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Docket Nos. 50-348
50-364

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

Joseph M. Farley Nuclear Plant
Steam Generator Alternate Plugging Criteria and Severe Accidents

Gentlemen:

By letter dated August 18, 1992, the NRC Staff requested additional information concerning severe accident induced failures of degraded steam generator tubes in support of the alternate plugging criteria. As a result, Southern Nuclear examined aspects of severe accidents as they relate to implementation of the alternate plugging criteria.

Southern Nuclear has completed an evaluation of the risk associated with the implementation of the alternate plugging criteria using a probabilistic argument. The evaluation took into account the following:

1. A steam generator tube rupture (SGTR) as an initiating event leading to core damage;

2. A transient-induced SGTR with subsequent core damage; and

3. A transient-induced SGTR after core damage.

As a result of this evaluation, the total SGTR-related frequency of core damage with containment bypass with the alternate plugging criteria implemented is estimated to be on the order of 4.7×10^{-7} per reactor year (RY). This frequency of core damage with containment bypass associated with SGTRs at Farley Nuclear Plant does not contribute a significant fraction of the risk associated with reactor events.

Southern Nuclear believes that the highest temperature of concern in the steam generator with elevated pressure should not exceed 1000°F. It is our engineering judgment that the frequency for event sequences that may result in steam generator tube temperatures above 1000°F is sufficiently low, i.e., less than 3.5×10^{-7}/RY, such that further attention is not

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warranted. At temperatures up to 1000°F, steam generator tubes left in service as a result of the alternate plugging criteria will not experience creep rupture failure during the duration of the severe accident transient.

In summary, it is our judgment that the probabilistic evaluation of severe accident risk addresses the NRC concern and the Farley alternate plugging criteria should be approved. Southern Nuclear's probabilistic assessment is attached for your review.

Alabama Power Company submitted its original alternate plugging criteria in February 1991. Since that time we have responded to several NRC Staff Requests for Additional Information, have revised the alternate plugging criteria to address Staff concerns, and incorporated recommendations from Staff consultants. Additional data have been incorporated into the burst and leak rate data bases which continue to support the alternate plugging criteria. The alternate plugging criteria is a key factor in making decisions affecting steam generators. Therefore, in order to facilitate approval of the alternate plugging criteria, Southern Nuclear requests a meeting with the NRC Staff during the week of April 5, 1993.

If there are any further questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

J. D. Woodard

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NEL-93-0111
Attachment

cc: Mr. S. D. Ebneter
Mr. T. A. Reed
Mr. G. F. Maxwell

ATTACHMENT

Assessment of Impact of
Alternate Tube Plugging Criteria on
Risk Related to Severe Accidents

ATTACHMENT

Questions posed by the NRC relate to the impact of steam generator tube alternate plugging criteria on plant risk. In response to those questions, the risk (which is measured here in terms of core damage frequency with containment bypass) related to steam generator tube rupture (SGTR) events with implementation of the alternate plugging criteria has been evaluated.

SGTR events are among a number of potential contributors to core damage frequency. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," provides an approach for estimating SGTR-related core damage with containment bypass frequency. In NUREG-0844, it has been postulated that, in addition to tube ruptures occurring as initiating events, consequential tube ruptures might occur as a result of high primary-to-secondary side pressure differentials associated with other initiating events, e.g., steam line break, prior to core damage. NUREG-0844 also addresses risk from tube leakage without core damage following an event that challenges tube integrity. In probabilistic risk assessment (PRA) analyses performed in support of NUREG-1150, "Severe Accident Risks for Five U.S. Nuclear Power Plants," Rev. 1, June 1989, it has been postulated that tube rupture could also occur following core damage, given the existence of a high reactor coolant system (RCS) pressure and elevated tube temperatures. The following discussion evaluates the total frequency of core damage with containment bypass due to the following:

1. A SGTR as an initiating event leading to core damage;

2. A transient-induced SGTR with subsequent core damage; and

3. A transient-induced SGTR after core damage.

<u>SGTR as an Initiating Event Leading to Core Damage.</u>

The preliminary Farley Nuclear Plant (FNP) Unit 1 and Unit 2 Individual Plant Examination (IPE) estimates the frequency of SGTR-related core damage with containment bypass as approximately 2.7×10^{-7} per reactor year (/RY). This accounts for single tube rupture initiating events. Implementation of the alternate plugging criteria will not increase the 2.7×10^{-7}/RY frequency because the probability of tube burst at normal operating differential pressure is so small. The probability of tube burst is negligible since the margins of Regulatory Guide 1.121 are maintained and the tube support plates provide restraint against burst.

