April 21, 2004

MEMORANDUM TO: ACRS Members

FROM: Bhagwat Jain, Senior Staff Engineer /***RA***/
Technical Support Staff

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS JOINT MATERIALS & METALLURGY AND THERMAL-HYDRAULIC SUBCOMMITTEES MEETING, FEBRUARY 3-4, 2004, ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on March 24, 2004, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc via e-mail:
ACRS Members
J. Larkins
R. Savio
H. Larson
S. Duraiswamy
ACRS Staff Engineers

MEMORANDUM TO: Bhagwat P. Jain, Senior Staff Engineer
Technical Support Staff

FROM: F. Peter Ford, Co-Chairman
Graham Wallis, Co-Chairman
Joint Subcommittee on Materials and Metallurgy and Thermal-Hydraulic Phenomena

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS JOINT MATERIALS & METALLURGY AND THERMAL-HYDRAULIC SUBCOMMITTEES MEETING, FEBRUARY 3-4, 2004, ROCKVILLE, MARYLAND

We hereby certify that, to the best of our knowledge and belief, the Minutes of the subject meeting issued March 24, 2004, are an accurate record of the proceedings for that meeting.

/Original signed by/ March 26, 2004
F. Peter Ford, Co-Chairman Date

/Original signed by/ April 1, 2004
Graham Wallis, Co-Chairman Date

Certified By
F. Peter Ford
Graham Wallis

Issued: 3/24/2004
Certified 3/26/2004
4/1/2004

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
JOINT MATERIALS & METALLURGY AND
THERMAL-HYDRAULIC PHENOMENA SUBCOMMITTEE MEETING
STEAM GENERATOR ACTION PLAN (SGAP) -
DIFFERING PROFESSIONAL OPINION (DPO) RELATED ITEMS
MARCH 24, 2004
ROCKVILLE, MARYLAND

Introduction

The ACRS joint Subcommittees on Materials and Metallurgy and Thermal-Hydraulic Phenomena held a meeting on February 3-4, 2004, with representatives of the staff and its contractors. The purpose of this meeting was to review the staff's resolution of certain steam generator action plan (SGAP) items which are associated with the differing professional opinion (DPO) on steam generator tube integrity, as well as the status of resolution of remaining items. Mr. Bhagwat Jain was the cognizant ACRS staff engineer and Designated Federal Official (DFO) for this meeting. The meeting was convened at 8:30 a.m. on February 3, 2004 and was adjourned at 5:30 p.m. on February 4, 2004.

Attendees

ACRS Members/Staff	NRC Staff	Contractors and Industry
F. Peter Ford(Co-Chairman)	Maitri Banerjee (NRR)	Saurin Majumdar (ANL)
Graham B. Wallis (Co-Chairman)	William Bateman (NRR)	Paul Amjco (SAIC)
Stephen L. Rosen (Member)	Chris Boyd (RES)	David Bradley (SAIC)
John Sieber (Member)	Jim Davis (RES)	Dave Kunsman (SNL)
Mario V. Bonaca (Member)	William Krotiuk (RES)	Robert Beatin (ISL)
Thomas S. Kress (Member)	Roy Woods (RES)	Don Fletcher (ISL)
Victor R. Ransom (Member)	Michelle Hart (NRR)	William Shack (ANL)
Bhagwat Jain (DFO)	Joel Page (RES)	Jim Riley (NEI)
	Joe Muscara (RES)	
	Allen Hieser (RES)	
	Louise Lund (NRR)	
	Ken Karwoski (NRR)	
	Steve Long (NRR)	

ACRS Members/Staff	NRC Staff	Contractors and Industry
	Walton Jenson (NRR)	
	Steven Arndt (RES)	

A complete list of all attendees is attached to the Office copy of these Minutes.

The presentation slides and handouts used during the meeting are attached to the Office Copy of these Minutes. The presentations to the Subcommittee are summarized below. No request from the public was received to make an oral presentation.

