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Subject: PWR Owners Group, Risk Management Committee
PWR Owners Group Transmittal of the PWROG Comments on NUREG-2195, "Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes," per PA-RMSC-1292

The purpose of this letter is to transmit the Pressurized Water Reactor Owners Group (PWROG) comments on NUREG-2195 which was available for public comment. The comments are included in Attachment 1, and are a deliverable from the PWROG PA-RMSC-1292 program.

The attached comments are considered non-proprietary. The comments are intended to provide suggestions to improve the clarity and completeness of NUREG-2195.

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 (205) 992-7037 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow
Chairman and Chief Operating Officer
PWR Owners Group

NJS:WAN:cah

Attachment 1: PWROG Comments on NUREG2195.pdf

cc: PWROG PMO
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ATTACHMENT A

Comments on the Draft Report for Comment Version of NUREG-2195, "Consequential SGTR Analysis for Westinghouse and Combustion Engineering Plants with Thermally Treated Alloy 600 and 690 Steam Generator Tubes"

The PWROG has reviewed the draft NUREG and is providing the following general comments.

The report provides considerable valuable new information on consequential SGTR; however, as a result of several conservative or incorrect assumptions made in performing the assessment, care should be taken in using the results outside of the context of screening. This is of concern for the treatment of CE PWRs.

A.1 USE OF MELCOR

Over the past decade there have been ongoing discussions in the technical community regarding the ability of SCDAP/RELAP5 [1] to model the thermally induced SGTR issue. These efforts included comparisons of SCDAP/RELAP5 with various versions of MAAP. During that time, many improvements were made to SCDAP/RELAP5. It was believed that these improvements impacted the ability to correctly predict the outcome of the failure rate between the hot leg/surge line and the flawed SG tube. While MELCOR predictions are similar to that of SCDAP/RELAP5, differences between the codes can impact relative timing assessments of thermally induced failures (See Reference 1). This may impact event timings, the structural failure sequence and potentially C-SGTR failure probabilities. Thus, using MELCOR introduces unnecessary complications in interpreting TI-SGTR outcomes across the plant designs.

A.2 CALVERT CLIFFS MELCOR MODEL

The model used for Calvert Cliffs appears to have several assumptions that may lead to questionable plant responses that could impact conclusions drawn in the assessment. The following subsections list these assumptions.

A.2.1 CC RCP Leakage is 21 gpm/RCP

This leakage value is not typical of CE PWRs. With controlled bleed-off (CBO) not isolated, leakage rates would be on the order of 2 gpm. If CBO is isolated, the leakage would be lower. The increased leakage results in more rapid depressurization of the RCS.

(In order to estimate the potential impact of the thermal hydraulic assumptions, Calvert Cliffs SBO results using the PCTRAN code was reviewed and compared with the early pre-core uncovery predictions from MELCOR. RCS pressure differences are illustrated in Figure A-1 and Figure A-2. Both cases suggest that the 21 gpm /pump assumed RCP leak rate resulted in a high system depressurization. The impact on core uncovery was assessed for early and delayed SBO. MELCOR prediction estimated early and delayed SBO core uncovery times of about 210 minutes and 450 minutes, respectively. The corresponding PCTRAN predictions are

270 minutes and 600 minutes, i.e., the core uncovery transient appears to be accelerated in the MELCOR model.)

A.2.2 SIT Pressure is Estimated at Around 700 psig

Calvert Cliffs Technical Specification Bases 3.5.1 indicates that SITs are to be maintained between 200 and 250 psig. The discussion of the analyses suggests that analyses were performed at 700 psig. A more realistic assessment of Calvert Cliffs SIT discharge would be closer to 200 psig. It is noted that this setpoint stipulation does not impact incipient failure timings.

A.2.3 Limited Details are Provided Regarding Other Calvert Cliffs Inputs

Given that the above fundamental inputs were not properly modeled for Calvert Cliffs, it would seem that more modeling details should be provided for review (perhaps another appendix) and inputs should be reviewed by the PWROG.

