DNBR transient curve for a loss of normal feedwater flow event.

Response:

Prior to reactor trip and insertion of the rods into the core, the loss of normal feedwater transient is the same as the transient response presented in SSAR subsection 15.2.6 for the loss of ac power to plant auxiliaries. The DNB results presented in Figure 15.2.6-12 for the loss of ac power to plant auxiliaries are applicable for a loss of normal feedwater and demonstrate that the DNB design basis is met.

Loss of normal Feedwater Flow (SSAR 15.2.7)

Comment 17 - (OITS-4481)

This analysis does not address compliance with the GDC 17 requirements. To satisfy GDC 17, the effects of a loss of offsite power on the loss of normal feedwater flow event should be considered.

Response:

Analysis is presented in the SSAR which addresses compliance with the GDC 17 requirements related to a loss of offsite power during a loss of normal feedwater flow event. The electrical grid disruption is assumed to be a consequence of tripping the reactor and turbine during the loss of normal feedwater event. Thus the loss of offsite power is assumed to occur following turbine trip during the loss of normal feedwater flow event. The impact of the loss of offsite power is to cause a coastdown of the reactor coolant pumps. Postulating a of loss normal feedwater flow event followed by a consequential loss of offsite power after reactor trip is the same scenario as is presented in SSAR Section 15.2.6 for the loss of ac power to station auxiliaries analysis. Therefore, the analyses presented in Revision 13 of SSAR Section 15.2.6 have been updated so that they are now applicable for the loss of feedwater flow case, which assumes a loss of offsite power.

Feedwater System Pipe Break (SSAR 15.2.8)

Comment 21 - (OITS-4485)

The DNBR results for this event should be included in the SSAR. The calculated minimum DNBR should be greater than the safety limit DNBR for acceptance.

Response:

As summarized in the response to Question 19, the methodology used for the AP600 feedline break analysis uses conservative assumptions such that interactions between the feedline break and the feedwater control system initially result in no feedwater flow being delivered to or from either steam generator. This assumption maximizes heatup of the RCS prior to reactor trip. At reactor trip, assumptions for the break are modified such that all remaining liquid in the faulted steam generator is rapidly blown down through a full double ended break to minimize cooling effects on the RCS. Offsite power is also assumed to be lost at the time of reactor trip resulting in a coast down of the reactor coolant pumps. The initial system response for the limiting feedline break analysis is similar to the loss of ac power to station auxiliaries event presented in the SSAR Section 15.2.6. The DNB results presented in Figure 15.2.6-12 of the SSAR for the loss of ac power to plant auxiliaries, are also applicable to a feedwater system pipe break. The results shown in Figure 15.2.6-12 of the SSAR demonstrate that the DNB ratio remains above the design limit and that no fuel failures occur for feedline breaks.

The acceptance criteria for this event does allow the DNBR to be below the design limit. Fuel failure is acceptable if radiological release criteria are met. As per ANS N18.2, feedwater system pipe ruptures are considered Condition IV Limiting Faults. The acceptance criteria specifies Condition IV faults shall not cause a release of radioactive material that results in an undue risk to public health and safety exceeding the guidelines of 10 CFR 100. Similarly section 15.2.8 of the Standard Review Plan (NUREG-0800) states "if the DNBR falls below these values, fuel failure (rod perforation) must be assumed ..." and also states that "Any activity release must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines."

Feedwater System Pipe Break (SSAR 15.2.8)

Comment 22 - (OITS-4486)

The Semiscale test data for feedwater line breaks (as discussed in Section 4.3.3.2 of NUREG/CR-4945, dated July 1987) showed that the steam generator heat transfer capacity remains unchanged until the steam generator liquid inventory is nearly depleted. This is followed by a rapid reduction to 0 percent heat transfer with little further reduction in the steam generator water inventory. In light of these test data, provide a discussion of the steam generator heat transfer model used in the feedwater line break analysis and verify the model is conservative as it is compared with the Semiscale test data. If the model is found to be nonconservative, reanalyze the feedwater line break event by using the model that is supported by the test data including Semiscale test data. With a heat transfer model consistent with the Semiscale test data, perform a sensitivity study of break sizes to identify the worst break size and provide the results for the staff to review.

