November 9, 2016

Responses to Various ACRS Member Comments (of April 2015) on the Draft C-SGTR Report

The attached table contains the responses to the ACRS comments of 2015, as of August 26, 2015. The table was edited in October 2016 without affecting the nature of the responses.

Appendices J, K, and L are added to the report as a results of these questions/responses. The draft version of the report issued for public comments (ML16134A029) in 2016 includes these appendices and those responses labeled as YES in the last column of the table.

Table 1. Summary of Comments on C-SGTR Subcommittee Meeting and Materials Reviewed

	Comment	Member	Response	Inserted into NUREG?
1	Analyses for the Westinghouse plant were performed with SCDAP, and analyses for the CE plant were performed with MELCOR. The draft NUREG stated that this was done to allow source terms to be calculated for future risk assessment evaluations. However, these codes have different 'built-in' assumptions that affect core heatup and degradation and heat transfer to steam that is transferred to RCS piping. SCDAP calculations were documented in a NUREG-CR report; but no detailed documentation for the MELCOR CE analyses was provided. Staff should provide the required supporting documentation and demonstrate that 'switching' codes did not affect conclusions related to the CE plant.	Banerjee Powers Shack Rempe	The potential changes that could result from switching codes, was a significant concern throughout the project. Analyses were run with the specific purpose of evaluating code differences. In deciding on the code to use for the project, the benefit of evaluating fission product releases was considered more important than the issues that could arise from switching codes. Supporting documentation, including the initial Sandia National Laboratories report that compares the MELCOR results to those of the equivalent SCDAP/RELAP analyses, has been provided. The comparison between SCDAP/RELAP and MELCOR results for this analysis can be found in the Sandia Report (D. Louie, et al. “A MELCOR Model of the Calvert Cliffs Two-Loop Pressurized Water Reactor and Containment for the Steam Generator Tube Rupture Scenarios”, Sandia National Laboratories, October 2012) on pages 65 to 89. The potential affects from different 'built-in' assumptions that affect core heatup and degradation and heat transfer to steam that is transferred to RCS piping were not explored. This analysis was described in the existing text in the original report:	NO

	Comment	Member	Response	Inserted into NUREG?
			"SNL compared results from the new deck against those generated using SCDAP/RELAP (Reference 1). This comparison required some modifications from the base version to more closely match the SCDAP/RELAP deck. They found that both codes predicted a similar sequence behavior and timing although some later events occurred at somewhat different times. The analysts also found that component failure was not similarly predicted which is not surprising considering that the hottest tube calculation was not included in the MELCOR analysis." No additional text was added to the report.	
2	CFD calculations should be well-documented for both the Westinghouse and CE plants. Although there is a NUREG/CR report for the Westinghouse plant, there is no such documentation for the CE plant. Please provide supporting documentation for the CE CFD analysis.	Rempe Shack	The CE plants utilized the same modeling approach and assumptions that are documented in NUREG-1922 for the Westinghouse plants. The goal of the CE analysis was to re-run the analysis and methods used for NUREG-1922 with the geometry for a CE plant. Assumptions for the modeling approach and the suggested application to system code models are documented in NUREG-1922. A paper documenting the CE modeling is referenced in the final NUREG.	NO
3	Why was the EPRI correlation used for estimating hot leg and surge line failure in the calculator (rather than results based on SCDAP analyses)?	Rempe Shack	This was a deliberate decision to shift away from the use of stress multiplier in lieu of the failure probabilities for the SG tubes, hot leg, and surge line. While a rigorous hot leg failure modeling and estimation of failure times effort was ongoing in parallel, the PRA task used the readily available EPRI correlation to estimate C-SGTR fractions consistently between both the W and CE cases. It is expected that when the more rigorous calculations are done, they would confirm that values used in the PRA are at reasonably bounding, without distorting the insights.	NO