A.2.4 Treatment of Hot Leg Creep

Pages 42 /43 of the draft NUREG note that only one creep model could be included in the MELCOR model despite that the hot leg includes both stainless steel and carbon steel constituent layers. The stronger of the two materials was used to model creep failure. This has a significant impact on results. What appears to be discussed later is that these calculations were not used in the hot leg material failure model. Some reference to that discussion should be provided in Section 3.4 of the NUREG.

A.2.5 Steam Generator Design

Calvert Cliffs analysis appears to have been performed for the CE-67 BWI replacement SG. This design is typical of Millstone Unit 2 and St. Lucie Unit 1. These plants are all typical of the CE-2700 MWt design. However, other CE units have different replacement generator designs from different manufacturers along with different fuel and upper plenum designs and power levels. To what extent will these factors influence the TI-SGTR failure likelihood for the non-analyzed designs?

A.2.6 In Order to Address the Above, the Following is Recommended

(1) Model changes made to SCDAP/RELAP5 in order to better simulate these transients as a result of MAAP/RELAP5 comparisons be reviewed in the context of MELCOR calculations so that the NRC can confirm that MELCOR has the necessary models such that predictions of MELCOR and SCDAP/RELAP5 would be close. A direct comparison between MELCOR and the other two codes for the SBO scenario would be helpful.

(2) The MELCOR CE parameter file be reviewed by the PWROG to confirm key plant inputs relevant to C-SGTR modeling (e.g., SG tube thickness, hot leg geometry, SIT injection pressure, and RCP seal leakage assumptions, etc.).

A.3 C-SGTR CALCULATOR AVAILABILITY

NRC provides several methods for assessing the failure probability on a plant-specific basis. The more complex method is said to be needed for CE PWRs. As the existing results may not be generally applicable for CE PWRs (as is noted in the text), it is not clear how NRC will use these results. If NRC intends on having the utilities upgrade the LERF models with this methodology, it would be helpful to know if NRC plans to release its SG tube frequency calculator software to the industry to allow these plants to benefit from availability of this tool so that they could perform plant-specific assessments using a MAAP driving parameters, if desired.

A.4 FLAW DATA

It is noted that the assessment uses limited flaw data based on in-service inspection data. Is there any intent to perform a broader review of this information? Why weren't more plants used in developing flaw distribution?

Are there plans to augment the flaw distribution with data from plants that have been decommissioned/shutdown?

A.5 DETERMINISTIC STRUCTURAL EVALUATION

a) NUREG-2195 would benefit from a short overview section summarizing the deterministic evaluation performed and identify how the deterministic analyses were used within the probabilistic framework.
b) Have benchmark studies been performed on the finite element analyses (FEA) and computational fluid dynamics (CFD) tools used for the assessment?
c) Section 4.2.1 of NUREG-2195 discusses surge line modeling. Please clarify, are stratification conditions taken into account in the surge line creep failure assessment? The section does not discuss this topic.
d) Section 4.3 of NUREG-2195 discusses SG lower head model. Was a divider plate modeled in the FEA for the SG lower head? If not please provide justification.
e) Weld overlay analysis in Section 4.4.6.1 of NUREG-2195 should account for the welding residual stresses of the weld overlay process. Are any residual stresses considered in the present analysis?
f) Note that some of the PWR reactor vessel nozzle dissimilar metal welds Alloy 82/182 (susceptible PWSCC) have applied the Mechanical Stress Improvement Process (MSIP®[1]) to redistribute the welding residual stresses and reduce susceptibility to PWSCC. Would this have any impacts on the SGTR evaluation?
g) Was PWSCC crack growth considered for Alloy 600/690 tubes? If not, please justify treatment.

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[1] MSIP® is a registered trademark of NuVision Engineering.

h) Where does K<15 and K>15 in table 6-2 come from. Why is K=15 selected as the threshold?

