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11555 Rockville Pike
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Subject: PWR Owners Group
Transmittal of PWROG Comments on NRC Document: RASP Handbook Consequential Steam Generator Tube Rupture (C-SGTR) Update [Accession Number M18011A027]

Dear M. Franovich:

The purpose of this letter is to transmit PWROG comments on NRC document: RASP Handbook Consequential Steam Generator Tube Rupture (C-SGTR) update.

The NRC has developed a risk analysis procedure for Containment Related Events that provides guidance to the NRC staff in developing Large Early Release Frequency (LERF) estimates. Currently this LERF handbook (Volume 5 of the Risk Assessment of Operational Handbook) contains guidance for only Consequential Steam Generator Tube Rupture (C-SGTR) events. The methodology includes a simplified high level estimate for potential screening and a detailed assessment. The intent of the process is for the NRC senior reactor analyst to estimate the C-SGTR LERF contribution from the CDF results obtained from the NRC Standardized Plant Analysis Risk (SPAR) model.

The process is conservative and uses a generic age related method to estimate Steam Generator Tube (SGT) condition. The crack depth and length distributions are based on a small sample of Steam Generators (SGs) representative of the operating fleet of Westinghouse and Combustion Engineering (CE) plants with thermally treated Alloy 600 and Alloy 690 tubes. The resulting conditional failure rates reflect cycle specific flaw generation and appear to be mean values. The process simplifies the non-Station Blackout (SBO) cases and provides a bounding treatment for them. These sequences can represent about 30% of the C-SGTR contribution to LERF. The primary conservatism in the Risk Assessment Standardization Project (RASP) process arises from the tacit assumption that all C-SGTR events are LERF contributors. While this is an expedient process, detailed MELCOR analyses performed for the State of the Art Reactor Consequence Analysis (SOARCA) program suggest that the C-SGTR releases are limited and, in many cases, considered to be small and/or late releases. Specific comments are provided in Attachment 1.

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The PWROG requests that the US NRC consider these comments in the next update of the RASP Handbook.

Correspondence related to this transmittal should be addressed to:

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If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken J Schrader

Ken Schrader, Chief Operating Officer and Chairman
PWR Owners Group

KS:am

cc: PWROG Management Committee
PWROG Risk Management Committee
PWROG PMO
J. Drake, US NRC
CJ Fong, US NRC

ATTACHMENT 1
WESTINGHOUSE REFERENCE: LTR-RAM-18-91 Rev 0

PWROG Comments on NRC Document:
RASP Handbook Consequential Steam Generator Tube Rupture (C-SGTR) Update
[Accession Number M18011A027]

The PWROG has reviewed "Risk Assessment of Operational Events Handbook: Volume 5 - Risk Analysis of Containment Related Events (LERF) (Currently contains only Consequential SGTR Events)", dated January 2018 [1] under Accession Number ML18011A027. As a result of the review the PWROG offers the following comments and editorial suggestions (see Table 1: Editorial Comments/Suggestions).

General Comments

The proposed Risk Assessment Standardization Project (RASP) process provides a direct method of establishing a conservative estimate of the Consequential Steam Generator Tube Rupture (C-SGTR) contribution to LERF. However, in light of recent MELCOR analyses of many related sequences performed in support of the State-of the Art Consequence Analysis (SOARCA) project, the assumption that all C-SGTR events sequence to a LERF endstate provides an excessively bounding treatment of the estimated ΔLERF metric. These assumptions could overstate ΔLERF findings resulting from the Significance Determination Process (SDP). This is particularly true since detailed MELCOR analyses suggest that many of these sequences result in no LERF impact.

Specific comments on the document are provided below and in Table 1.

Comments on RASP Handbook Consequential Steam Generator Tube Rupture (C-SGTR) Update:

1) **Thermally Induced Steam Generator Tube Rupture (TI-SGTR)**

Section 2.2 Assumptions and Ground Rules of RASP, Item 4 LERF Model states:

> "If C-SGTR and core damage occur, large early release of fission products is postulated. Thus, the numerical value of LERF is the same as the numerical value of CDF followed by C-SGTR.
>
> This applies to all sequences in which both core damage and C-SGTR occur, regardless of which one occurs first"

Given the current state of knowledge in this area, this is a very conservative position. Detailed results of the SOARCA project (NUREG/CR-7110 [2]) indicate that despite the fact that C-SGTRs are bypass events the degree of radiological release to the environment may not be contributors to LERF. Even for the unmitigated short term station blackout (STSBO) with early loss of turbine driven auxiliary feedwater (TDAFW), 24 hour releases of iodine and cesium to the environment were estimated to be below 1% of core inventory. While these events resulted in early core damage and subsequent TI-SGTR, the release of fission products to the environment was predicted to not be "large". This prediction is a consequence stemming from two factors:

1. Plate-out/iodine removal in the SG (even one that has dried out) was significant (with a Decontamination Factor (DF) of approximately 7).