	Comment	Member	Response	Inserted into NUREG?
			This explanation is not intended to go into the draft NUREG.	
4	During the discussion of slide 11, it was observed that the pressurizer should not be shown with a liquid level. However, the staff indicated that in some of the analyses, there were times when the pressurizer did have a liquid level. Such unexpected phenomena should be investigated. Again, if the underlying MELCOR analyses were documented, it would provide more confidence in the results.	Skillman	Pursuing the cause of the filled pressurizer as predicted by code including verifying against other code models and flooding relations was planned. This item was dropped when the scope of the project was reduced. The comparison of the pressurizer liquid level with SCDAP/RELAP (Fig 4-10 on page 83 of Sandia report) shows that SCDAP/RELAP also predicted refilling of the pressurizer with liquid water for the CE plant. The MELCOR analysis does predict a slower drop in pressurizer water level than the SCDAP/RELAP analysis. The SGAP Westinghouse analyses run with SCDAP/RELAP and described in NUREG/CR-6995 also predict a filled pressurizer with a partially voided RCS. This can be determined from the fact that the pressurizer is water-filled (Fig 4.7 on page 62) at the same time that the core is dry as indicated by the presence of superheated gas in the RCS (Fig 4.18 on page 71). The mechanism for a filled pressurizer with a partially voided RCS is as follows: Gas flows through the surge line and pressurizer to RVs at a rate sufficient to flood water in the surge line but not sufficient to flood the water within the larger diameter of the pressurizer itself for both plant designs. As long as a sufficient steam flow exists to flood the surge line the pressurizer water does not drain out. As the water boils off in the core, the water level drops thereby reducing the steam flow to the extent that the pressurizer begins to drain. Some pressurizer draining also occurs during intermittent RV	NO

	Comment	Member	Response	Inserted into NUREG?
			closures. Significant condensation in the pressurizer could potentially contribute to maintaining the pressurizer water level. Code-calculated condensation rates and relief valve discharge rates can be verified. Note that, although the surge line is depicted in drawings and the animations as a straight run, the actual surge line is quite long containing a few bends and an extended horizontal section. Additional documentation of the MELCOR analyses has been provided to the ACRS members in a recent transmittal. No changes were made to the report.	
5	The approach taken for the hottest tube analysis (e.g., the number of tubes assumed to be in the hottest location, the justification for this assumption, and how it was implemented) for CE plants should be better documented. The Westinghouse analyses relied on Figure 17 of NUREG 1922; but the CE approach apparently relied on the expert opinion of several NRC staffers. The added discussion should also provide insights about the required area to ensure that SG depressurization will occur.	Bley Corradini Rempe	It should be stressed that: 1. The method for determining the hottest tube was the same as for Westinghouse. The hottest tube temperature came from the CFD analysis in the same manner as for the Westinghouse analysis, not from expert opinion. 2. The expert opinion was for how many tubes would fail. In actuality the number of tubes that reach the failure criteria changes with time. For the Westinghouse analyses only one, flawed tube is likely to fail whereas, for the CE design considered, multiple tubes fail using the standard failure model. Documentation describing the approach taken to calculate the	NO

	Comment	Member	Response	Inserted into NUREG?
			hottest tube has been provided. The major impact of the number of tubes assumed to fail is that, if enough tubes fail, the primary system depressurizes before the hot leg fails resulting in substantially more releases to the environment. Further increasing the number of tubes that fail beyond that point do not significantly affect thermal hydraulic behavior. The number of tubes failed was used in the MELCOR calculations to test system feedback. The nominal value used was 20, which was within the range provided. The criteria used to make this choice is that it was consistent with (potential values of 10 and 100 were quoted). The uncertainty bands in NUREG-1922 fig 17 illustrated some variability in the normalized tubesheet inlet temperature distribution. The text also indicated that a direct application of the distribution for the hottest tube would be somewhat conservative because the hottest section of the plume shifted resulting in substantial temperature fluctuations near the hottest section of the plume. Features of the Westinghouse SG designs that substantially promoted this temperature fluctuation were the distance from the hot leg to the tubesheet and the fact that the hot leg flow enters the inlet plenum at an angle relative to the plate separating the inlet and outlet plena. The flow enters the CE SG normal to this plate and much closer to the tubesheet resulting in substantially less fluctuation of the plume location. Consistent with failure conditions in earlier TH analyses and	

