Severe Accident Analysis Report

Title: **Probabilistic Risk Perspective of the Steam Generator Tube Structural Integrity Issue Related to the Interim Steam Generator Tube Plugging Criteria for the Catawba Nuclear Station**

SAAG File: 87

Project : Catawba Plant Support

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I. Introduction

The steam generator tubes of a pressurized water reactor are an integral part of the primary system boundary. Technical specifications provide repair limits to ensure that degraded tubes which cannot adequately perform this function are either sleeved or plugged. The interim tube repair limits for Catawba Unit 1 consist of voltage amplitude criteria rather than the traditional depth-based criteria. Thus, the interim repair criterion represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement. This change in practice has raised some concerns about the structural integrity of the steam generator tubes. This probabilistic risk study evaluates the implications of allowing Catawba Unit 1 to run a full cycle after using the interim repair criteria.

II. Method

Since the publication of WASH-1400, it has been recognized that the primary risk to the public from nuclear power plants results from low probability high consequence events involving severe overheating and damage of the reactor core. Other potential accidents that do not involve core damage result in very low risk. Changes to the plant, such as system modifications or procedure revisions, can be evaluated to determine their impact on the core damage frequency and the public health risk. These quantifiable characteristics can be compared to the normal plant risk to determine the safety significance of a change.

NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," also found that core damage risk following a tube rupture is far more significant than the risk associated with the release of coolant activity. The NUREG found the public health risk of non-core melt releases to be more than a factor of 1000 lower than the risk of tube rupture sequences involving core melting. Therefore, a probabilistic risk assessment method which evaluates the change in core damage risk in a manner consistent with the Catawba PRA, can be used here to evaluate the safety significance of the interim steam generator tube plugging criteria.

III. Calculations

The interim plugging criteria may impact the structural integrity of the steam generator tubes. Therefore, the risk implications of the interim plugging criteria are evaluated by assessing the impact on plant risk that might result from changes in the structural integrity of the steam generator tubes. Changes in the structural integrity of the tubes could impact plant risk by means of the following:

- Increased potential for random tube rupture which results in core damage,
- Increased potential for a transient to induce a tube rupture which results in core damage,

For the two concerns listed here, the risk associated with each one is a product of the frequency of the initiating event, the probability of core damage, and the consequences of core damage. The risk calculations all follow the same general form:

Risk = (Initiator Freq.)x(Conditional Prob. of Core Damage)x(Public Health Conseq.)

If the interim tube plugging criteria has an effect on any part of this three part equation, then it has an impact on the risk. The review of each concern is discussed below.

A. Random Tube Rupture

The first concern is that the interim tube plugging criteria may significantly increase the chance of a random tube rupture. Of the three parts of the risk calculation: (1) initiator frequency, (2) conditional probability of core damage, and (3) consequences, only the initiator frequency could be affected by the interim plugging criteria. The Catawba PRA assumed a steam generator tube rupture initiator frequency of $9x10^{-3}$/RY. This was based on industry data that involved seven tube ruptures in the history of operating PWRs. The Catawba PRA did not explicitly consider the existence of degraded tubes in the analysis. To determine the effect on random tube rupture risk, it must be determined if the interim tube plugging criteria would significantly change the Catawba tube rupture frequency.

The type of tube degradation experienced at Catawba causes cracks which are confined to the part of the tubes surrounded by the tube support plates. Unless these cracks are exposed by movement or deflection of the support plate, they will not rupture. Since the support plate does not move under normal operation, there is essentially no risk of a random tube rupture associated with the interim tube repair criteria. In fact, the full length inspection of all the Catawba tubes during the last outage leads to additional confidence that the probability of a random tube rupture at Catawba is less than the value assumed in the Catawba PRA Study.