2. Even when hot leg failure does not occur first, thereby preventing the C-SGTR, a subsequent failure of the hot leg is still possible. This failure, results in dispersion of much of the remaining

radiological material into containment. This behavior is predicted in representative SBO core damage scenarios evaluated in support of the SOARCA project. Based on these analyses, the overall release of fission products in the interval from the onset of the C-SGTR and failure of the RCS boundary was also estimated to be below 1% of the iodine inventory with minimal releases following hot leg failure.

2) Secondary Side Break (SSB) Pressure Induced SGTRs (PI-SGTRs)

The consequential SG tube failure probability for Large SSB is given in Tables 2.4-1 and 2.4-2 as 0.02. The document provides no clear basis for 0.02; the value does not appear to be derived from NUREG-2195 [3]. A discussion of the basis should be added to footnote (4).

Footnote (4) notes that the majority of the secondary side break events are small SSBs. As noted in Section C.2.2 of Reference [3], main steam line events that have occurred in practice are typically small leaks that generally are terminated following a manual trip. Thus in assigning the SSB initiating event frequency the analyst should confirm that the selected frequency applies only to the large break and that flow restrictors are not available to limit the depressurization. Small SSBs result in slow depressurization of the RCS, and a gradual depressurization of the secondary side thus reducing the SG tube wall pressure difference compared to the unrestricted large break case.

The process does not differentiate between SSBs inside and outside of containment. These events are considered differently in NUREG-2195. Consider noting (as discussed in NUREG-2195, page C-10) "Breaks inside the containment do not contribute to containment bypass probability, and breakers outside containment could be isolated via MSIVs. Bounding calculations in Section 7.4 also shows that the SLB scenarios followed by pressure induced C-SGTR has small contributions to both large early release frequency (LERF) and to CDF."

SSBs followed by an induced SGTR are expected to respond similarly to a spontaneous SGTR with a faulted SG. NUREG/CR-7110 MELCOR analyses of spontaneous SGTR with faulted SGs (Section 5.4.3) indicate that the releases for unmitigated scenarios occur late with first fission product gap releases from the fuel occurring after 27 hours. Thus, mapping of these events into LERF can be considered overly-conservative. This should be identified within the process notes.

3) ATWS pressure induced C-SGTR

Treatment of ATWS varies from that presented in NUREG 2195 (page 7-72) which notes the following for ATWS events that result in SG tube pressure differences in excess of 3200 psid (unfavorable moderator temperature coefficient (MTC) events):

> "For unfavorable MTC when the pressure exceeds 3200 psi (~22.1 Mpa); rupture of one or more components in primary system and the occurrence of core damage are assumed. C-SGTR is not considered for LERF analysis since most releases will be into the containment through failed primary components."

The likely failure mode for these internal releases is a result of stretching the upper head bolts, creating a leak from the reactor vessel to the containment. While, there is considerable uncertainty in the ATWS assessment, Appendix 2-D includes the contribution of ATWS with failure of pressure relief to be a ~10% contributor to C-SGTR induced LERF. Given the epistemic uncertainty associated with the estimated ATWS contribution some note regarding this uncertainty may be warranted.

In Table 2.4-1/-2 Category Identifiers 4a and 4b include SLOCA and MLOCAs. Since these events by their very nature provide pressure relief to the RCS, they shouldn't be considered in the ATWS bin as contributors to C-SGTRs. Larger MLOCAs may not need control rod insertion to mitigate.

The section covers favorable and unfavorable SG design types, but does not also discuss favorable and unfavorable MTCs. Favorable MTCs can extend a significant part of the fuel operating cycle particularly for less reactive cores. Thus ATWS events occurring with favorable MTCs will not exceed the 3200 psid threshold and have a lower potential of C-SGTR. In assessing the ATWS contribution to LERF it is recommended that the unfavorable exposure time (UET) factor be considered in the assessment.

Section 2.5.1 Overview of RASP further notes that, "For ATWS and Large SSB sequences, the uncertainties are large due to lack of supporting analyses". While the comment is generally correct, ATWS results of deterministic success criteria assessments for Westinghouse plants is available in WCAP-15831-P-A [4]. These analyses provide useful insights into the grouping of events within the spectrum of ATWS transients.