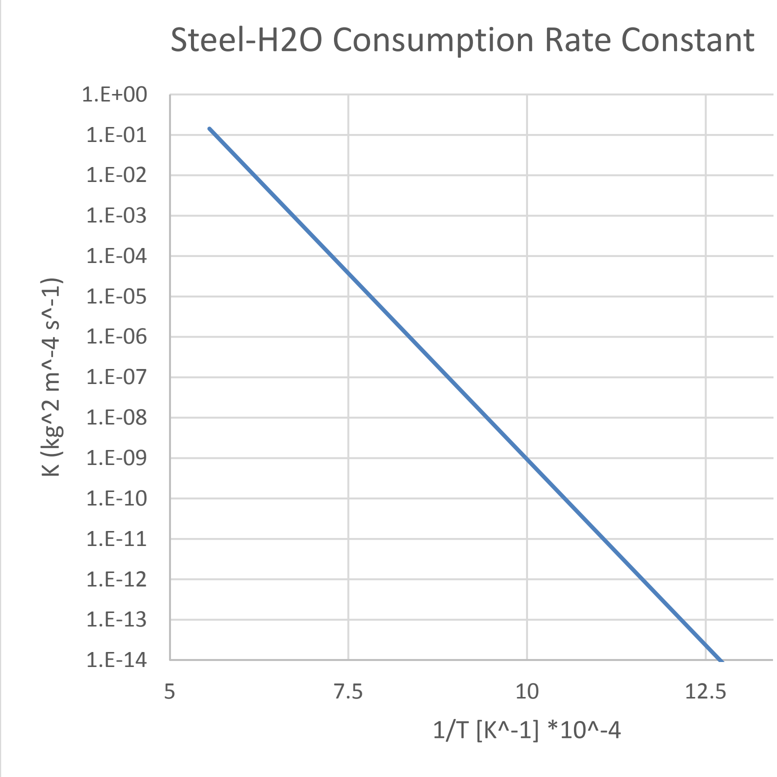
	Comment	Member	Response	Inserted into NUREG?
			with those of the hot leg a failure was assumed to result in a double ended guillotine break. For the evaluated scenarios in which the tubes failed first, the failure of 20 tubes was sufficient to depressurize system before hot leg failure. The MELCOR input was set up to allow the number of failed tubes to be varied from 1 to 100 in order to allow evaluating the number of tube failures required to depressurize. This analysis was not performed because of the reduction in scope. The assumption of number of hottest tubes was intended to be used solely for TH feedback and not for the tube temperature distribution to be combined with flaws to calculate failure. Since a distribution was unavailable at the time the calculator simulations were conducted, some assumption had to be made regarding the choice of this temperature distribution. The existing assumption for the TH analysis was considered reasonable. It can be seen in the CFD visualizations that the temperature distribution is continuous, not discrete. Ideally a continuous temperature distribution from the hottest to coldest tube would be used to evaluate failure. The MELCOR analysis only provides discrete temperatures (at a given elevation) hottest tubes, average-hot tubes, and cold tubes. The shape of the temperature distribution from CFD can be superimposed on the transient behavior calculated by MELCOR to obtain a time-variant continuous temperature distribution. As long as system temperature continues to increase and system pressure remains high, additional tubes would continue to reach the same failure condition as that of the	

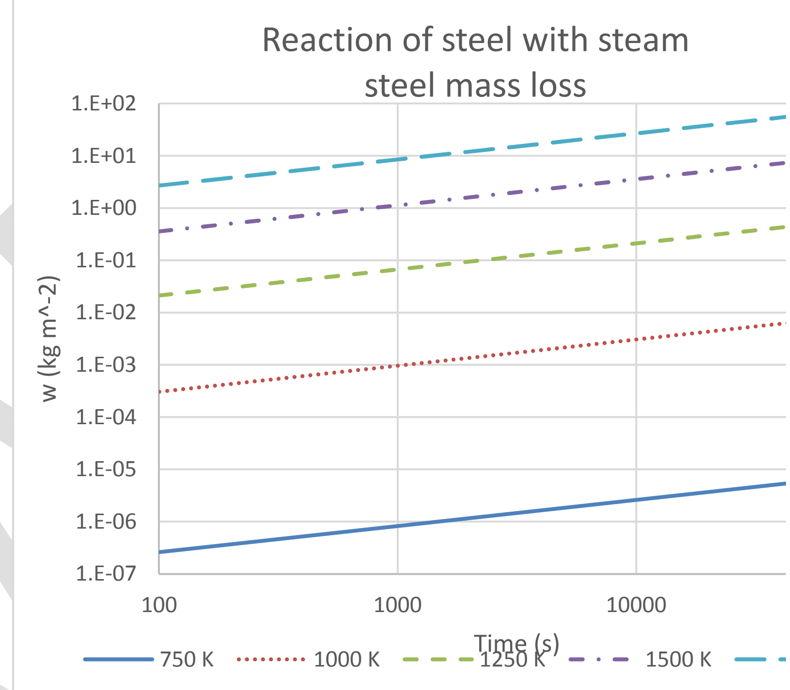
	Comment	Member	Response	Inserted into NUREG?
			hottest tube upon failure. A hand calculation of the rate of nominally-unflawed tube failure was subsequently performed by assuming a shape of the SG tubesheet inlet temperature distribution* and evaluating the rate at which tubes reach the same failure temperature as the hottest tube at the calculated failure condition. For the assumed shape of the temperature distribution entering the tubesheet, the calculated rate of subsequent tube failure ranged from one failure every 5 seconds to one failure every 2 seconds. *A temperature distribution at the tubesheet inlet had been generated for a Westinghouse design in NUREG-1922. A similar method is used to predict the temperature distribution at the tubesheet inlet for CE plants. This is documented in reference 7 of chapter 3. A shape that visually approximates the temperature field as calculated by the CFD analysis was used to estimate the tubesheet inlet temperature distribution and the spatial temperature distribution within the tubes. A reference to the updated CE plant CFD modeling is included in section 3 of the report (ref. 7).	
6	Assumptions related to relief valve (e.g., SRV and PORV) performance when exposed to temperatures expected during a severe accident are not supported with data. This should be explicitly noted.	Corradini	OK – the following text was added to the document: "The valve failure criteria were varied; not based on failure data, but rather in order to evaluate the possible failure criteria that would possibly result in fission product releases to the environment. During the analyses for the SGAP tube failure was the criterion used to consider that containment	YES Section 3.6.1