The PRA analyzed random tube ruptures as a possible way to cause a core melt accident and concluded that SGTRs contribute an insignificant amount to the overall core damage frequency and risk. In other words, the probability of a random tube rupture leading to core damage was less than 1E-10 per reactor-year (RY). Since the frequency of random tube rupture associated with the interim tube repair criteria is likely lower than the value assumed in the Catawba PRA study, it can be concluded that the interim tube plugging criteria contributes an insignificant fraction of the total core damage and public health risk.

B. Transient Induced Rupture

Another potential concern about the interim tube plugging criteria is that if a large differential pressure transient were to occur, the steam generator tubes may be stressed to the point that one or more tubes may rupture. If the transient were to proceed to core damage, fission products would be released to the atmosphere without the benefit of the containment barrier. Also, the loss of primary coolant through the failed tubes would increase the likelihood of core damage.

For this concern, the three component equation must be modified slightly to the following;

Risk = (Init. Freq.)x(Prob. of Tube Failures)x(Prob. Core Damage)x(Conseq.)

For the equations above, the probability of a single tube rupturing is likely very different from the probability of multiple tube ruptures. Also, the conditional probability of core damage and the consequences would be different for multiple tube ruptures than for a single tube rupture. However, to simplify the calculation, conservative values will be used for each parameter in estimating the potential risk.

The first component is the frequency of transients that produce large differential pressures. Table 1 presents a screening analysis to identify the types and frequencies of transients that cause large differential pressures. Table 2 presents some additional information concerning estimated sequence frequencies associated with the most likely transient to result in large differential pressures. These sequences can be characterized as a steam line break with a failure to terminate safety injection prior to the pressurizer becoming water solid. The sequence frequency is approximately 3.3E-4/RY for sequences with approximately 2335 psi differential pressure.

Westinghouse has calculated that if the tubes are exposed to a differential pressure of 2335 psi at the end of the cycle, the conditional probability for a single tube failure is approximately 7E-5. The probability of multiple tube ruptures is limited by the inspection of all tubes; however, for conservatism, we can also assume that this is the probability for multiple tube failures and that it applies for the full cycle.

Next the probability of core damage is estimated, given that there has been a transient that resulted in a large differential pressure, and that one or more steam generator tubes have failed. NUREG-0844 estimated a probability of 1E-2 for sequences of this type and the Catawba PRA estimated a probability less than 1E-6. For conservatism, a value of 1E-2 will be used for the conditional probability of core damage.

The consequences from a tube rupture core melt sequence have been taken from the Catawba PRA, release categories 1.02 and 1.04. The whole-body person-rem risk calculated for these release categories is ~1.37E+07 per event.

From the combination of the above;

Risk = (Init. Freq.) (Prob. of Tube Failure) (Prob. Core Damage) (Conseq.)

Risk = (3.3E-4) (7E-5) (1E-2) (1.37E+07) = 3.2E-03 wbpr

The Catawba PRA has calculated the normal plant risk from internal events to be 2.24E+01 Person-Rem per year. In comparison, the risk of 3.2E-03 Person-Rem per year associated with steam generator tube rupture following a large delta pressure transient is a very small fraction of the total plant risk.

V. Comparison to NUREG -0844

The PRA study done here has estimated that the public health risk associated with the interim steam generator tube repair criteria is on the order of 3.2E-03 Person-Rem/RY. NUREG-0844 also analyzed the public health risk of SGTR. A comparison of the Catawba specific results and the NUREG results can be made by converting the NUREG results to equivalent Whole-Body Person-Rem exposure. Using a conversion factor of 1 latent fatality/15,000 Person-Rem, the NUREG latent fatality total of 1.7E-3 latent fatalities/RY, is equivalent to approximately 25 Person-Rem. The Catawba specific results are nearly five orders of magnitude lower than the NUREG value, for which it was concluded that, "risk from steam generator tube rupture events is not a significant contributor to total risk at a given site, nor to the total risk to which the general public is routinely exposed."

