UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Information)

November 24, 1982

SECY-82-466

15pp

FOR: The Commissioners

FROM: Executive Director for Operations

SUBJECT: PRELIMINARY STAFF ANALYSIS OF STEAM GENERATOR TUBE RUPTURE RISKS

PURPOSE: To provide the Commission with information on a preliminary staff analysis of steam generator tube rupture risks, including multiple tube failures.

DISCUSSION: At the November 18, 1982 Commission briefing on steam generator related activities, the subject of multiple steam generator tube ruptures, and the public risk associated with them, received considerable attention. A copy of a preliminary staff analysis of the public risk associated with all tube rupture events is enclosed to provide you with current staff thinking on this subject. This analysis is undergoing review and a final version will be included in the integrated program package on steam generator tube degradation and tube ruptures. This draft indicates that the public risk associated with all causes of steam generator tube rupture is about 20 man-rem/year.

It is NRR's current view that four of the potential actions under consideration may prove to be justifiable candidates for regulatory requirements. The most significant are the requirements for Secondary Water Chemistry and Condenser Inservice Inspection Programs. These two items rate very favorably when evaluated against the criteria of occupational exposure and the prevention of significant but less than core melt releases. Their implementation would

XA Copy Has Been Sent to PDR

Contact:
Harold Denton, Dir., ONRR
X27691

8212130066 XA

also have a net economic benefit. The other two items are the Prevention and Detection of Loose Parts by Visual Inspections and Quality Assurance Work Procedures, and Changes to the Tube Inservice Inspection Requirements. These items would be valuable in preventing significant but less than core melt releases, and also have net economic benefits.

Jack W Roe for
William J. Dircks
Executive Director for Operations

Enclosure:
Draft Summary of Risk Analysis
for Steam Generator Tube Rupture
Events

DISTRIBUTION:
Commissioners
SECY
OGC
OPE
REGIONAL OFFICES
OCA
OIA
OPA
EDO
ELD
ACRS

DRAFT

APPENDIX A

Summary of Risk Analysis for Steam Generator Tube Rupture Events

DRAFT

Summary of Risk Analyses for Steam Generator Tube Rupture Events

As stated in the introduction of this report, staff concerns relative to steam generator tube degradation stem from the fact that the steam generator tubes are a part of the reactor coolant system (RCS) boundary and that tube failures result in a loss of primary coolant. In addition, the steam generator tubes constitute a particularly important part of the RCS boundary since their failure allows primary coolant into the steam generators where its isolation from the environment is not fully assured. The release of primary coolant into the environment has two major safety implications. The first is the direct release of radioactive fission products, and the second is the loss of cooling water which is needed to cool the core. An extended loss of cooling water outside of containment would result in the depletion of the initial RCS inventory and ECCS water without the capability to recirculate the water as would be the case for any LOCA inside containment. The major safety concerns are therefore related to events in which a loss of steam generator tube integrity occurs in combination with a loss of secondary system integrity. There are two classes of events with these characteristics: (1) tube ruptures followed by a loss of coolant through the secondary system due to failure of the SG safety or relief valve (due to valve failure or operator error) or main steamline failure; and (2) a secondary system failure (steam line break, feedwater line break or stuck open valve) with a consequential failure of previously degraded steam generator tubes.

The probability and consequences of tube ruptures, particularly those events which also involve a loss of secondary system, has been the subject of several studies. The staff's contractor, SAI, analyzed the single SGTR and concluded that, "the dominant accident sequences leading to core melt following an SGTR are: (1) a loss of offsite power and failure of both diesels to start, i.e., total power loss, and (2) failure of the auxiliary feedwater system. The core melt probability was found to be 3×10^{-7}/RY to 1×10^{-7}/RY.

failure of 10 or fewer tubes has no significant effect on peak clad temperatures, that failure of 10 to 20 tubes would increase the peak clad temperatures to approximately 2400°F for tube failure during refill and to 2300°F for tube failure during reflood; and that failure of more than 20 tubes would have only slightly adverse to slightly benefical effects. These trends are shown in Figure 1 taken from an INEL report CVAP-TR-78-015. Therefore the probability of a core melt following a LOCA would be increased for those events in which 10 to 20 tubes fail. Although the analyses and experiments do not indicate that core melt would occur, a core melt probability of 0.1 has been assigned to the LOCA events (10 to 20 SGTRs) showing elevated peak clad temperatures. This is intended to account for plant-to-plant variations and for uncertainties in this and other aspects of the ECCS response to a LOCA. In order to estimate the probability of experiencing 10 to 20 tube failures during a transient such as a LOCA or MSLB, operating experience with degraded tubes was studied.

We have therefore made the following estimate of SGTR rupture during a LOCA or main steamline break (MSLB) based on consideration of three plants which have previously experienced SGTRs during normal operation.

