September 28, 2001

MEMORANDUM TO: Roy Zimmerman, Deputy Director
Office of Nuclear Regulatory Research

FROM: Thomas L. King, Director **/RA/**
Division of Systems and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: COMPLETION OF SUBTASK MILESTONE IN STEAM GENERATOR ACTION PLAN

The purpose of this memorandum is to record the completion of a subtask milestone in the Steam Generator Action Plan that is associated with the Steam Generator (SG) Differing Professional Opinion (DPO). This subtask is item 3.4a, entitled, "Perform system level analyses to assess the impact of plant sequence variations (e.g., pump seal leakage and SG tube leakage). The due date for this item is listed in the SG Action Plan as October 1, 2001. This item is also identified in the RES Operating Plan under item 1A1AF, "Maintaining adequate safety margins associated with steam generator tube integrity."

The overall objective of this task is to develop a better understanding of the conditions seen by the steam generator tubes (and other RCS components) during severe accidents. Potential steam generator tube rupture during severe accidents is risk important because the ruptured tube presents a release path for fission products which bypasses the containment. This work is in response to the explicit recommendation of the ACRS Ad Hoc Subcommittee on a DPO. The technical objective of this specific subtask is to investigate the importance of different accident sequence variations in establishing the pressure and temperature environments seen by the steam generator tubes during severe (core melt) accidents. Typically, as a result of past studies, the staff has primarily focused on the sequences which intuitively would represent the worst conditions for the tubes, i.e., the largest pressure differential across the tubes for a given core damage event. However, other circumstances such as reactor coolant pump seal leakage, steam generator tube leakage, or primary system relief valve leakage could create different severe accident thermal hydraulic conditions, which may produce a more severe challenge to the SG tubes. For example, pump seal leakage could lead to loop seal clearing and higher temperature steam entering the tube bundle. The purpose of this study was to determine if reasonable sequence variations (based on past probabilistic evaluations) would pose a unique threat that warranted closer examination. These analyses were performed to examine system level behavior and were thus performed with the SCDAP/RELAP5 code. The Zion plant model was used for these analyses because it is representative of a large class of PWRs with U-tube steam generators.

The detailed results of this study are provided in the attached report.

In summary, the results were consistent with past studies and indicated that unflawed tubes would survive through the severe accident, primarily because other RCS components, namely the surge line or hot leg, fail first, depressurizing the primary system and reducing the pressure loading on the SG tubes. In order to characterize the margin to failure for the tubes, we perform analyses that ignore the predicted surge line failure and continue the system heatup until tube failure is predicted. The relative time differential between predicted surge line failure and tube failure (assuming surge line failure had not occurred) is a rough measure of the margin available. The margin seen in these calculations, for the limiting sequences, is on the order of 11 -20 minutes and is comparable to the margin seen in past analysis although slightly smaller. The effect of tube flaws, together with flaws (and/or residual stresses) in other RCS components, is being considered by DET as part of the integrated evaluation of SG tube integrity.

Further, the analyses performed reveal that the most challenging scenario remains one in which core damage proceeds at relief valve set points and with a steam generator fully depressurized. Importantly, we did not see an adverse sensitivity to partial RCS depressurization resulting from leakage, either from pumps seals, relief valves or reasonable steam generator leak rates (up to 100 gpm). This was because we did not predict loop seal clearing of the reactor coolant pump loop seals and the primary effect of leakage was to lower RCS pressure, which is beneficial to the steam generator tubes. This conclusion may have significant benefits for the tubes from a probabilistic standpoint since it reduces the set of scenarios which challenge the tubes. This work will be coordinated with DRAA as part of their integration of the risk assessment. In another sensitivity case we determined that depressurization of a non-pressurizer loop steam generator (past studies assumed the pressurizer loop steam generator was faulted) produced no difference in the tube performance prediction. Finally, the study addressed the effects of tube plugging since many of the PWRs operate with a portion of the tubes plugged. The results also indicated that the effect of tube plugging (15% of the tubes were assumed to be plugged) was to produce a slightly more severe condition, albeit a small effect.

Future studies will address some of the modeling assumptions inherent in the SCDAP/RELAP5 analysis including those issues associated with the heat transfer in the hot leg and phenomenological behavior related to loop seal clearing as well as an integrated treatment of uncertainty in mixing behavior. This work will also be coordinated with the CFD analysis to address mixing behavior in the steam generator inlet plenum.

Attachment: As stated

cc: A. Thadani, RES
M. Mayfield, RES
J. Muscara, RES
S. Newberry, RES
E. Thornsbury, RES
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W. Jensen, NRR
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S. Long, NRR
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