	Comment	Member	Response	Inserted into NUREG?
			had been bypassed. The initial CE simulations indicated that, if tubes failed while the steam generator secondary side was depressurized, the secondary-side relief valve opened for a short period before closing (if they opened at all) resulting in a small amount of fission product releases to the environment. This behavior may be scenario dependent."	
7	Melting temperatures assumptions for Inconel and stainless steel presented by the staff appear high. In addition, result plots should note when materials are failed (current plots show molten materials in the hot leg, surge line, and SG tubes. Likewise, the impact of SS or carbon steel oxidation in steam on analysis results should be considered (it wasn't clear if MELCOR considered this oxidation in the analysis). Some types of iron oxide liquefy at temperatures as low as 540-600 °C. Dr. Powers indicated that steel oxidation would be precluded if there were small amounts of hydrogen present; however, it is not clear that this was considered or if the hydrogen would be present in locations of concern.	Rempe	The melting temperatures presented in the slides for stainless steel and Inconel (1725 K (1452 °C)) originate from the SGAP analysis. These temperatures are consistent with those listed in the RELAP/SCDAP (1671 - 1727 K) and MELCOR (1700 K) manuals. These melting temperatures are consistent with typical listings of LWR melting temperatures such as that shown in NUREG/CR-6042 (R-800 course material). The lowest melting temperature for iron listed in this are for eutectics with Zr (~940 °C) and B_4C eutectics (~ 1150 °C). Steel reactions with Zr and with B_4C are modeled in MELCOR. Steel oxidation of RCS components are typically not considered in severe accident analyses. Oxidation of RCS components was not considered during the SGAP analysis. The influence of oxidation of core components was analyzed during the SGAP. It was concluded that variations in oxidation of additional metal affect absolute failure timing but do not significantly affect the relative failure timing of different components which is of interest for evaluating whether the containment is bypassed.	YES Sentence added at the end of Section 3.6.1; Appendix I is added.

	Comment	Member	Response	Inserted into NUREG?
			MELCOR contains a steel oxidation models but they applied in components in the COR module rather than the HS (Heat Structure) module use to model the RCS piping. The effects of oxidation are analyzed below to assess the possible impacts of oxidation in the RCS. The MELCOR steel-H2O oxidation model was used. External sources for steel oxidation or steel oxide melting were not sought since it is expected that the major oxidation mechanisms should have been captured during the study of degradation of steel present in the reactor core. The steel-H2O rate constant in MELCOR is calculated using the following equation. $K(T) = 2.42 * 10^9 * \exp(-42{,}400/T)$ The analysis is continued assuming that the steel-H2O rate constant listed in MELCOR applies to units of kg and m^2. This was verified in the literature. A paper by the same author as the primary reference in MELCOR (J. F. White)* but published 3 years after the MELCOR reference lists the following parabolic rate constant: $w^2/t = 2.4 * 10^{12} * \exp(-84{,}300/(RT))$, Where w is the weight gain (in fact the mass of oxygen added to steel) per unit area in mg/cm^2, R is the gas constant in cal/(mole-K), T in K, and t in s. Applying the universal gas constant of R = 1.987 cal/(mole-K)	