<u>Ginna</u> - NRC-sponsored burst tests of S/G tubing indicate that wall penetrations of at least 75% and 88% are necessary to result in rupture during MSLB and during normal operation, respectively. The tube which ruptured on January 25, 1982 is estimated to have been degraded 40% through wall in April 1981. Assuming an 88% penetration at the time of rupture and a linear penetration rate, the amount of penetration would have exceeded 75% for about 2.6 months. We have assumed that 11 other tubes which were plugged in previous years were plugged as a result of long wear scars. Taking no credit for ECT to accurately size these indications (because of differential coil limitations in detecting long wear scars), we estimate that two additional tubes could have penetrated beyond 75% (but less than 88%) for varying periods up to 2.2 months. The total number of calendar months since initial startup at Ginna is about 158 months. The vulnerable fraction of operating time was therefore 0.03 (i.e., 4.8/158).

total of 412 reactor months since initial startup of the three plants considered. Thus, the probability for being vulnerable to tube rupture during transients such as LOCA or MSLB is .028.

The probability for rupturing a given number of tubes decreases with the number of tubes. The assumption of a .5 probability that tube ruptures during MSLB will involve two or more ruptures should be reasonably conservative when the following factors are considered:

(1) There was no potential for multiple ruptures at Prairie Island 1.

(2) The possible vulnerability for multiple ruptures (2 SGTRs only) existed for only 20% of the period of vulnerability to a single SGTR at Ginna.

(3) Possible vulnerability to multiple ruptures may or may not have existed at Point Beach and Surry at time of actual rupture events. If vulnerability did exist, it would have been for shorter periods of time than the period of vulnerability to single SGTR during MSLB.

The probability of $>$10 SGTRs during MSLB will be much smaller than the probability of $\geq$2 SGTRs. It is very unlikely that as many as ten tubes could be vulnerable to rupture during a transient without some warning indication (small leak, single SGTR during normal operation). An assumed probability of .1 that ruptures during MSLB will involve more than 10 tubes should be conservative. Summarizing, the probability of being vulnerable to tube ruptures during transients and therefore the assumed failure probability on demand are as follows:

$P(\geq 1 \text{ SGTR}) = .03/D$

$P(2 \text{ to } 10 \text{ SGTR}) = .012/D$

$P(> 10 \text{ SGTR}) = .003/D$

For the single tube rupture cases, an initiating event probability of 2×10^{-2}/RY is used This is based on actual operating experience (i.e., 4 tube ruptures in Westinghouse plants in 240 reactor-years of operation for Westinghouse plants; insufficient operating experience is available for CE and B&W plants to justify a smaller value for those plants). The probability of a multiple tube rupture as an initiating event was assumed to be 3×10^{-3}/RY. This corresponds to the point-estimate for an event which has not yet been experienced (i.e., assuming one multiple SGTR for the 353 reactor-years of operation for PWRs).

An additional aspect of these sequences needs to be addressed before the risk analysis can be completed. This is the probability of a main steamline failure (between the containment and the MSIV) as a result of overfilling the steam generator. The concern for a combined main steamline break following an SGTR event was highlighted by the filling of the Ginna steamlines with water. The staff has assessed the change in the probability of failure of the MSL due to the increased stress levels associated with the deadweight of water in the steamlines. Analyses have been performed of the increase in stress levels which would result from filling of the steam lines on several plants. Information extracted from analyses on the Ginna, Zion 1, Waterford 3, and Oconee 3 plants indicate that, although in some cases the spring hangers may be loaded slightly beyond their operating range, they will not fail and that the stress levels in the main steamline will in all cases remain within code-allowable limits. On this basis the staff concludes that the probability of failure of the main steamline is not increased by the dead weight loading. Accordingly the estimates of risk in this report for event sequences which consider failure of the main steamlines are based on a conservatively determined probability of main steamline failure of 1×10^{-3} per demand.

The event sequences in this category are as follows:

Sequence 4B

Event	Probability
1. Multiple SGTR (greater than 10 tubes)	3×10^{-4}/RY
2. SG overfill	1.0
3. SG Safety Valve Challenge	1.0
4. SG Safety Valve Sticks Open	3×10^{-2}
5. Failure to depressurize RCS to atmospheric before RWST is exhausted (~ 1 hr for a 170°F cooldown)	.5
	5×10^{-6}/RY

Sequence 5A

Event	Probability
1. Multiple SGTR (2 to 10 tubes fail)	3×10^{-3}/RY
2. SG overfill	1.0
3. Main Steamline Failure	10^{-3}
4. Failure to depressurize to atmospheric before RWST is exhausted (~5 hrs for a 200°F cooldown)	10^{-2}
	3×10^{-8}/RY

Sequence 5B

Event	Probability
1. Multiple SGTR (greater than 10 tubes)	3×10^{-4}/RY
2. SG Overfill	1.0
3. Main Steamline Failure	10^{-3}
4. Failure to depressurize to atmospheric before RWST is exhausted (~ 1 hr for a 170°F cooldown)	.5
	1.5×10^{-7}/RY

Sequence 6A

Event	Probability
1. Mutiple SGTR (2 to 10 tubes)	3×10^{-3}/RY
2. MSIV fails to isolate SG	10^{-3}
3. Failure to depressurize to atmospheric before RWST is exhausted (~ 5 hrs for 200°F cooldown)	10^{-2}
	3×10^{-8}/RY

category, the corresponding public dose in man-rem/event and man-rem/plant life).