IV. Conclusions

The Catawba interim repair criterion represents a departure from the past practice of explicitly enforcing a minimum wall thickness requirement for the steam generator tubes. A probabilistic risk analysis has been performed to assess the impact of running a full cycle following the use of the interim steam generator tube plugging criteria. Two accident scenarios were identified that could be affected by steam generator tube

structural integrity concerns. This analysis has determined that the core damage frequency and public health risk are low, when compared to the normal risk associated with plant operation as estimated in the Catawba PRA. Since the Catawba PRA did not explicitly consider the existence of degraded tubes, the risk value calculated in this analysis could conservatively be considered as the change in calculated risk due to the existence of tubes that are degraded within the limit allowed by the interim tube plugging criteria; even by such a conservative criterion the "combined risk" is less than that found to be acceptable in NUREG-0844. In actuality, tubes exhibiting some level of degradation have always existed. Therefore, it can not be concluded that the actual operating risk has increased by the implementation of the IPC, it is concluded that the contribution to the calculated risk due to the existence of degraded tubes, as permitted by the IPC, is insignificant.

Also, these risk values have been shown to be much less than the risk values calculated in NUREG-0844, which the NRC has concluded that, "risk from steam generator tube rupture events is not a significant contributor to total risk at a given site, nor to the total risk to which the general public is routinely exposed."

The quantitative result are summarized in Figure 1.

VI. References

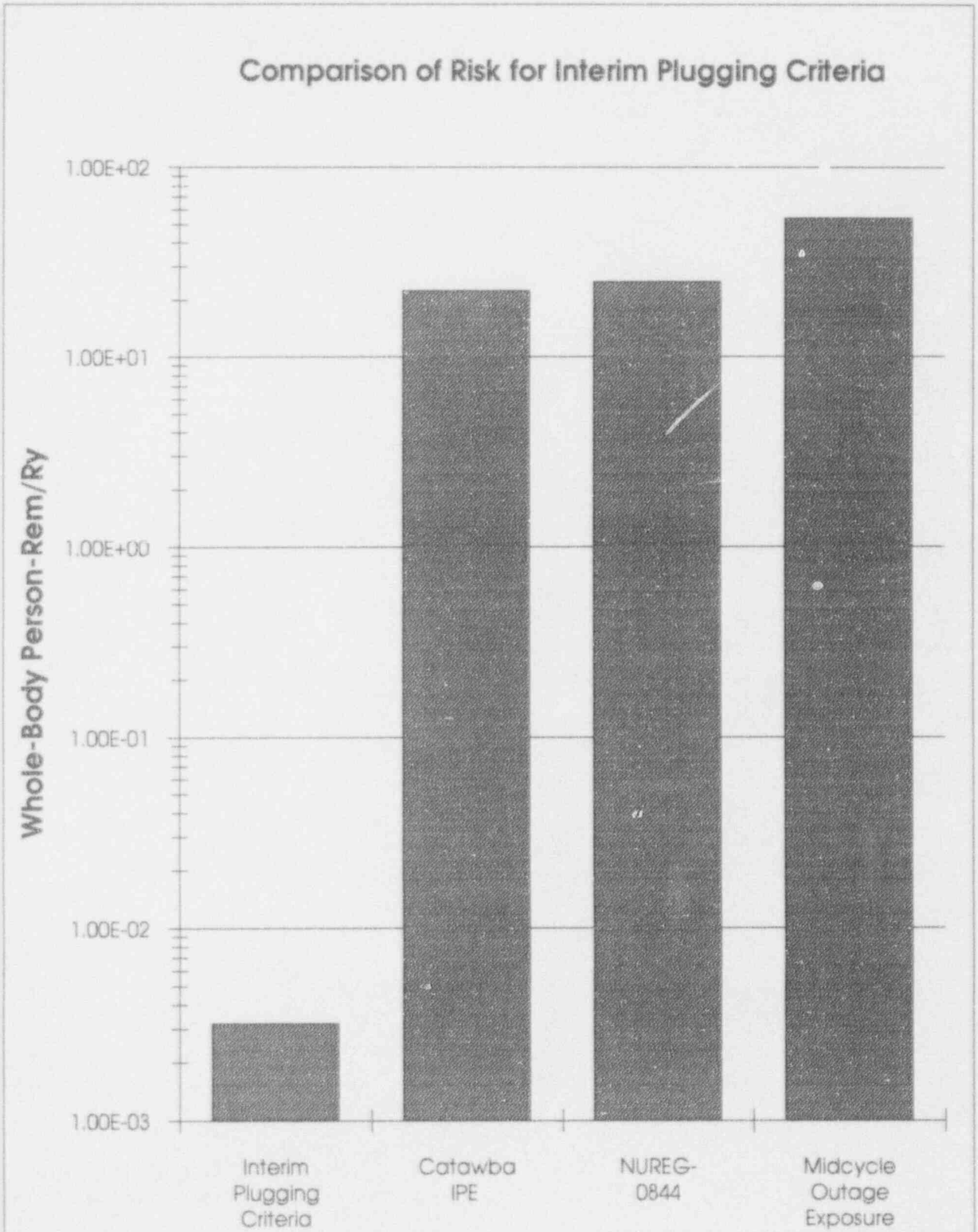
1. Catawba Nuclear Station Unit 1, Probabilistic Risk Assessment Study, Duke Power Corporation, September 1992.

