UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

November 13, 2009

MEMORANDUM TO: ACRS Members

FROM: Christopher L. Brown, Senior Staff Engineer ***/RA/***
Reactor Safety Branch A, ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE MEETING, SEPTEMBER 24-25, 2009 - ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on November 6, 2009, as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc w/o Attachment: E. Hackett
C. Santos

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 – 0001

November 13, 2009

MEMORANDUM TO: Christopher L. Brown, Senior Staff Engineer
Reactor Safety Branch A, ACRS

FROM: Dana Powers, Chairman
Materials, Metallurgy, and Reactor Fuels Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE MEETING, SEPTEMBER 24-25, 2009 - ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on September 24-25, 2009, are an accurate record of the proceedings for that meeting.

/RA/ 11/06/09

Dana Powers, Chairman, Date
Materials, Metallurgy, and Reactor Fuels
Subcommittee

Certified by: D. Powers   Issued: 11/13/09
Certified: 11/06/2009

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF ACRS MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE MEETING
SEPTEMBER 24-25, 2009
ROCKVILLE, MARYLAND

The ACRS Materials, Metallurgy, and Reactor Fuels Subcommittee held a meeting on September 24-25, 2009, in the Commissioner's Hearing Room, 11545 Rockville Pike, Rockville, MD. The purpose of this meeting was to review closure of the Steam Generator Action Plan. Christopher Brown was the designated Federal Official for this meeting. The Subcommittee received no written statements or requests for time to make oral statements from the public. The Subcommittee Chairman convened the meeting on September 24, 2009 at 8:30 a.m. and adjourned on September 25, 2009 at 12.00 p.m.

ATTENDEES:

ACRS Members

D. Powers, Chairman	W. Shack
D. Bley	J. Stetkar
J. Sieber	S. Banerjee
M. Bonaca	S. Armijo
O. Maynard	

ACRS Staff

C. L. Brown, Designated Federal Official

NRC Staff

K. Karwoski, NRR	T. McGinty, NRR
R. Hardies, NRR	J. Hixon, RES
E. Murphy, NRR	G. Carpenter, RES
D. Beaulieu, NRR	T. Lupold, RES
K. Armstrong, RES	C. Boyd, RES
S. Sancaktar, RES	B. Palla
K. Coyne	E. Fuller
D. Fletcher	

Other members of the public attended this meeting. A complete list of attendees is in the ACRS Office File and is available upon request. The presentation slides and handouts used during the meeting are attached to the official copy of the meeting transcript.

Opening Remarks and Objectives:

Dr. D. Powers, Chairman of the ACRS Materials, Metallurgy, and Reactor Fuels Subcommittee, convened the meeting at 8:30 a.m. The purpose of this meeting is to discuss the status of activities associated with the resolution of the remaining items in the Steam Generator Action Plan. The presenters include representatives from the NRC's Office of Nuclear Reactor Regulation (NRR) and the Office of Nuclear Regulatory Research (RES). The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee. The rules for participation in the meeting are announced as part of the notice of the meeting previously published in the *Federal Register*. The Committee did not receive written statements or requests for time to make oral statements from members of the public.

Overview of the Steam Generator Action Plan

For several years, the NRC's Steam Generator Action Plan focused research on a variety of issues in connection with the steam generator tube failures. The staff has completed its work to close all Steam Generator Action Plan items. The purpose of the 2-day meeting is for the ACRS Subcommittee to review the remaining task items to be closed in the Steam Generator. The desired outcome following the subcommittee meeting and the Full-Committee meeting is that the ACRS will issue a letter which finds acceptable the staff's closeout of each item in the Steam Generator Action Plan that ACRS has not previously reviewed and closed. The task items to be discussed are 3.1.k, 3.4, 3.5, 3.10, 3.11, and 3.12.

Essentially, all of the task items are directly related to the work to define the risk associated with severe accident induced tube ruptures leading to containment bypass. The work involved the following technical areas of research: thermal-hydraulics, steam generator tube material failures; reactor coolant system material failures; component behavior studies, and PRA. The thermal-hydraulics work takes the PRA sequence being evaluated and determines the fluid temperatures and pressures as a function of time. These conditions are used as inputs to the reactor coolant system material failure and component behavior models. The thermal-hydraulic information and material failure information are logically combined into a PRA model to determine the risk associated with the consequential steam generator tube rupture issue.