A.6 SPECIFIC COMMENTS

Additional detailed comments are provided in the table below.

Specific Comments on Draft NUREG-2195	
Page 1-2, 3rd Para. Line 21	The text notes that "although SGTRs have previously been considered in risk analyses, C-SGTRs have typically not been considered" This statement is incorrect. The PWROG developed a Level 2 model guidance report (WCAP-16341-P [4]) which explicitly includes thermally induced and pressure induced SGTR failures. (See "Westinghouse Owners Group Simplified Level 2 Guidance," R.E. Schneider, J. Armstrong, PSA'05, September 11-15, 2005.) C-SGTR was based on flaw data available in NUREG/CR-1570 [3]. This approach is used in LERF assessments of many PWROG members.
Table 2-1, Third Row	First column. Main steam line break or inadvertent opening of safety relief valve will tend to cool the RCS as well, resulting in lower SG tube delta pressures (closer to 1500-1700 psi).
Table 2.1-1, fifth row	The ATWS challenge is associated with "unfavorable" moderator temperature coefficients.
Table 2.2-1	RCP leakage for base SBO analyses set at 21 gpm per pump. This is typical of a Westinghouse plant without SHIELD®[2]. CE plants with Flowserve or similar seals and Westinghouse plants with SHIELD® mechanical seals or Flowserve pumps or CE plants have very low leakages following SBOs.
General	Loss of batteries is assumed to result in loss of indication of SG level and overfill of SG resulting in failure of TDAFW. FLEX equipment is intended to operate following an extended loss of all AC power. Given the investment in implementing these backup systems, this feature should be discussed as potential alternate scenarios.

[2] SHIELD® is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

Specific Comments on Draft NUREG-2195	
Section 3.4	The report discusses the modeling of the two regions hot leg as being needed for a realistic representation of the CE hot leg. While not clearly discussed. the CE plant hot legs have a stainless steel clad liner and a carbon steel pipe. As only one property could be used in the model, it was determined that the most limiting of the two be considered. Thus, the assessment used the minimum creep failure for either material. It was noted that this resulted in a delayed hot leg failure time of 2.5 hours. It is likely this will have a significant impact on the CE conditional failure rates, and a determination on the impact of the C-SGTR probability should be estimated if it is anticipated these values be used in regulatory analysis.
Page3-50	Page 3-50 it is stated that "*The difference in the prediction HL failure timing was found to vary greatly simply by the assumption of material (stainless or carbon steel) – approximately 2.5 hours. Because the SG calculator and FE calculations are providing more precise estimates of component failure timing, updating the HL creep modeling within MELCOR was not prioritized over other modeling aspects that provide information not available from other sources.* *While this difference in failure timing is not directly applicable as an additional uncertainty in failure timing for this analysis it does underscore the importance of using the correct creep-rupture-related material properties. It indicates that this material property can make the difference of whether a SG tube or a HL fails first.* " Confirm that MELCOR creep estimates were not used in the failure model. Reference this discussion in Section 3.4
Section 4 Figures (various)	The following figures do not have axis labels (Figures 4-3, 4-5, 4-6, 4-7, 4-8, 4-17, 4-25).
Page 4-41	Section 4.4.6.1 1st line "pressurized" should be "primary". (Primary water stress corrosion cracking (PWSCC))
Section 5.2.1.1.1	Report notes that Argonne National Laboratory (ANL) developed a model for axial part through wall flaws. Please provide reference for the ANL contribution.
Section 5.2.2.1.1	Provide reference for ANL test
Page 5-5	Recommend that the reference literature on the data for creep rupture be expanded.
Page 5-5	Model assumes creep failure based on the 95% L-M creep rupture parameters. Would conclusion be changed if mean values were used.
Page 5-15	Figure 5-9(b) does not show model predictions.
Page 5-20, line 24	Typo. Inadvertent inserted 9.
Page 5-21	Please provide references for Idaho National Laboratory (INL) data.