2. Catawba Unit 1, Technical Support for Steam Generator Interim Tube Plugging Criteria for indications at Tube Support Plates, Westinghouse Electric Corporation, WCAP-13494, Revision 1, 1993.

3. NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity, U. S. Nuclear Regulatory Commission, NUREG-0844, September 1988.

Table 1 - Sequences Which May Result in Substantial Pressure Differentials Across the Steam Generator Tubes (Chapter 15 Sequences)

Differential Pressure Across SG Tubes	FSAR Chapter 15 Sequence Description	Estimated Sequence Frequency
2485 psid (pressure is limited by the three pressurizer safety valves)	15.1.5 Steamline Break Outside Containment and 15.2.8 Feedline Break	4.3E-8/RY (See Table 2)
	15.1.4 Spurious Secondary System Valve Opening	Bounded by SLB Frequency
	Total Category	**4.3E-8/RY**
2335 psid (pressure is limited by the three pressurizer PORVs)	15.1.5 Steamline Break Outside Containment and 15.2.8 Feedline Break	3.3E-4/RY (See Table 2)
	15.1.4 Spurious Secondary System Valve Opening	Bounded by SLB Frequency
	Total Category	**3.3E-04/RY**
2150 - 2335 psid (pressure dependent on time of SI termination)	15.1.5 Steamline Break Outside Containment and 15.2.8 Feedline Break	6.7E-4/RY (See Table 2)
	15.1.4 Spurious Secondary System Valve Opening	Bounded by SLB Frequency
	15.8 ATWS	Does not apply to period under consideration.
	Total Category	**6.7E-3/RY**
1800 - 2150 psid	15.8 ATWS	Does not apply to period under consideration.
1450 - 1800 psid	15.8 ATWS	**1.1E-5/RY**
1400 - 1450 psid	15.4.1 Zero Power Control Bank Withdrawal	Screening value 1E-2/RY
	15.4.2 Control Bank Withdrawal at Power	Screening value 1E-2/RY
	15.4.8 Rod Ejection	Some very low value - e
	Total Category	**2E-2/RY**



Comparison of Risk for Interim Plugging Criteria
Whole-Body Person-Rem/Ry
1.00E+02
1.00E+01
1.00E+00
1.00E-01
1.00E-02
1.00E-03
Interim Plugging Criteria
Catawba IPE
NUREG-0844
Midcycle Outage Exposure

Figure 1