4) Loop Clearing Flowrate

Section 2.2 Assumptions and Ground Rules of RASP, Item 5 RCP Loop seal clearing identifies 300 gpm/ pump as the RCP seal leakage sufficient to clear the RCS loops seal. Note that for current generation RCPs, flowrates in this range are low probability and are associated with failure of all RCP seal stages. To facilitate the guidance, the following clarification is suggested:

> For plants with the low-leakage RCP seals, it would be appropriate to assume that the loops are not cleared for scenarios where the seals do not fail. For plants without low-leakage seals, loop seal clearing should only be considered for scenarios that include a failure of all RCP seal stages.

5) Comments on Example Application Appendix 2-D

Event 8 identifies a two event sequence for ATWS, CEAs do not insert (RPS) and RCS is pressure limited. However, no failures of AFW or PORV operation are noted. Can some additional explanation be provided as to the basis for mapping this event as core damage? The authors of Appendix D map this into a bin with failure of pressure relief.

In evaluating ATWS contribution to CDF, do the SPAR models consider UET factors?

Why is sequence 36 mapped to category ATWS, while Sequence 8 (also an ATWS) is mapped to Category 4?

Example estimates Category 2 contributions, but the discussion and the basis for the table values are unclear.

Table D-4 includes plant name in some of the descriptions. Plant name should be removed.

References

[1] "Risk Assessment of Operational Events Handbook: Volume 5 - Risk Analysis of Containment Related Events (LERF) (Currently contains only Consequential SGTR Events)", dated January 2018. [Accession Number M18011A027]

[2] NUREG/CR-7110, Vol. 2, "State of the Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis," Sandia National Laboratories, January 2012.

[3] NUREG-2195, "Consequential SFTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes," USNRC DRAFT, March 2017.

[4] WCAP-15831-A, WOG Risk-Informed ATWS Assessment and Licensing Implementation Process, Westinghouse Electric Company," August 2007.

Table 1: Editorial Comments/Suggestions		
CMT #	**Section**	**Comment**
1	S 2.1.1, 2nd ¶	Consider rewording to focus on failure mechanisms and not tied to DBAs. For example, "C-SGTRs can result from one of two mechanisms: pressure induced or temperature induced. Pressure induced SGTRs arise from a spectrum of plant transients that, during the course of the event expose SGTs to pressure differences across the tubes that are significantly greater than the nominal SGT pressure load." These events include: steam line break, feed line break, stuck open SG safety valve or atmospheric dump valve, and anticipated transients without scram. Pressure loads resulting from these conditions have a remote potential of propagating to a SGTR and should a tube fail, in the absence of other failures these events do not proceed to core damage and therefore would not result in large radiological releases."
2	S 2.1.1, 3rd ¶	Consolidate statements on severe accident related SGTRs for clarity: for example, thermally-induced SGTR refers to those events caused post-core damage mainly due to creep rupture of the steam generator tube caused by the high pressure superheated steam exiting the core in the presence of a dry, depressurized steam generator. These severe accident sequences are referred to as HDL (or H/D/L) which stands for high primary pressure and dry steam generator(s) with low secondary side pressure.
3	S 2.1.1, 4th ¶	"NUREG 1750" should be "NUREG 1570"
4	S 2.1.1, 4th ¶	"The method focuses on estimating the C-SGTR frequency, and its <u>potential contribution to</u> LERF based on analyzing individual dominant accident sequences, whenever possible."
5	S 2.2 2nd item under 1.	For thermally-induced SGTR, where core damage already occurred, C-SGTR is defined as the SGTR that occurs prior to failure of an "other RCS component" <u>capable of depressurizing the RCS</u>, such as hot leg or surge line.
6	S 2.2 3rd item under 3.	Consider replacing "will" with "may"
7	S 2.2 5th item	Discussion states "Loop seals are more likely to clear when the water in the loop seals is heated and a rapid depressurization occurs". Isn't it more of a pressure imbalance between the core and the primary side of the SG which is exacerbated with leakage via the RCP seals?
8	S 2.2 5th item	Second sentence notes that "RCP seal leakage of 300 gpm/pump …is modeled as a cleared loop seal." Note that for qualified seals (which are the only ones that remain in the fleet) the only leakage condition which exceeds this level results from a failure of all RCP seal stages.
9	S 2.2 6th, sentence, next to last ¶ on item 5.	Note: "300 gpm" should be "300 gpm/pump" or "from a single RCP seal"
10	S2.2 summary paragraph	These leakage <u>sequences typically represent conditions with the failure of all seal stages and</u> are generally well delineated in PRA studies.

Table 1: Editorial Comments/Suggestions		
CMT #	**Section**	**Comment**
11	S2 summary paragraph	"Such sequences are assumed to lead to a consequential steam generator tube rupture end state".
12	Table 2.4-1	Note 3: 0.5 should be 0.05 to align with table and discussion