Background

In a memorandum of July 20, 2000, the Executive Director of Operations (EDO) requested that the ACRS review a differing professional opinion (DPO) on various issues associated with the rupture of steam generator (SG) tubes during a main steam line break that could result in a release of radioactivity with containment bypass. In February 2001 the ACRS offered their opinions on the DPO in NUREG-1740, "Voltage-Based Alternative Repair Criteria". The ACRS endorsed a condition monitoring program in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes" and concluded that although the criteria could provide adequate protection of public health and safety, there were certain DPO contentions that merited further technical attention. Consequently, the SGAP was modified by the staff to address the ACRS concerns. The ACRS in a report of October 18, 2001 to the chairman noted that the Committee's recommendations to resolve the DPO concerning SG integrity issues had been incorporated into staff's existing SGAP. The SGAP has the objective of differentiating between the probabilities of radioactivity release during design base accidents, such as a main steam line break (MSLB) and those during a severe accident initiated by e.g., a station blackout (SBO). Of particular concern is the probability of accident sequences that might lead to failure of the steam generator tubes and potential release of radioactivity with containment bypass.

The Committee's concerns are addressed and identified in the SGAP items 3.1 through 3.11. A discussion on each of these items is summarized in these Minutes.

Opening Remarks (P. Ford and G. Wallis, ACRS)

Co-Chairman Ford convened the meeting. He commented that he would like to hear the staff's criteria for closing out certain SGAP items related to DPO, and how the staff has met the overall objective of assessing the risk associated with various severe accident sequences . Co-Chairman Wallis expressed the similar sentiment and commented that it is not clear to him what the inputs and outputs were for various tasks, and how they relate and fit into modeling an entire accident sequence.

Staff Introduction (Joe Muscara, RES)

Mr. Muscara stated that their presentation to the joint Subcommittee was intended to provide a progress report on various SGAP issues and to address the concerns raised in the DPO associated with non-destructive examination (NDE), steam generator tube integrity under main steamline break (MSLB), and iodine spiking. In response to Member Bonaca's question regarding the scope, Mr. Muscara stated that the staff will address only SGAP items 3.1 through 3.9. Progress on Items 3.10 and 3.11 will be addressed in future briefings.

Mr. Muscara stated that the staff's presentation would emphasize the technical work that led to completion of certain SGAP tasks, subtasks and milestones. Although some SGAP milestones have been closed, work in some of the same areas is continuing based on lessons learned and refinements. He committed to periodically update the SGAP. He mentioned that considerable research has been completed in the area of SG tube integrity inservice inspection and nondestructive evaluation, thermal- hydraulics, primary system component response during severe accidents, probabilistic risk assessment (PRA), and iodine spiking.

In response to questions from Co-Chairmen Ford and Wallis, and Member Kress, Mr. Muscara stated that the SGAP consists of a number of building blocks necessary to meet the overall objective of the SGAP. The staff is closing out some of the building blocks. If additional work or refinement is needed, then new tasks will be added as appropriate.

Mr. Muscara stated that the staff has a program underway to integrate all steam generator research activities in order to address the potential for containment bypass during a design-basis accident or a severe accident. For example, the PRA may identify likely combination of events that lead to temperature and pressure time histories defined by thermal-hydraulics. These thermal-hydraulics inputs are then used to evaluate the SG tube integrity by making use of flaw distribution, probability of flaw detection, and SG integrity models. The integrated program will be completed by end of fiscal year 2005. Co-Chairman Ford questioned the staff's schedule considering the incomplete status of several building blocks, such as PRA and thermal-hydraulics and crack modeling inputs. Mr. Muscara said the staff is comfortable with the schedule.

<u>SGAP Item 3.1 'Investigate the effects of depressurization during a main steamline break on steam generator tube integrity,' Presentation (S. Majumdar, ANL and W. Kroutiuk, RES)</u>

This SGAP item relates to the DPO concern that the blowdown forces and movement of the SG tube support plate (TSP) caused by the SG depressurization following a MSLB could cause cracks to form, to grow and to unplug, thereby leading to much higher primary-to-secondary side leakage than has been considered currently by the staff. The SGs of interest in the DPO were those fabricated using Alloy 600 tubes with carbon steel TSPs, which experience tube locking at TSP due to carbon steel corrosion product depositing in the crevices between the SG tubes and the TSPs.

