ENCLOSURE

SAFETY EVALUATION REPORT

REVIEW OF SUPPLEMENT 1 TO WCAP-10698,

EVALUATION OF OFFSITE RADIATION DOSES FOR

A STEAM GENERATOR TUBE RUPTURE ACCIDENT

INTRODUCTION

In a May 24, 1985 letter to the NRC, the Steam Generator Tube Rupture (SGTR) Subgroup of the Westinghouse Owners Group (WOG) submitted Supplement 1 to WCAP-10698, Evaluation of Offsite Radiation Doses for an SGTR Accident, to support the resolution of the licensing issues associated with an SGTR accident. This Safety Evaluation Report documents the staff review of the results and methodology presented in Supplment 1 to WCAP-10698.

As a result of the January 1982 SGTR at the R. E. Ginna Plant, the NRC has questioned the assumptions used in the safety analysis of a design basis SGTR, including the operator action time assumed in terminating leakage from the primary to the secondary coolant systems, and the qualification of the equipment assumed to be used in the SGTR recovery. In response to these concerns, a subgroup of utilities in the WOG was formed to address the licensing issues associated with an SGTR event on a generic basis. In December of 1984, the subgroup submitted WCAP-10698, SGTR Analysis Methodology To Determine the Margin to Steam Generator Overfill, which presented the

development of a design basis SGTR analysis methodology. Supplement 1 to WCAP-10698 presents the evaluation of potential offsite doses for a design basis SGTR in the absence of steam generator overfill. The subgroup also plans to submit by November of 1985 an evaluation of the consequences of steam generator overfill resulting from an SGTR.

WCAP-10698 presented results from the following tasks in the development of a design basis SGTR analysis methodology: (1) development of LOFTTR1, an analytical model which is a modified version of LOFTRAN, that incorporates improved models for break flow and the steam generator secondary side, and an improved capability to simulate the operator actions for SGTR recovery; (2) determination of operator action times for design basis application based on the guidelines of Revision 1 of the WOG Emergency Response Guidelines issued in September 1983; (3) sensitivity studies to identify conservative values of plant parameters; (4) single failure analysis of the design basis equipment; and (5) application of the methodology to a reference plant.

The evaluation of offsite doses presented in Supplement 1 to WCAP-10698 used steam release rates to the environment and thermal and hydraulic parameters for the primary and secondary sides which were calculated using the LOFTTR1 computer code, and operator action times developed in WCAP-10698. In addition, the single failure analysis and sensitivity studies of Supplement 1 relied heavily upon the corresponding results of WCAP-10698. It should also be noted that staff review of the subgroup's evaluation of the consequences of steam generator overfill could potentially lead to changes in the analysis assumptions used in evaluating the radiological consequences of a design basis SGTR accident. Thus, the results and conclusions of this SER will be modified

as appropriate if staff review identifies the need for significant changes in the design basis SGTR analysis methodology presented in WCAP-10698. WCAP-10698 is currently under review by the staff with SER issuance for WCAP-10698 and for the evaluation of overfill consequences projected for the second quarter of FY1986. It should be noted, however, that the review of the dose analysis methodology (Section 5.0) presented in Supplement 1 with its assumptions and models of coolant activity levels and iodine transport processes is not dependent upon the results of the review of these other submittals.

Supplement 1 to WCAP-10698 presents the results of the following tasks: selection of a reference plant and site; single failure analysis to determine the worst single failure with respect to offsite doses; calculation of the mass releases to the environment using the results of the LOFTTR1 analyses from WCAP-10698 for mass releases prior to termination of the primary to secondary leakage, and the results of an analysis based on a continuation of the SGTR recovery actions in the WOG Emergency Response Guidelines for mass releases during the period between leakage termination and the end of the accident; and the development of the dose analysis methodology.

DISCUSSION

The evaluation of offsite doses in Supplement 1 to WCAP-10698, was performed for a reference plant and site. Atmospheric dispersion factors which were representative for typical Westinghouse plants were used in the dose calculations. The reference plant, as described in Section 4.1 of WCAP-10698, was selected on the basis of a preliminary analysis which provided estimates of the relative time to overfill for several representative Westinghouse plant

types. The calculations to determine the relative time to overfill compared the secondary side steam volume to the equilibrium break flow rate, defined as the break flow rate at the primary pressure at which outgoing break flow is balanced by incoming safety injection flow. The calculations did not consider accident system response and operator actions.