Thus, application of the alternate plugging criteria has a negligible impact on the frequency of SGTR-related core damage events.

<u>Transient-Induced SGTR With Subsequent Core Damage.</u>

Severe accident sequences involving high primary pressure and low secondary side pressure may affect steam generator tube integrity. The frequency of containment bypass related to consequential tube failure and subsequent

related core damage accidents is estimated as the frequency of occurrence of loss of secondary side integrity events, in which the secondary side pressure is reduced to essentially atmospheric conditions, multiplied by the conditional probability of core damage and by the probability of failure of a steam generator tube following such events. For events in which secondary side integrity is <u>not</u> challenged, it is expected that the secondary side likely will be maintained at pressure via operation of steam dumps, atmospheric relief valves, or steam generator safety valves.

Events constituting a loss of secondary side integrity which challenge tube integrity include the following: a break between the containment wall and the main steam isolation valves (MSIVs); a break downstream of the MSIVs, with a consequential failure to close at least one of two MSIVs in a given steam line; or a stuck open secondary safety or relief valve(s) in each case leading to complete depressurization of the secondary side. The frequency of stuck open secondary valves with core damage is estimated in the FNP IPE to be about 1×10^{-6}/RY. The initiating event frequency for steam line breaks (both upstream and downstream of MSIVs disregarding MSIV closure) is estimated as 2.5×10^{-3}/RY, consistent with initiating event frequencies in the IPE. The conditional probability of core damage, given a steam line break, is estimated to be on the order of 1×10^{-3}, taking into account failures of auxiliary feedwater, safety injection, and human error. The steam line break induced core damage frequency (CDF) is then estimated to be (2.5×10^{-3}/RY x 1×10^{-3}), or approximately 2.5×10^{-6}/RY. This is consistent with the preliminary IPE predictions.

NUREG-0844 estimates the probability of occurrence of a consequential single tube rupture during a steam line break event to be 2.5×10^{-2}. For events with lower primary-to-secondary side pressure differentials than assumed for steam line breaks, NUREG-0844 estimates tube failure probabilities correspondingly less than this steam line break value. Implementation of the steam generator tube alternate plugging criteria requires that the rupture probability be shown to be less than the NUREG-0844 value, with 90% confidence.

Using the NUREG-0844 conditional probability of tube failure given a steam line break, the frequency of core damage with containment bypass for steam line breaks with consequential tube failure is then estimated to be: (1×10^{-6}/RY + 2.5×10^{-6}/RY) X (2.5×10^{-2}) = 9×10^{-8}/RY.

Note that the above assessment does not take credit for existing FNP emergency operating procedures (EOPs) (e.g., FNP-1(2)-EEP-3.0, -ECP-3.1, etc.) which provide guidance to the operators for response to a tube rupture (including a tube rupture with a loss of secondary integrity), thereby reducing the likelihood of core damage. Further, the frequency of 2.5×10^{-6}/RY as estimated above assumes no credit for closure of the MSIVs following a break downstream of the MSIVs; thus, using that frequency in the calculation of containment bypass frequency following a consequential tube rupture for this group of events is conservative.

NUREG-0844 assigns a conservatively high probability of multiple consequential steam generator tube failures, given a pressure differential of 2600 psi, of 2.5×10^{-2}. Using this value, the frequency of induced multiple tube rupture with core damage and containment bypass is also calculated to be 9×10^{-8}/RY, for the same reasons given above.

The total frequency for transient-induced SGTR with subsequent core damage is:

Single SGTR	9×10^{-8}/RY
Multiple SGTR	9×10^{-8}/RY

	1.8×10^{-7}/RY

In conclusion, implementation of the alternate plugging criteria requires that the probability of tube rupture during a steam line break be maintained less than 2.5×10^{-2} with 90% confidence. This ensures that the risk associated with alternate plugging criteria with regard to transient-induced SGTRs is maintained below NUREG-0844 estimates.

Transient-Induced SGTR After Core Damage.