The staff presented the results of its tests and finite-element analyses which were performed to evaluate the stability of cracks in the SG tubes subjected to stresses due to flexing of TSPs. Flexing of TSPs during a MSLB transmits both axial and bending loads to the SG tubes that are locked at the carbon steel TSPs. The staff concluded that the bending stresses in the SG tubes are small compared to the axial stresses, and that the effects of bending loads on the leakage from flawed SG tubes is also small.

The staff also calculated hydraulic loadings on the internal components of a SG following a MSLB for use in the structural evaluation of the stability of cracks subjected to such loads. The staff used the TRAC-M three dimensional model of the SG for these calculations. The staff stated that it had verified TRAC-M code's ability to predict acoustically dominated thermal-hydraulic transients such as those caused by a MSLB by comparing TRAC-M predictions against the Edwards Pipe Blowdown Experiment and the Loss-of-Fluid Test (LOFT) Semi scale Blowdown Test.

The staff presented the results of its study of locked SG tubes. In response to a question from Member Bonaca regarding the maximum displacement of TSPs if no tubes are locked, the staff stated that the TSP would displace by almost 2 to 3 inches. The staff, using the TRAC-M code, calculated tensile stresses in the SG tubes locked at the TSP junctions for varying numbers of locked SG tubes. The staff concluded that if only one or two SG tubes are assumed to be locked at the TSPs, the calculated axial stresses exceed the ultimate tensile strength of the SG tubes. Therefore, the allowable circumferential crack length in these SG tubes could be severely limited. By assuming an increasing number of SG tubes locked at the TSP, the staff concluded that the tensile stresses on individual SG tubes decrease significantly, and the allowable circumferential crack length thereby increases. The staff's study also showed that even if a few percent of the SG tubes are locked at the TSP, the dynamic loads associated with a MSLB will have little impact on the integrity of the SG tubes unless extensive circumferential cracking is present.

In support of its argument regarding locking of SG tubes, the staff stated that results of an extensive study on the pullout forces at Dampierre nuclear plant strongly support the conclusion that SGs with drilled hole carbon steel TSPs will have a sufficient number of SG tubes locked at the TSP and that the dynamic loads will be of little concern. The staff did concede that a more extensive review of industry data, especially from US PWR plants, on the expected number of locked SG tubes in degraded SGs and the SG tube pullout data would strengthen this conclusion. Since the staff's conclusions regarding MSLB-induced additional leakage depend heavily on confirmation of these data, the Subcommittee felt that the staff should review US industry SG tube pullout data and the extent of tube locking at tube support plates (TSPs) in degraded Sgs.

SGAP Item 3.2 'Complete investigation of jet penetration of adjacent tubes' (Presentation (J. Muscara, RES)

Mr. Muscara stated that the staff has closed out this item. The Committee in its report of October 18, 2001, had concluded that, based on the results of the research on the effects of jet impingement on adjacent tubes, the probability of damage progression is low enough that it can be neglected in the accident analyses. The Committee agreed with the staff on the resolution of this item.

SGAP Item 3.3 'Develop experimental information on source term attenuation on the secondary side of steam generators (ARTIST Program),' Presentation (C. Boyd, RES, David Kupperman and William Shack, ANL)

The staff stated that the objective of the Aerosol Trapping in Steam Generator (ARTIST) program is to provide data on the behavior of simulant aerosol in the secondary side of a steam generator. This program is being conducted at the Paul Sherrer Institut (PSI) in Switzerland. The staff expects the data to be available by 2007.The staff plans to incorporate the data into the MELCOR analysis.

This item was provided to the Subcommittee for information only.