	Comment	Member	Response	Inserted into NUREG?
			the equation becomes: $w^2/t = 2.4 * 10^{12} * \exp(-42{,}426/T)$ Since the units of w^2 are mg^2/cm^4, to convert to rate to kg^2/m^4, the constant should be multiplied by 10^{-4}. This was also the factor used in the conversion of the Urbanic-Heidrich constant for Zr in the MELCOR manual. The MELCOR manual refers to w as the mass of metal oxidized per unit area whereas the paper refers to w as the weight gain per unit area. Assuming that the oxidation product is FeO the ratio of weight gain to metal mass oxidized should be the ratio of atomic weights - about 16/56 or 0.29. In parabolic reaction rate this translates to a factor of about 10 which corresponds to MELCOR correlation for the stainless steel reaction rate. The following calculation is conducted with the rate constant as listed in the MELCOR manual. The parabolic rate constant for steam-H2O reaction is shown in the following figure.	

	Comment	Member	Response	Inserted into NUREG?
			 In the CSGTR analyses RCS failures typically occur when temperatures are substantially below 1750 K. The temperatures are rapidly rising limiting the time at high temperatures. Steel mass loss at a fixed temperature over the course of one day is shown for select temperatures below 1750 K in the following plot.	

	Comment	Member	Response	Inserted into NUREG?
			 The corresponding loss of steel thickness assuming a density of 8000 kg m^{-3} is shown in the following plot.	

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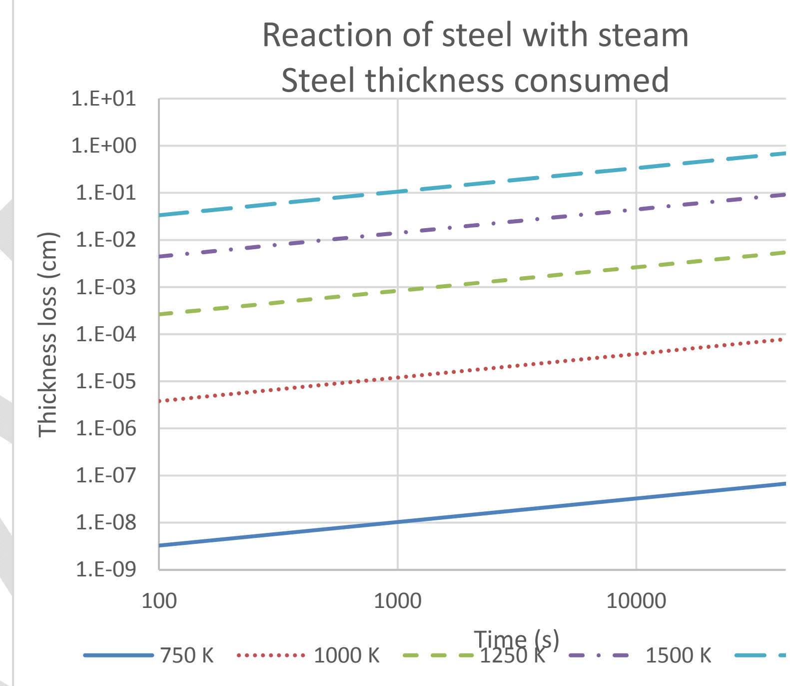

The steel-H2O model in the MELCOR reference manual** predicts no appreciable oxidation (~> 1mm) except for extended durations (~ 1 day) at near the melting point. It is assumed that, at these temperatures, failure by creep will occur long before oxidation is significant.
The stainless steel oxidation paper presented steam oxidation for different stainless steels and mild steel with the data points falling in the same general range. It is assumed that the relation is generally applicable to other steels.
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	Comment	Member	Response	Inserted into NUREG?
			The approach to RCS steel oxidation taken in this report is consistent with how phenomena is handled in severe accident analysis and previous consequential steam generator tube rupture analyses. Since the existing oxidation model does not predict appreciable oxidation in the absence of the hydrogen affect except at high temperatures, no attempt was made to consider the influence of hydrogen on oxidation, to identify low-melting-point iron oxides, to consider additional heat and hydrogen generation, and to consider the effects of stainless steel foaming including insulation for the oxidation of RCS components. If additional effects of foaming other effects are significant they should probably be considered first for the core where temperatures are hottest. More in depth analyses of steam oxidation of steel are available. Powers[***] reviewed the mechanisms governing steam oxidation of steel including the thermodynamics and kinetic principles of the various reactions. Of course we would like to hear of any information that indicates that relevant phenomena are not considered and that predictions made the above model may deviate substantially from what would actually occur in a hypothetical accident. No changes were made to the document. [*]J. T. Bittel, L. H. Sjodahl, and J. F. White, Oxidation of 304L Stainless Steel by Steam and by Air, Corrosion-NACE, Volume 25, No 1, January 1969.	