Specific Comments on Draft NUREG-2195	
Section 5.2.1.2	Provide a figure that shows the angle "β" & "θ" for the circumferential flaws.
Section 7.1 first bullet	NRC notes that MELCOR natural circulation predictions are compared to CFD models. CE plants have performed natural circulation tests. Is there interest in confirmation of predictions against test data?
Section 7.1.8 (line 25)	Section notes that W results should not be considered as "generic results". The statements notes that the results apply to a specific SG design, configuration, and geometry of the plant systems and specific hot leg and surge line connections. Given that statement, what is intention of Westinghouse analysis. Are TI-SGTRs to require plant-specific analyses with generator specific results?
Page 7-55	Text in paragraph below Figure 7-22, timings used for discussion off a factor of ten. Please check values.
Page 7-29 (and other locations)	700 psig does not apply to Calvert Cliffs.
Page G-4	NRC notes low LERF contribution due to availability of 2 TDAFW pumps. Current plants also have additional SBO DGSs and new FLEX equipment.
Page 7-66	PRW should be PWR
Section 7.3.1.2	What is purpose of sensitivity? The RCS hot leg is a clearly defined parameter.
Page 7-10, Section 7.1.3	Section references 6.1.1. No such section.
Figure 7-8	Legend SG-> SG Tube
Section 7.1.4.5	Referenced sections at beginning of section should be 7.1.4.1-7.1.4.3

A.7 REFERENCES

1. Kenton, M., "Severe accident analysis of a PWR station blackout with the MELCOR, MAAP4 and SCDAP/RELAP55 codes," Nuclear Engineering and Design, Volume 234, 2005, pages 129-145.