Since all of the sequences which involve a loss of secondary integrity also involve a significant release of primary coolant to the environment, even without a core melt, the probability and consequences of the corresponding non-core melt sequences have been analyzed. These sequences assume correct operator action to depressurize the RCS to atmospheric pressure at a rate corresponding to a 100°F/hr cooldown. Therefore, the probability of these sequences is the same as for sequences 1 through 8 above, except for the assumption relative to operator action. A summary of the probability, primary coolant release, I^{131} release and offsite dose is presented in Table 3.

These analyses indicate that the probability of a core melt from all causes associated with steam generator tube ruptures is small (6×10^{-6}/RY) and that the total public exposure associated with these events, over the lifetime of a plant, is 750 man-rem.

The analyses also indicate that the probability of a large release without a core melt is reasonably small (3×10^{-4}/RY) and the total public exposure associated with non-core melt SGTR events, over the lifetime of a plant is 40 to 840 man-rem.

10. Letter, "Maximum Stresses in Zion Unit 1 Main Steam Piping" to K. R. Wichman, NRR, DL, from Long C. Shiek, Lawrence Livermore Laboratory of October 20, 1982.

11. Letter, "Stress Analysis of Hydrotest in Main Steam Line A at Waterford S.E.S. Unit 3," to K. R. Wichman, NRR, DL, from C. R. Hammond, Union Carbide Corporation of October 19, 1982.

12. Letter, "Oconee Nuclear Station" to Harold R. Denton, NRR, from H. B. Tucker, Duke Power Company of November 1, 1982.

Table 2 Summary of Probability and Consequences
for SGTR Events Leaderto Core Melt

Sequence Loss of SG Integrity	Probability (1/RY)	Release/ category	Release/ event (man-rem/event)	Release/ plant life (man-rem)
1. Single SGTR + Loss of SG Integrity (SGSV Stuck open)	1.5×10^{-7} to 15×10^{-8}	PWR-4	2.7×10^{-6}	.4 x 40
2. Single SGTR + Loss of SG Integrity (SLB)	5×10^{-9}	PWR-4	2.7×10^{6}	.01 x 40
3. Single SGTR + Loss of SG Integrity (MSIV failure)	2×10^{-8}	PWR-4	2.7×10^{6}	.05 x 40
4a) Multiple (2 to 10) SGTRs + Loss of SG Integrity (SGSV stuck open)	9×10^{-7}	PWR-4	2.7×10^{6}	2.4 x 40
4b) Multiple (>10) SGTRs + Loss of SG Integrity (SGVS stuck open)	5×10^{-6}	PWR-4	2.7×10^{6}	13.5 x 40
5a) Mutliple (2 to 10) SGTRs + Loss of SG Integrity (SLB)	3×10^{-8}	PWR-4	2.7×10^{6}	.08 x 40
5b) Multiple (>10) SGTRs + Loss of SG Integrity (SLB)	1.5×10^{-7}	PWR-4	2.7×10^{6}	.4 x 40
6a) Multiple (2 to 10) SGTRs + Loss of Integrity (MSIV failure)	3×10^{-8}	PWR-4	2.7×10^{5}	.08 x 40
6b) Multiple (>10) SGTRs + Loss of SG Integrity (MSIV failure)	1.5×10^{-7}	PWR-4	2.7×10^{6}	.4 x 40
7. SLB + Single SGTR	1.5×10^{-9}	PWR-4	2.7×10^{6}	.005 x 40
8a) SLB + Multiple (2 to 10) SGTRs	1.2×10^{-8}	PWR-4	2.7×10^{6}	.03 x 40
8b) SLB + Multiple (>10) SGTRs	1.5×10^{-7}	PWR-4	2.7×10^{6}	.1 x 40
<u>LOCA</u>				
1. LOCA + Multiple (10-20) SGTRs	1.2×10^{-8}	PWR-4	2.7×10^{6}	.08 x 40
<u>Loss of Decay Heat Removal</u>				
1. SGTR + Loss of All Feedwater	2×10^{-7}	PWR-4	2.7×10^{6}	.5 x 40
2. SGTR - Loss of RHR	3×10^{-7}	PWR-4	2.7×10^{6}	.8 x 40
	6×10^{-6}/RY			18.8 x 40 = 752