Response:

Section 4.3.3.2 of NUREG/CR-4945 (Reference 1) states:

> "*The normalized heat transfer versus normalized liquid mass (normalized to initial values) for the three tests are shown in Figure 64. For the 100% and 50% break test (S-FS-6 and S-FS-11), the heat transfer remained at 100% until the liquid mass reached about 5 to 10%. The heat transfer then reduced to about 90% over the next 5% reduction in liquid mass followed by a rapid reduction to 0% heat transfer. For the 14.3% break test (S-FS-7), the heat transfer remained at 100% until liquid mass reached about 20% followed by a rapid reduction to 0% heat transfer starting at 8% liquid inventory. Although a slight break size dependency is indicated by these results, the basic trend is very similar: the heat transfer remains at nearly 100% until the liquid inventory is nearly depleted. This is followed by a rapid reduction to 0% heat transfer with little reduction in mass. The Combustion Engineering (CE) FSAR Appendix 15B[221] assumes for a feed line break that there is 100% heat transfer until the liquid inventory is depleted, followed by a step change reduction in the heat transfer to 0%, which is nonconservative based on Semiscale scaled results.*"

Attached is Figure 64 from NUREG/CR-4945 which shows the Semiscale steam generator heat transfer as a function of steam generator liquid mass.

The AP600 feedline break analysis is performed using a modified version of the LOFTRAN code (Reference 2 & 3). The steam generator heat transfer model used does not assume 100% heat transfer until the liquid inventory is depleted, followed by a step change reduction in heat transfer to 0%.

Degradation of the steam generator heat transfer is calculated by LOFTRAN as the steam generator inventory decreases. The model and equations used to calculate degradation of steam generator heat transfer as steam generator inventory is depleted are described in Section 4.3 of Reference 2. The LOFTRAN model predicts similar heat transfer phenomena to that observed in the Semiscale tests.

Figure 1 shows the normalized faulted steam generator heat transfer as a function of normalized steam generator liquid inventory predicted during the AP600 feedline rupture analysis. The predicted heat transfer is approximately 100% until the liquid mass reaches approximately 11%. Below 11% mass, the heat transfer rapidly drops to 0%. The AP600 LOFTRAN predicted heat transfer characteristics agree very well with the Semiscale test results.

References:

1   G. G. Loomis, "Summary of the Semiscale Program (1965 - 1986)," NUREG/CR-4945, EGG-2509, July 1987

2   T. W. Burnett, et.al., "LOFTRAN Code Description," WCAP-7907-P-A, April 1984

3   E. L. Carlin, "LOFTRAN & LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1, August 1997
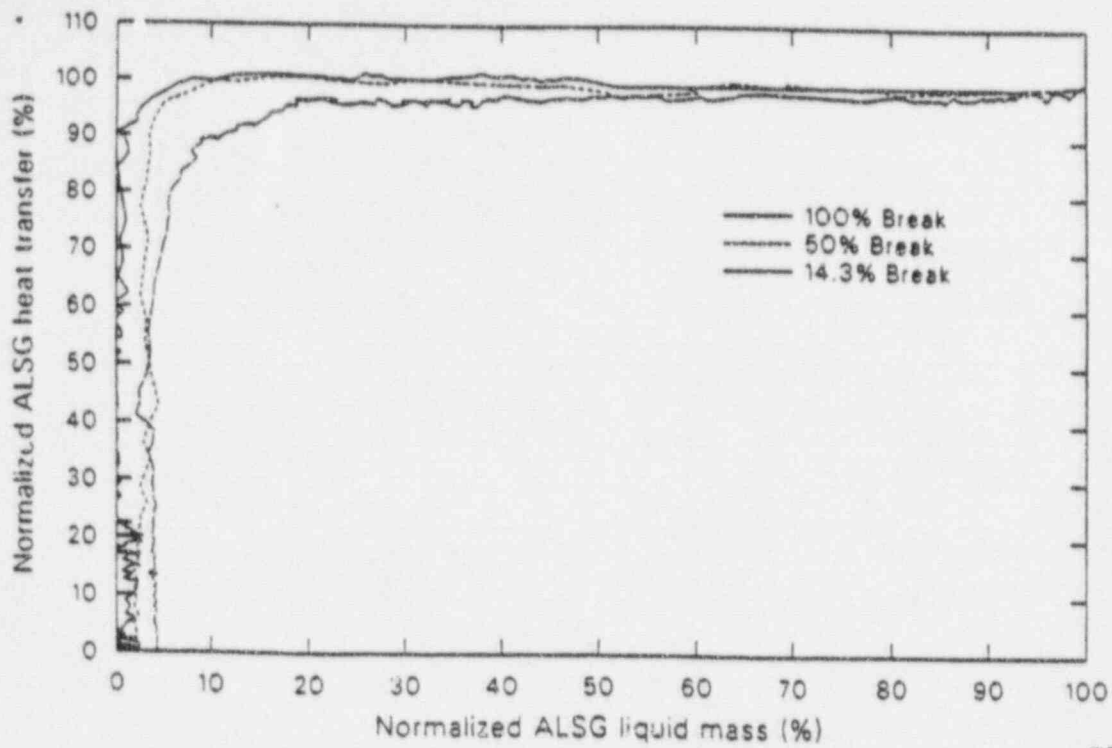
Figure 64. Affected loop steam generator normalized heat transfer versus normalized liquid mass for Tests S-FS-6, S-FS-11, and S-FS-7.
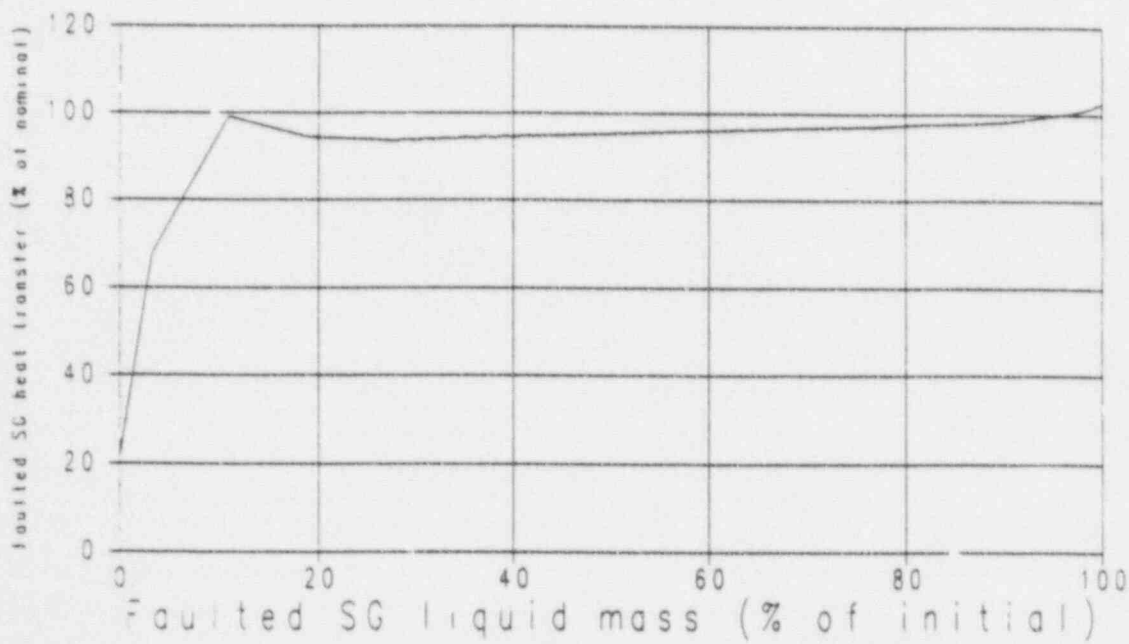

Figure 1   Faulted steam generator normalized heat transfer versus normalized liquid mass as predicted by LOFTRAN for the AP600 feedline rupture