SGAP Item 3.4 'Develop a better understanding of steam generator tube behavior under severe accident conditions,' Presentation (C. Boyd, RES, Don Fletcher, ISL, and S. Majumdar, ANL)

An additional concern raised by the DPO is that severe accident sequences, in which the primary system remains pressurized, are more likely to evolve into SG tube rupture accidents than the staff predicts. Important examples of such severe accidents are SBO, small break accidents, and anticipated transients without scram (ATWS). These severe accidents are of interest because they impose high thermal and pressure loads on the SG tubes.

The staff presented the results of thermal-hydraulic analyses of the steam/water flow processes in the upper plenum, hot leg, SG plenum and SG tubes that would exist during a severe accident scenario. The staff demonstrated an impressive use of computational fluid dynamics (CFD) to predict the mixing fraction in the inlet plenum of the steam generator following an accident, and the resultant increase in temperatures of, for example SG tubes, surge lines, etc. Both one-dimensional system code calculations and CFD simulations were used. The main outputs of the thermal-hydraulic analyses are the component temperatures that result from the convective heating due to the flow of superheated steam from the core. As the fluid temperatures increase in the core, hot leg and SG tubes, component failures may be induced in these components. The rate of delivery of core decay heat energy from the core to these components determines the heating rate and the rate of temperature increase. The decay heat is evolved in the core and produces superheated steam which rises into the upper plenum due to buoyancy as a result of the decreased density of the heated steam. The buoyant hot steam then flows along the top portion of the hot leg to the steam generator inlet plenum. Cooled steam from the SG then returns by countercurrent flow along the bottom of the hot leg to the reactor vessel. The net energy transport to the steam generator, which produces heating of the hot leg and steam generator components, depends on this complex countercurrent flow process.

The staff stated that its thermal-hydraulic analyses thus far have used a one-dimensional system code calculation, tuned to agree with the 1/7th scale SG test data. These analyses have then been used to determine the flow to the SG and to set the boundary conditions for a more mechanistic CFD simulation of the hot leg to SG flow process. The CFD results are very insightful and demonstrate the power of these methods. However, Co-Chairman Wallis commented that the quantitative value of this work is limited by the need to empirically set the boundary conditions at the hot leg entrance. This CFD work should be extended to include sufficient parts of the upper plenum and core flow process to permit mechanistic calculation of the hot leg entrance conditions. Furthermore, this would permit the mixing process surrounding the cold "plume" emerging from the hot leg and descending into the reactor pressure vessel to be modeled in much the same way as the hot plume was modeled by the CFD calculation in the SG inlet plenum.

Co-Chairman Wallis also commented that it is apparent that the amount of heat which goes to the steam generator depends on all of the convection and heat transfer processes between the core and the steam generator. It cannot be assumed as an input variable based on the 1/7th scale test, as appears to be the case in the SCDAP/RELAP analysis. The 1/7th scale tests contain several features that may be atypical of the full scale plants, such as the method of cooling the steam generator tubes.

Several Subcommittee Members expressed concern that the uncertainty in the calculated thermal response of the primary system components under severe accident conditions may be too great to use such results to discern whether the pressurizer surge-line or the SG tubing fails first. Member Ransom was of the opinion that the primary system conditions, (i.e. high temperature steam under natural circulation), involves phenomena beyond the prediction capability of a one-dimensional thermal-hydraulics system code. In addition, the uncertainty in the associated heat transfer correlations, friction correlations, and material properties under these conditions are also large. Even when the calculation is augmented with CFD modeling, as has been done, the uncertainty of the predicted outcome may still be very large.

SGAP Item 3.5 'Develop improved methods of assessing risk associated with steam generator tubes under accident conditions,' Presentation (David Bradley, SAIC)

The staff described its planned approach to PRA and current status of their work under this SGAP item. The staff conceded at the outset that they have just started on this task and therefore, there aren't any results to report to the Subcommittee. The staff stated that the tasks scope is limited to severe accident induced containment bypass scenarios that are driven by SG tube failure. The staff mentioned that it has published a draft methodology report in June 2003, and is now going to undertake an application of methodology to a Westinghouse four-loop plant. In response to a question from Co-Chairman Wallis, the staff responded that the issues are more complicated than they had originally thought and they anticipate the methodology to change drastically after its application to a sample plant. The methodology relies on continuous interaction with other elements of the program i.e., tube integrity analysis, thermal-hydraulic analysis, and analysis of other reactor coolant system (RCS) components to determine when those failures might happen before tubes would fail.