	Comment	Member	Response	Inserted into NUREG?
			** The "in the MELCOR reference manual" refers to both the use of the 9 rather than 8 as the exponent for the reaction rate and the interpretation of the parabolic rate referring to metal mass consumed rather than mass gain (oxide mass gained – metal mass consumed). *** D.A. Powers, "A Review of Steam Oxidation of Steels; The Forgotten Source of Hydrogen", in *Proceedings of the Workshop on the Impact of Hydrogen on Water Reactor Safety, Volume II of IV*, NUREG/CR-,2017 , SAND81-0661, Sandia National Laboratories, January 26-28, 1981	

	Comment	Member	Response	Inserted into NUREG?
8a 8b 8c	A critical review of the final document should be performed to summarize information already discussed in supporting documents and discuss information not discussed elsewhere. Section 5 doesn't discuss the volumetric flaw model? Models described in Section 5 are already described in NUREG/CR-6575 and NUREG/CR-6756 (This information should either be deleted or moved to an appendix). As noted above, there are no detailed reports for the CE CFD and MELCOR analyses like there were for the W plants. The report notes that the CFD analysis is an update of what was used in NUREG 1788, but the changes should be documented. Section 6 describes what is done to generate flaw distributions. It provides no technical basis for the process. The viewgraphs (see slides 65-77) have more information than this section. The letter report (A Letter Report on Flaw Database and C-SGTR Calculator Flaw Input, ISL, M.A. Azarm, et.al. December 2014 ML12202A302) on the technical basis is weak. The report should provide empirical results (in-service inspection based) and comparisons with proposed distributions. This was not in the letter report, although the viewgraphs have the basic plots (The final report should include the actual data points). The report should also discuss why the 'tail' ends at a probability of	Corradini, Ballinger Shack Rempe	8a 8b (volumetric flaw model) 8 b-c The CE CFD modeling is documented in reference 7 of Chapter 3. See response for comment 2. 8c. Appendix K is added to the report. It supplements the existing information with information from the viewgraphs mentioned.	N/A

	Comment	Member	Response	Inserted into NUREG?
8d	0.4 when there is only one flaw greater than 2.1 cm. The report should discuss differences in the equations for the flaw rates equations for 690 tubes that are shown in different places in the ISL report (note that this also occurs for Inconel 600 material). Also, the report should justify the basis for combining data from different steam generators in different plants. The distribution shown in Figure 8 of the ISL report doesn't seem justified if one considers information in Figure 8 of the ISL report. Comparison results between the calculator and MELCOR (or SCDAP) should be presented in the document.		8d The base TH failure calculations did not assume any flaws in the hottest tubes and a single flaw in the main tubes range. Standard creep rupture models and previously used parameters were used to assess failure. A one-to-one comparison is difficult and potentially may be misleading since both the model assumptions and the tools are very different.	
9	The final NUREG should discuss the planned overall process and connect to the guidance in the User Need letter. The final NUREG should also discuss applications of the research (applications that have already been performed or anticipated). The impact of FLEX on results should also be discussed.	Rempe	The draft NUREG is intended to be solely a technical document. Any ties into guidance and other applications are planned to be discussed in an accompanying internal NRC document that will follow the issuance of the NUREG. This subject has been already discussed with the NRR contact and understanding has been established at that level. I would also stay away from associating this report with FLEX since it was not studied and was not considered to be in the scope. A high level discussion may not have sufficient pedigree.	NO

10	Current assumptions related to loop seal clearing should be documented and justified. Different opinions can be found in NUREG/CR-6695 and various sections (e.g., Section 8.1, Section 3.7) of the draft NUREG.	Rempe	PRA assumptions listed in Section 7.1 and 8.1 in report The assumptions in section 3.7 do not contradict NUREG/CR-6695 but rather build upon them. The issue was not explored fully. Any difference is not expected to be a significant issue for the CE configuration analyzed in this work. With the reduction in the scope of the project a TH assessment of loop seal clearing for CE was not conducted. It was simply noted that the loop seals did not clear in the simulations that were run. One of the reasons that this TH analysis was not prioritized following the reduction in scope was that a high degree of containment bypass was concluded for CE even in the absence of loop clearing as a result of the high temperatures that the SG tubes are exposed to for the given design. Since the impact of loop seal clearing primarily results from hotter (near core temperature) gases reaching steam generator tubes, which already occurs in the CE design analyzed even for closed-loop-seal natural circulation, the additional impact of loop seal clearing on risk for CE is not expected to be significant. The initial intent to address loop seal clearing for this project was to test the different failure mode hypotheses and to determine whether apparent differences in loop seal behavior were inherent to designs, due to differences in codes, or differences in user choices. The plan was to perform a quick related "hand calculation" to ascertain what parameters would be important to both hypothesized failure modes and expected behavior, verify these relevant parameters in the input decks, and the run a series of simulations to test the extent to which the failure modes affected behavior. Only a	YES; Sentence added to the end of Section 3.6.1; Appendix J is added.