The staff notes that the selection of a reference plant based on the above estimates of the relative time to overfill does not assure the selection of the most conservative plant design with respect to potential offsite doses. Operator action time and system response time, which depend on plant specific equipment, operating procedures and individual plant design and parameters, must be considered in determining the duration and severity of the accident and the amount of radioactivity released to the atmosphere. The evaluation presented in Supplement 1 to WCAP-10698 was based on a reference plant with representative atmospheric dispersion factors, instead of a conservative plant design with bounding atmospheric dispersion factors. The staff concludes that the offsite dose calculations presented in Supplement 1 constitute representative examples of the application of the proposed design basis SGTR analysis methodology to a reference plant and site, but are not bounding cases. Plant specific analyses will be necessary to demonstrate that the radiological consequences of a postulated SGTR accident at an individual plant meet the acceptance criteria of Section 15.6.3 of the Standard Review Plan (NUREG-0800, Rev. 2, July 1981).

The single failure analysis to determine the worst single failure with respect to offsite doses and sensitivity studies to identify conservative (with respect to offsite doses) plant conditions, parameters, and other analysis assumptions

presented in Supplement 1 relied heavily upon the results of the single failure analysis and sensitivity studies in WCAP-10698 which were used to identify conservative assumptions with respect to margin to overfill. (The margin to overfill is defined as the steam space volume remaining below the steam generator outlet nozzle when the primary to secondary leakage is terminated). As stated in Supplement 1, it is expected that most of the conservative assumptions and initial conditions which were used in the evaluation of the margin to overfill would also be conservative with respect to offsite doses. This is based on the fact that both offsite doses and the potential for overfill are primarily dependent upon the amount of primary to secondary leakage and the amount of steam released from the ruptured steam generator.

The staff agrees that, in general, conditions and assumptions which are conservative with respect to overfill would also be conservative for offsite doses. The decrease in the margin to overfill as a result of a postulated single failure or a conservative analysis assumption is due to the increased operator action time and system response time required to complete the recovery action. The increased operator action time and system response time would prolong the accident and generally lead to increases in the release of radioactivity to the environment.

As discussed in Supplement 1, however, a decrease in the margin to overfill represents the additional net accumulation of water in the secondary side of the ruptured steam generator. Net accumulation of water increases with increases in the amount of primary to secondary leakage, but decreases with increases in the amount of steam released from the ruptured steam generator. (This follows from mass continuity considerations if one neglects

interdependency effects.) For those cases in which the amount of steam released to the atmosphere does not change, conservative conditions with respect to overfill would also be conservative with respect to offsite doses. In these cases the decrease in the margin to overfill is a result of an increase in the amount of primary to secondary leakage due to increased operator action time and system response time. This prolongs the accident and results in increased releases of radioactivity to the environment.

The single failure analysis presented in Supplement 1 has identified and examined those cases which result in increases in the amount of steam released from the ruptured steam generator. In addition, the analysis identified an estimated proprietary hydraulic parameter which was conservative with respect to offsite doses, but was not conservative with respect to margin to overfill. This assumption is discussed in Section 5.2 of Supplement 1 and was investigated in various case comparisons, including a comparison of calculated doses for Cases 1 and 5.

Based on the above findings, the staff concludes that the single failure analysis and sensitivity studies in Supplement 1 have identified the worst single failure and the analysis assumptions which are conservative with respect to offsite doses. This conclusion is based upon the following: staff review of the sensitivity studies and equipment failure evaluation in WCAP-10698 to assure that conservative plant conditions, parameters, and analysis assumptions and the worst single failure with respect to margin to overfill have been properly identified; the generic applicability of the single failure analysis in WCAP-10698; and the use of the assumption which was identified in Section 5.2 of Supplement 1 to be conservative with respect to offsite doses but not

with respect to margin to overfill in subsequent applications of this methodology for the evaluation of offsite doses from an SGTR accident.

The results and conclusions of this SER will be modified as appropriate if the review of WCAP-10698 identifies the need for significant changes in the results of the sensitivity studies and equipment failure evaluation presented in WCAP-10698. In addition, the single failure analysis presented in WCAP-10698 is based on the WOG Emergency Response Guidelines which are applicable to nearly all Westinghouse plants, and a design basis equipment list which identifies sufficient principal equipment to terminate primary to secondary leakage for all Westinghouse plants. The generic applicability of the analysis may be limited, however, based on plant specific differences which would affect changes in operator action times and system response times required to complete the recovery operation as a result of a postulated single failure. For example, the staff notes that the results of the single failure analysis in WCAP-10698 may not apply to two loop Westinghouse plants. If, as a result of the staff review of WCAP-10698, it is determined that the single failure analysis is not generically applicable, then plant specific analysis to determine the worst single failure with respect to offsite doses may be required.