The staff provided an overview of the PRA approach that included sequence definition, binning and quantification of those sequences to determine accident sequence frequencies and also to determine the probability that the tubes would fail before failure of other RCS components.

The staff stated that it plans to use the Comanche Peak nuclear plant PRA model that it judged to be the most capable of the available PRA models. The staff reported that they are reviewing Comanche Peak PRA model against the ASME standard for PRA model and will enhance it as appropriate. In this study, thermal-hydraulic results based on the Zion plant will be used for RCS components failure analyses. Co-Chairman Ford raised a concern regarding this mismatch as this may render the application of PRA analysis to a specific plant invalid. The staff responded that their goal is only to demonstrate the application of methodology and not obtain any plant specific results.

At the conclusion of the presentation, the Subcommittee felt that at this stage it can not determine the adequacy of the results of the severe accident scenario under study until staff makes further progress in the development of a probabilistic risk assessment methodology involving the propagation of uncertainties during the accident progression.

SGAP Item 3.6 'Assess the technical basis for improving the probability of crack detection in steam generator tubes,' Presentation (David Kupperman and William Shack, ANL)

The staff addressed the issue of probability of detection (POD) analysis to determine if any improvements can be made over the current use of a constant POD for the flaws in steam generator tubes as POD depends on flaw size.

The staff stated that an eddy current NDE analysis round robin was performed using industry qualified teams and flaws in the ANL/NRC steam generator mock-up that had signals similar to those observed under field condition. The staff described the SG mock-up that consisted of 400 tubes. Each SG tube contains nine test sections for a total of 3600 test sections. Over 300 of these test sections have flaws in outer diameter (OD) and inner diameter (ID) of the tubes. Test sections having outer diameter (OD) flaw were evaluated with the dye penetrant

The objective of the round robin effort was to evaluate and quantify the inspection reliability of the current methods being used for inservice inspection for the flaws. The methods were validated by employing both laboratory and field generated flaws. The round robin included 11 different teams analyzing exactly the same data. The resulting POD curve is a function of flaw depth, voltage, location, and m_p, the stress magnification factor in the ligament for 7/8" diameter alloy 600 tubing.

The staff reported that Argonne National Laboratory has developed an eddy current multiparameter algorithm which provides a significant improvement in the technology required to detect and size SG tube flaws over the previous capabilities. The staff argued that although such a development may be currently limited commercially, it provides a valuable standard to the staff against which current commercially available eddy current analyses can be evaluated.

In summary, the staff presented a detailed analysis of the POD issue. In its analysis, the staff demonstrated that the continued use of a POD value of 0.6 was conservative for eddy current voltages in the range 1-2 volts where the observed POD value approaches 0.9. Several Members (Kress, Ford, Wallis) expressed that this may very well be the case, but its absolute validity in terms of the degree of conservatism will be a function of inputs from tasks associated with the prediction of the range of flaw depths (where the POD may well be less than 0.6), and the range of axial loads on the SG tube during an MSLB due to ranges in the extent of tube/tube sheet locking.

The Subcommittee concluded that the ultimate measure of adequacy of POD will depend on the extent of the uncertainties in the analyses and their propagation through the accident sequences. The Subcommittee will evaluate the adequacy of this individual precursor task after the staff progresses further in the development of the probabilistic risk assessment methodology.