2. Constellation Energy Document, C0-SC-002, "PCTRAN Simulations," January 2010.

3. U.S. Nuclear Regulatory Commission NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," March 1998.

4. Westinghouse Report, WCAP-16341-P, Rev. 0, "Simplified level 2 Modeling Guidelines," November 2005.

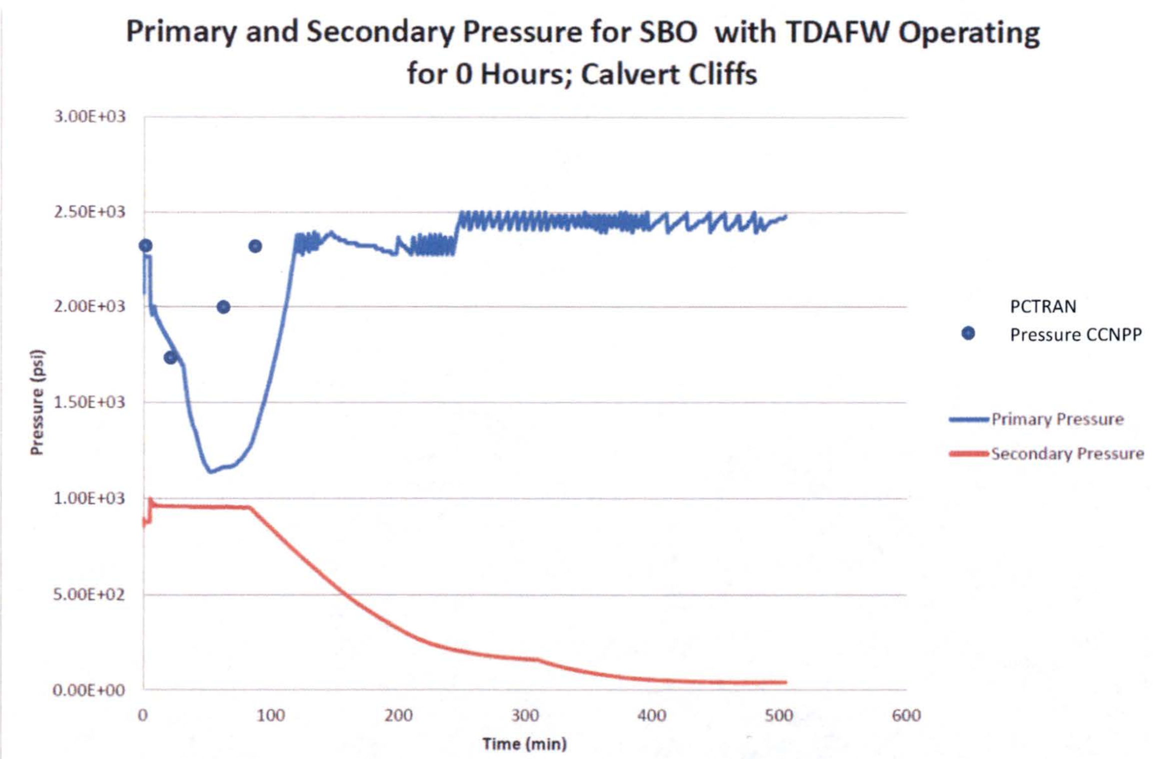

Figure A-1 **Comparison of MELCOR predictions with CCNPP PCTRAN Analytical Predictions (Early onset SBO)**

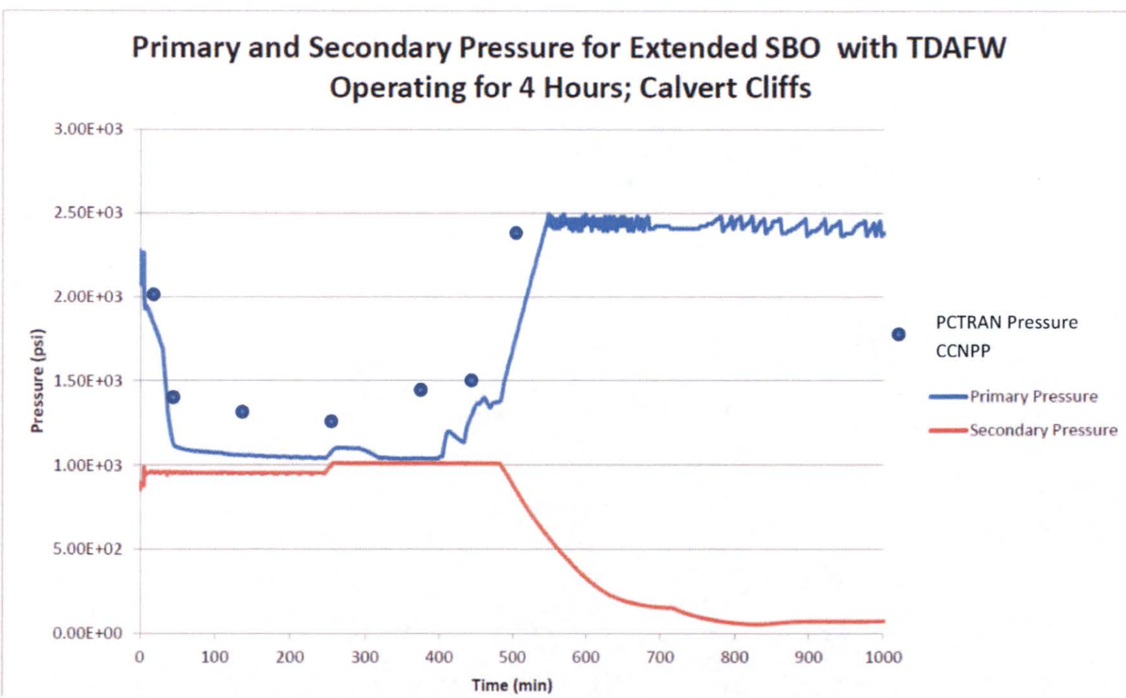

Figure A-2 **Comparison of MELCOR predictions with CCNPP PCTRAN Analytical Predictions (Delayed SBO)**