			general outline for approaching the problem was developed when initially planning the work. The text for loop seal clearing in the TH section reflects this initial outline. The direction to wrap up the work arrived before the loop-seal-clearing issue was revisited. The assumptions described in the TH analysis section for loop seal clearing, do not factor into results since neither the geometry nor system code models are changed. Rather these assumptions factor into how the results are interpreted and to help decide what to look for. The assumptions for loop seal clearing do not differ appreciably from those in NUREG/CR-6995. One additional factor is considered explicitly: the upper-vessel-to-downcomer leakage. The knowledge of the influence of this leakage is not new. In fact individuals involved with the SGAP and NUREG/CR-6695 indicated that core-to-downcomer bypass leakage had also been a considered during the development of the system code inputs. A choice of a small upper-core-to-downcomer leakage area for these Westinghouse analyses was found to result in loop seal clearing. What was planned to be explored further during this study is the expectation that the amount of seal leakage that results in loop seal clearing depends on both the assumed upper-vessel-to-downcomer leakage area and (perhaps to a lesser extent) RCS-to-containment heat transfer. Additional detail of the expected behavior follows: Upper loop seal water can be lost by 3 different ways:	

			1. Flow over to downcomer or out of RCP seal prior to bubble formation or if bubble shrinks or water level oscillations (bubble shrinking/not initially forming). In fact, in the absence of upper-vessel-to-downcomer leakage a bubble should not even form until either loop seal water reaches saturation or until SG side water level drops to the horizontal pipe section of the seal thereby allowing steam to bubble through. (seems to be a new consideration for this report) 2. Entrainment to RCP seal once (or if) steam flows through upper loop seal (lower loop seal must still be intact to maintain differential pressure. This is the primary mechanism for loop seal clearing described in NUREG/CR-6995. 3. Evaporation/flashing. This is an additional mechanism described in NUREG/CR-6995. To create sufficient differential pressure across the upper loop seal to cause steam to bubble through it (and thereby remove inventory by mechanism 2) other in-leakage to the upper horizontal part of the cold leg must not be significant. This means that 1. the lower loop seal (downcomer-core) must be intact, and 2. the upper-vessel-to-downcomer leakage area shouldn't be large relative to the RCP seal leakage area. If one of the other in-leakage pathways is open gas driven by the evaporation of any saturated water in the system would take that pathway rather than bubbling through the upper cold leg loop seal. Things to verify in the inputs should this issue be analyzed in	

			the future: How do the flow resistances across core and SGs in the code input compare with measurements? How do the Westinghouse and CE flow resistances compare, including the relative flow resistances between SGs and core? What is the maximum range of pressure drop and pressure drop difference achievable? – i.e. neglecting any liquid flashing to steam, what would the steam pressure drops across core and SG tubes be for an infinite volume of steam at cold legs if flow is limited by choked condition at the SRVs (parallel channel problem). How much does flashing affect behavior – from lower head and from loop seals? How do the elevations of the downcomer skirts in the inputs match expectations? How do these elevations and those of the loops differ between Westinghouse and CE designs and how would this be expected to affect clearing behavior? How much condensation is occurring? How does the magnitude compare to that of the Westinghouse calculations? Are differences due primarily to differing geometry surface-area or because of differing heat transfer coefficients? Do the Westinghouse and CE RVs have differing discharge rates?	
11	Additional tests should be performed to provide additional data for phenomena being modeled in CFD analyses. The report should acknowledge this, and it should be considered in the ACRS review of the research program.	Powers Banerjee	The CFD methods have been compared to test data at 1/7th scale. The benchmark indicated that peak temperatures entering the tubesheet, as well as tube bundle flow rates, are predicted adequately by CFD methods at 1/7th scale (ref. NUREG-1781). Further testing in this area has been considered (in a PSI proposed experiment) but is cost prohibitive and the full proposal has been dropped. The proposed testing at 1/7th scale (with improved instrumentation) would be expected to provide only marginal improvements over the existing data set.	NO
12	The draft NUREG states that some replacement steam generators for Westinghouse plants are more similar to the CE steam generators. When queried during the meeting, staff (Chris Boyd)	Rempe	It is removed.	Y; Removed from Section 3