SGAP Item 3.7 'Assess the need for better leakage correlations as a function of voltage for 7/8" steam generator tubes,' Presentation (David Kupperman, ANL, William Shack, ANL, L. Lund, NRR, and K. Karowski, NRR)

The staff stated that the database comparing eddy current voltage and leak rate for 7/8" diameter SG tubes contained 31 data points and exhibited high scatter. In its effort to assess the effect of additional new data on the correlation, the staff determined that additional data from Beaver Valley in 2001 and from Sequoyah in 2002 made the leakage-voltage correlation worse; however, addition of new data from Diablo Canyon in 2003 made the correlation better. The staff in its presentation concluded that the leakage methodology is acceptable because GL95-05 specifies necessary actions in the leak rate calculations when the correlation is weak and specifies how to account for the uncertainty in the correlation. Furthermore, staff stated that the overall methodology for determining the amount of leakage and assessing its consequences is conservative. The staff committed to monitor the database while plants are implementing alternate repair criteria. Co-Chairman Wallis asked the staff to clarify his understanding that if a probe indicates the voltage larger than a certain amount, some action will be taken. The staff stated that the tube in such cases will be plugged.

At the conclusion of the staff's presentation, the Subcommittee was satisfied and agreed with the staff's position that it is unlikely that an improved correlation between eddy current voltage and leakage for 7/8" diameter SG tubes will be achieved. Although there is a qualitative understanding of the reasons for the unacceptable scatter in the relationship, it is apparent that the accumulation of further unqualified data will not improve the leakage correlation. The Subcommittee concurred with the staff that this will not increase the risk, since, the choice of a 2 volt limit for the 7/8" diameter SG tubes provides sufficient conservatism. The existing correlation of leakage with eddy current voltage for 7/8" diameter SG tubes therefore is adequate.

SGAP Item 3.8 'Monitor the predictions of flaw growth for systematic deviations from expectations,' Presentation (David Kupperman and William Shack, ANL)

The staff stated that they do not postulate individual flaw (crack) growth rates. Rather, there is a distribution of growth rates that is expected to be observed based on the previous cycle. The flaws in the SG tubes are allowed to remain inservice under the voltage based alternate repair criteria at the beginning of the cycle, and then at the end of cycle a comparison is made to see how well the flaw growth is predicted by the methodology. Currently nine plants are authorized to implement alternate repair criteria outlined in generic letter GL95-05. The staff agreed with the Subcommittee that flaw growth is not linear and individual flaws grow slowly until they coalesce with their neighbors.

The staff stated that the main objective of this task is to assess how well the methodology is able to predict the end of cycle conditions. The probability of prior cycle detection (POPCD) accounts for probability of detection and the potential for new indications to develop during the course of a cycle. In response to Co-Chairman Wallis's question regarding the accuracy of methodology for predicting flaw growth, the staff clarified that the actual probabilities of burst and leakage are compared with the projected probability of burst and the probability of leakage. There was considerable confusion during the discussion as to what is being predicted. The staff clarified that the voltage is being predicted at the beginning and at the end of the cycle. Co-

Chairman Wallis asked the staff if it was monitoring for change in voltage, and the staff concurred that indeed it is monitoring voltage growth. But the staff asserted that its acceptance criteria are based on the probability of burst and the probability of leakage.

Member Sieber stated that at a given point in time when you recognize a flaw, the question is: should you have detected it before? And if you didn't, you can't measure the crack growth rate. If you can't measure the crack growth rate, then you can't tell what condition of the flaw is going to be at the end of the cycle.

The Subcommittee stated that it is not in the position to decide on the adequacy of staff's response on this issue until the staff makes progress in the development of the crack initiation and growth model covered in the SGAP item 3.10.

SGAP Item 3.9 'Assess the need for a more technically defensible treatment of radionuclide release to be used in safety analyses of design-basis events,' Presentation (Michelle Hart, NRR)

The staff stated that they have looked for but could not find any additional data on the iodine spiking phenomenon for the SG tube rupture and MSLB type events. The staff further stated that their reexamination of the unadjusted raw data does not indicate a clear dependency of spiking rate on pre-incident coolant iodine activity concentration. Member Kress challenged the staff's conclusion and stated that the Ad Hoc Committee looked at the same database and found clear dependency. The staff defended its position and maintained that based on existing data its current modeling regime is conservative. Co-Chairman Wallis expressed his concern that the staff 'thinks' that the current modeling regime is conservative but has not produced any analysis or evidence to support its thinking. The staff needs to develop a mechanistic understanding of the iodine spiking issue.