After core damage, the steam generator tubes could be exposed to pressures and temperatures conducive to consequential tube failure. Investigations of the behavior of steam generator tubes after core damage have concluded that the dominant mechanism for the tube failure is tube burst at high tube temperatures and stress levels. The investigations found that the frequency of creep rupture failure of the tubes is several orders of magnitude smaller than the frequency of tube burst. Thus, the focus of the core damage accident sequences is confined to loss of secondary side integrity events, i.e., events in which the RCS is at nominal operating pressure and the secondary side is depressurized. However, the probability of occurrence of these sequences is very small.

The IPE assumes core damage if the peak fuel clad temperature exceeds 1200°F for a sustained period. This relatively low temperature was chosen to allow for margin between code predictions and actual core conditions. and also because it coincides with the core exit thermocouple indication at which plant operators would enter the inadequate core cooling (ICC) emergency operating procedures. In the ICC procedure, the operators are instructed to depressurize the reactor coolant system. The IPE takes no credit for prevention of core damage via operator recognition and action upon seeing core exit thermocouple temperatures of 1200°F. Thus, for core damage sequences in which the pressurizer PORVs are available, there is only a small probability that the operators would fail to follow the ICC

procedure and depressurize to low primary-to-secondary side differential pressure conditions, at which steam generator tube burst would not be expected.

The frequency of core damage with a loss of secondary side integrity and a failure to depressurize the RCS is estimated, using the FNP IPE core damage frequencies, and assuming a value of 0.1 for operator failure to depressurize per the ICC procedure, as $(1 \times 10^{-6} + 2.5 \times 10^{-6}/RY) \times (0.1) = 3.5 \times 10^{-7}/RY$. With alternate plugging criteria the conditional probability of tube failure under these conditions will be limited to 4.8×10^{-2} at the temperatures expected in the steam generator tubes following core damage (up to 1000°F). Note that this probability is two orders of magnitude higher than the mean value estimated by expert judgment in NUREG-4551, Vol. 2, Rev. 1, Part 1, "Evaluation of Severe Accident Risks: Qualification of Major Input Parameters," Dec. 1990. (a study performed in support of NUREG-1150). The higher estimate presented here, compared to NUREG-4551, is not due to the APC; rather, it is a result of the use of a more conservative modeling evaluation for the purposes of this investigation. NUREG-4551 did not consider the probability of tube burst as a tube failure mechanism following core damage. Even with this higher estimate, the frequency of tube rupture related containment bypass following core damage is estimated to be: $(3.5 \times 10^{-7}/RY \times 4.8 \times 10^{-2}) = 1.7 \times 10^{-8}/RY$. This frequency is small compared to potential SGTR related core damage sequences due to other scenarios noted herein.

Southern Nuclear believes that the highest temperature of concern in the steam generator with elevated pressure should not exceed 1000°F. It is our engineering judgment that the frequency for event sequences that may result in steam generator tube temperatures above 1000°F is sufficiently low , i.e., a small portion of the $3.5 \times 10^{-7}/RY$ frequency discussed above, such that further attention is not warranted. At temperatures up to approximately 1000°F, steam generator tubes left in service as a result of the alternate plugging criteria will not experience creep rupture failure during the duration of the severe accident transient.

In summary, for those events in which core damage occurs and leads to a challenge of steam generator tube integrity, implementation of the alternate plugging criteria will maintain containment bypass risk to a frequency on the order of $1.7 \times 10^{-8}/RY$.

Conclusions

While there are conservatisms in several NUREG-0844 parameters used in this evaluation (e.g., the probability of multiple consequential tube ruptures), this is appropriate in order to avoid understating potential risk. The

total SGTR-related frequency of core damage with containment bypass with APC is estimated to be:

SGTR Initiating Events Contribution	2.7×10^{-7}/RY
Transient-Induced Single and Multiple SGTR with Subsequent Related Core Damage	1.8×10^{-7}/RY
Transient-Induced SGTR After Core Damage	1.7×10^{-8}/RY
Total	4.7×10^{-7}/RY

NUREG-0844 estimated a total SGTR-related core damage with containment bypass frequency (referred to in the NUREG as "core-melt probability") of about 4×10^{-6}/RY for Westinghouse plants, and concluded that SGTRs do not contribute a significant fraction of the risks associated with reactor events at a given plant. Because the frequency of SGTR-related core damage with containment bypass estimated in the foregoing for FNP is significantly less than that estimated in NUREG-0844, the same conclusion relative to acceptability of risk remains valid for the Farley Nuclear Plant following implementation of the proposed steam generator tube alternate plugging criteria.