The staff has reviewed the evaluation of offsite doses for the single failure cases considered in Supplement 1 to WCAP-10698. Mass releases from the ruptured and intact steam generators to the atmosphere were determined from LOFTTR1 analyses (described in WCAP-10698) for the period from accident initiation to the termination of primary to secondary leakage. Mass releases for the period from leakage termination to the end of the accident, assumed to be 8 hours,

were determined from an analysis based on SGTR recovery operations in the WOG Emergency Response Guidelines. Revision 1 of the Emergency Response Guidelines provides for three alternate means of performing the post - SGTR cooldown. The method using steam dump, Guideline ES-3.3, was selected for evaluation of the mass releases since it results in conservative results for the offsite dose evaluation. The ES-3.3 guideline specifies the actions required to bring the Reactor Coolant System down to Residual Heat Removal System temperature and pressure levels. This is accomplished by using steam dump to the condenser, or using the power operated relief values of the intact and ruptured steam generators if the condenser is unavailable.

The dose analysis methodology as presented in Supplement 1 to WCAP-10698 uses assumptions for the initial primary and secondary coolant activity concentrations, the radiological consequences of iodine spiking, a coolant iodine spiking model for the accident initiated iodine spike case, and primary to secondary system leakage in the intact steam generators which are consistent with those in Section 15.6.3 of the Standard Review Plan. In the determination of iodine transport to the atmosphere, the methodology presented in Supplement 1 discusses the volatilization of iodine in the primary coolant due to flashing and atomization, and the scrubbing of iodine contained in the steam phase and atomized droplets for release points which are below the steam generator water level. It does not, however, explicitly describe the models and assumptions used in the determination of iodine transport in the faulted generator. Thus, no staff review of the iodine transport models was possible, and independent staff verification using the iodine transport models referenced in the Standard Review Plan will be necessary on a case-by-case basis. It is the staff's position that plant specific analyses should

provide a detailed description of, or reference, the explicit iodine transport models used in the analyses.

The staff concludes that the dose analysis methodology presented in Supplement 1 to WCAP-10698 is generally consistent with Section 15.6.3 of the SRP and, thus, is acceptable with the exception of the iodine transport models which will be reviewed on a case-by-case basis.

CONCLUSIONS

The staff has reviewed the methodology and results presented in the evaluation of offsite doses for an SGTR accident in Supplement 1 to WCAP-10698. The staff concludes that the dose analysis methodology used in the evaluation is acceptable with the exception of the determination of iodine transport to the atmosphere for which explicit models and assumptions were not provided. Independent staff verification using the iodine transport models referenced in the SRP will be necessary on a case-by-case basis.

The staff notes that the offsite dose calculations presented in Supplement 1 were based on a reference plant and reference site and, thus, did not constitute bounding cases for all reactors and sites. Plant specific analyses will be necessary to demonstrate that the radiological consequences of a postulated SGTR accident at an individual plant meet the acceptance criteria of Section 15.6.3 of the SRP.

The results and conclusions of this SER will be modified as appropriate if staff review of WCAP-10698 and of the subgroup's evaluation of the consequences of steam generator overfill identifies the need for significant changes in the design basis SGTR analysis methodology presented in WCAP-10698.

IMPLEMENTATION

As discussed above, plant specific evaluations of offsite doses using appropriate plant specific mass releases and thermal and hydraulic parameters for the primary and secondary systems will be necessary for individual plants. The evaluation should consider the worst single failure and plant conditions, parameters, and assumptions which are conservative with respect to offsite doses. The results of the single failure analysis and sensitivity studies in Supplement 1 are acceptable, provided the single failure analysis in WCAP-10698 is generically applicable and the staff review of WCAP-10698 does not identify the need for significant changes. If, as a result of the staff review of WCAP-10698, it is determined that the single failure analysis is not generically applicable, then plant specific single failure analyses to determine the worst single failure with respect to offsite doses may be required. In addition, the plant specific evaluations of offsite doses should use the analysis assumption which was identified in Section 5.2 of Supplement 1 to be conservative with respect to offsite doses but

which was not conservative with respect to margin to overfill.

The plant specific analysis should provide sufficient information for staff review, including the following information as a function of time during an SGTR, to allow an independent evaluation to be made by the staff of the radiological consequences:

(1) Total mass releases and mass release rates from the ruptured steam generator to the atmosphere,

(2) Total mass releases and mass release rates from the intact steam generator(s) to the atmosphere,

(3) Primary to secondary system leakage flow rate in the faulted generator (break flow rate),

(4) Pressure differential between the RCS and the ruptured steam generator,

(5) Water level above the break location in the ruptured steam generator,

(6) Mass of water in ruptured steam generator,

(7) Pressure in the ruptured steam generator, and

(8) RCS hot leg and cold leg temperatures in the ruptured loop.

In addition, it is the staff's position that plant specific analyses should include a detailed description of, or reference, the explicit iodine transport models used in the analyses.