At the conclusion of this presentation, the Subcommittee concluded that the staff has elected to not accept its recommendation that they should develop a mechanistic understanding of the iodine spiking issue. Instead, they have used an equation the Ad Hoc Subcommittee had developed to show a correlation between the iodine spiking factor and the coolant iodine concentration together with an assumed dependence on pressure difference. The Subcommittee felt that bounding accident analyses such as these are inconsistent with the current move toward more realistic safety analyses and principles of good regulation, and encouraged the staff to take advantage of iodine studies available in the literature to develop a mechanistic understanding of the phenomenon.

The Subcommittee encouraged the staff to reexamine the treatment of spiking factors for radioactive iodine release to ensure that adequate conservatism is maintained in the dose calculations.

SGAP Item 3.10 'Develop a better mechanistic understanding of tube cracking processes'

Progress on this issue will be addressed in future briefings.

SGAP Item 3.11 'Resolve Generic Safety Issue 163, "Multiple Steam Generator Tube Leakage'

Progress on this issue will be addressed in future briefings.

Subcommittee Comments, Concerns and Recommendations

Overall, the joint Subcommittee Members were satisfied that the approaches that are being used to resolve the majority of the technical issues outlined in NUREG-1740 are appropriate. However, Members felt that until staff makes further progress in the development of a probabilistic risk assessment methodology involving the propagation of uncertainties during the accident progression, it is difficult at this stage to determine the adequacy of the results of the severe accident scenario under study in SGAP item 3.4. The staff's treatment of spiking factors for radioactive iodine release covered in SGAP item 3.9 needs to be reexamined to ensure that adequate conservatism is maintained in the dose calculations. The Subcommittee wanted the staff to include uncertainties in the thermal-hydraulics analyses of the severe accident condition to assess the relative probability of radioactivity release with and without containment bypass. The Subcommittee Co-Chairmen stated that they will evaluate the adequacy of SGAP items 3.6 'Assess the technical basis for improving the probability of crack detection in steam generator tubes,' and SGAP item 3.8, 'Monitor the predictions of flaw growth for systematic deviations from expectations,' after the staff has made sufficient progress in the development of the crack initiation and propagation model in SGAP item 3.10. The Subcommittee believes that existing correlation of leakage with eddy current voltage for 7/8" diameter SG tube is adequate and that SGAP item 3.7 is appropriately closed.

Staff Commitments

The staff will brief the full Committee on February 5, 2004. The staff will discuss its progress in resolving the issues identified by the ACRS in NUREG-1740 with the Subcommittee (on Materials and Metallurgy and on Thermal Hydraulic Phenomena) on a periodic basis.

Subcommittee Decisions and Follow-up Actions

The ACRS Subcommittees on Materials and Metallurgy and on Thermal Hydraulic Phenomena decided to hold meetings, as needed, to discuss the progress made by the staff in resolving the issues identified by the ACRS in NUREG-1740.

Background Material Provided to the Subcommittee Prior to this meeting

1. Subcommittee status report
2. Proposed Schedule
3. The Committee also reviewed the following documents that the staff provided to ACRS in support of the February 2004 Briefings:

SGAP Item No.	Document Title/Date	Public/Non-Public
3.1.a and b	Memo to Mayfield from Eltawila, 12/30/02, “Calculation of Steam Generator Tube Support Plate and Tube Loads Following a MSLB or FWLB Using TRAC-M;”	Non-Public
	Att: RES Report SMSAB-02-05, 9/02	Non-Public
3.1.d thru h	Memo from Mayfield to Strosnider, 12/26/02, “Closure of Steam Generator Action Plan Items 3.1d) to h);”	Non-Public
	Att: NUREG/CR-XXXX, **“**Sensitivity Studies of Failure of Steam Generator Tubes During Main Steam Line Break and Other Secondary Side Depressurization Events”	Non-Public (Proprietary)
3.1.i	Memo from Mayfield to Strosnider, 7/14/03, “Closure of Steam Generator Action Plan Item 3.1i);	Non-Public
	Att: Technical Letter Report: Tests and Analysis Of Failure of Degraded Tubes Under Internal Pressure and Bending Loading”	Non-Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.2	Memo from Mayfield to Strosnider, 7/9/02, "Closure of Steam Generator Action Plan Items 3.2 and 3.6;"	Public
	Att: NUREG/CR-6756, "Analysis of Potential for Jet-Impingement Erosion from Leaking SteamGenerator Tubes during Severe Accidents;"	Public
	Att: NUREG/CR-6774, "Validation of Failure And Leak-Rate Correlation for Stress Corrosion Cracks in Steam Generator Tubes"	Public
3.4.a	Memo from King to Zimmerman, 9/28/01, "Completion of Subtask Milestone in Steam Generator Action Plan;"	Public
	Attached report: ISL-NSAD-NRC-01-004	Non-Public
3.4.c	ISL-NSAD-TR-02-03, "Tube-to-Tube Temperature Variations During the Station Blackout Event" (Draft), 8/02	Non-Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.4.e.1	Memo from Rosenthal to Barrett, Wermiel, Banerjee, ACompletion of Preliminary Milestone from Steam Generator Action Plan, 8/31/01	Non-Public
	Att: NUREG-1781, "CFD Analysis of 1/7th Scale Steam Generator Inlet Plenum Mixing During a PWR Degraded Core Accident," 10/03	Public
3.4.e.2	Draft Report, "CFD Prediction of Full-Scale Steam Generator Inlet Plenum Mixing for the Evaluation of Scale Effect," 3/02	Non-Public
3.4.e.3	Memo from Eltawila to Holahan, "Preliminary Results from SGAP Item 3.4.e.3 Related to the CDF Evaluation of Inlet Plenum Mixing," 2/25/03	Non-Public
3.5.a	Memo from Cunningham to Chokshi and Rosenthal, **"**Transmission of a Proposed Framework for Analysis of Severe Accident Induced Steam Generator Tube Ruptures,**"** 4/1/02	Non-Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.5.b	Memo from Newberry to Strosnider, "Closure of Steam Generator Action Plan Items 3.5(b) and 3.5(c), "Severe Accident Induced-Steam Generator Tube Rupture (SAI-SGTR) Methodology Report," 6/30/03 (Note: Contrary to this memo item 3.5.c is not yet closed)	Public
	Draft Report "Methodology for Assessing Severe Accident-Induced Steam Generator Tube Rupture," 6/03	Public
3.6	Memo from Mayfield to Strosnider, 7/9/02 "Closure of Steam Generator Action Plan Items 3.2 and 3.6;"	Public
	Att: NUREG/CR-6785, "Evaluation of Eddy Current Reliability from Steam Generator Mock-Up Round-Robin," 9/02	Public
3.7	Memo from Barrett to Sheron and Borchardt, "Steam Generator Action Plan - Completion of Item Number 3.7 (TAC No. MB7216)," 4/25/03;	Public
	Letter from Bateman to Marion (NEI), "Exclusion of French Data from the Steam Generator Degradation Specific Management Database," 10/8/02	Public

SGAP Item No.	Document Title/Date	Public/Non-Public
3.8	Memo from Strosnider to Sheron and Borchardt, "Steam Generator Action Plan - Completion of Item Number 3.8 (TAC No. MB0258)," 1/3/02	Public
3.9	Draft Writeup, "ASGAP Item 3.9 - Iodine Spiking	Non-Public

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Note: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers), 1323 Rhode Island Avenue, NW, Washington, DC 20005 (202) 234-4433

<u>Presentation Slides and Handouts Provided during the Subcommittee meeting</u>