	indicated that this was an original thought by the staff; but later found to be unsubstantiated. Hence, statements to this effect should be removed from the final NUREG.			page 3-7.
13	The draft NUREG states that states where C-SGTR can occur can be identified from existing Level 1 PRAs. PRA information doesn't consider conditions that could led to thermally-induced SGTR. In addition, assumed values for operator actions don't consider adverse human behavior that may occur in such events. Hence, conclusions about the importance of this event (based on existing PRAs can be misleading). In particular, this may be true for two-train CE plants.	Stetkar	It is true that further PRA modeling can be made to identify possible additional sequences of interest. The current scope is limited to the potential major sources of C-SGTR identified over the last 2 decades. No additional PRA modeling work is in the current scope. However, a detailed investigation in an ongoing PRA project has been made. Its conclusions are summarized in Appendix L. The PRA project in question is not publicly available.	Appendix L is added to the draft report.
14	Results in the draft NUREG list significant figures that are not supported by the analysis. Staff should go through the report and revise numbers to reflect accuracy supported by the analysis.	Ballinger	The significant figures are due to the small time steps involved in the finite-element analyses (numerical calculations) to ensure accuracy and precision of the algorithm. Our general principle is to leave the number of significant figures as is, except for reporting the final results. Otherwise, we get occasions where a reader thinks calculations using intermediate results are in error, since they may not match due to round-off. Accordingly, we have removed the significant figures in Table 4.4 (Sec 4.5) summarizing the failure times.	
15a 15b	The final report needs to be upgraded with respect to clarity and completeness. This is especially important because of the staff's intent to use it for subsequent applications. The final NUREG should better document limitations and uncertainties in this effort. - How do you consider the fact that uncertainties in detecting flaw sizes decreases as flaw size increases?	Corradini, Ballinger Stetkar Shack Rempe	15a 15b1 Small flaw sizes do not contribute to C-SGTR probability of the W-plant, therefore the impact of flaw detection probability (Probability of Detection –POD) is minimal. For the CE plant however, the small flaw sizes do contribute to C-SGTR and therefore the POD should be considered for	

	- Handling of uncertainties related to structural analyses is easier to justify than methods for treating uncertainties in the TH analyses. How do you obtain randomizing values for temperature and pressure? - How do you consider uncertainties in plant-to-plant measurements? - Appropriate treatment of uncertainty in operator actions.		sensitivity evaluation. However, the C-SGTR probability for the CE plant is large enough such that the effect of POD would be secondary. 15b2-Uncertainties in TH analysis/use of uncertainties in TH analyses It was not practical to combine the uncertainties resulting from TH with others due to the use of different tools. Nevertheless, as part of the initial deck generation, Sandia National Laboratories performed an uncertainty analysis involving multiple MELCOR simulations to characterize the variability in component failure timing resulting from expected variations in thermal hydraulic parameters. This work provided a distribution for TH-induced variability in absolute and relative failure timing that could be applied to other analyses. The documentation for the TH uncertainty analysis can be found in the Sandia Report (D. Louie, et al. "A MELCOR Model of the Calvert Cliffs Two-Loop Pressurized Water Reactor and Containment for the Steam Generator Tube Rupture Scenarios", Sandia National Laboratories, October 2012) on pages 92 to 102 and G1 to G9 (150 -158 in pdf). It is assumed that the resulting shape of distributions (width) resulting from TH variability, if not the absolute timing, can be applied to the calculation of failure using other tools. Uncertainties were not evaluated further due to the reduction in scope of the project. There had been an intent to further evaluate uncertainties to distinguish between parameters that shift failure times of both tubes and other RCS components equally (and therefore do not substantially affect releases) and those that shift the failure times of tubes relative to that	

			of other components (and could therefore potentially affect releases to the environment). No additional changes were made to the TH section of the report. 15b3-Uncertainties in plant-to-plant measurements: Majority of parameters used in this study are plant specific with the exception of the flaw data which was generic; across industry. Theoretically, flaw samples can be randomly generated from the basic distribution to address the variability across plants, although it is not currently addressed within the report. 15b4-Uncertainties in operator actions Uncertainties associated with operator actions within EOP can be handled similar to the current state of PRA practices. Uncertainties associated with operator actions within SAMG or FLEX is not well understood. These uncertainties are not currently treated within this study.	