Dr. Powers provided a general overview and discussed the ACRS past involvement with this issue. He discussed the long standing relationship between the staff and the ACRS concerning the voltage-based repair criteria and what actions the staff should take as part of their plan on steam generators. Dr. Powers commented that the Letter Report generated by the Full Committee will contain enough background information to bring the Commission up to speed on this issue.

Mr. Tim McGinty, Director of the Division of Policy and Rulemaking in the Office of NRR, gave opening remarks and introduced the speakers.

Mr. David Beaulieu provided a historical overview, the desired outcome of the meeting, and future steam generator research activities regarding steam generators. He stated that during 1985-1990, the issue of consequential steam generator tube ruptures was first identified. Instead of the steam generator tube rupture itself being the initiating event, consequential tube ruptures refers to those tube ruptures that may be caused as a result of another initiating event (e.g., a very large MSLB that leads to high differential pressure across the steam generator tubes) or a severe accident condition that leads to failure of steam generator tubes. For severe accident induced consequential tube ruptures, Mr. Beaulieu stated that the concern was that the high temperature gases created during core damage sequences could cause steam generator tubes to be the first component of the reactor coolant pressure boundary to fail, resulting in a potential containment bypass and the release of large amounts of radioactive material outside containment.

Mr. Beaulieu discussed the Differing Professional Opinion. The Differing Professional Opinion was associated with an industry relaxation request. Mr. Beaulieu continued his presentation by mentioning that the Differing Professional Opinion was referred to the ACRS for resolution. After extensive public meetings and review of the issues that were raised in the Differing Professional Opinion, the ACRS published NUREG-1740 to present conclusions and recommendations. In particular, the ACRS concluded that the methodology being used to quantify the risk of containment bypass due to high-temperature challenges to tubes was "not technically defensible."

Steam Generator Action Plan Items 3.4.a-g (Thermal-Hydraulics - System Level Analyses and Computational Fluid Dynamics)

Steam Generator Action Plan item 3.4 directed the staff to develop a better understanding of RCS conditions and component behavior under severe accident conditions. Dr. Christopher Boyd along with Dr. Don Fletcher (Information Systems Laboratories) discussed the work done to complete task items 3.4.a-g in the Steam Generator Action Plan. Dr. Boyd was the lead for the CFD modeling and Dr. Fletcher performed the SCDAP/ Reactor Leak and Power Safety Excursion (RELAP5) modeling. Dr. Boyd indicated that the section of the plan he worked on focused on the thermal-hydraulic behavior of the system during severe accidents. He discussed the primary challenge to the tubes when the plant is in a "high-dry-low" condition. Various scenarios were also discussed to produce the "high-dry-low" condition. He also discussed a typical scenario called the "fast scenario." In the fast scenario, everything is assumed to fail, such as, loss of offsite power, failure of diesel generators to start, and failure of all auxiliary feedwater systems. Also, the core uncovers, oxidizes and produces significant power. He said after system heat up occurs, induced failure is predicted for RCS components. Dr. Boyd said that a more likely scenario involves auxiliary feedwater or operator actions that significantly delay the failure time. He discussed some of the key technical highlights of the work performed, such as predicted hot leg failure (with loop seal filled). Dr. Boyd explained some of the thermal-hydraulic issues raised in NUREG-1740 by the ACRS. The ACRS ultimately requested staff to develop a model to predict counter-current flow in the hot leg using CFD. In addition, ACRS requested the staff to provide additional analysis of reactor coolant pump loop seal clearing to support the system code models. NUREG/CR-6995 (draft) summarizes the work that has gone into addressing these issues. The work improves the understanding of the thermal-hydraulic behavior of the plant and addresses key criticisms of past analyses and covers task items 3.4.a, b, d, and f. The supporting CFD analyses (3.4.c, e, and g) are covered in NUREG-1781 and NUREG-1788 and NUREG-1922 (draft).

Dr. Fletcher indicated that the purpose of the SCDAP/RELAP 5 thermal-hydraulic analysis is to determine the plant configurations, conditions, and accident event sequence scenarios that can lead to containment bypass through induced steam generator tube failure. The risk associated with the accidents is affected by the order in which the reactor coolant system component structural failures occur. In particularly, he said that if a hot leg pressurizer surge line with the reactor vessel lower head fails, these failures lead to depressurization of the RCS into the containment, and the depressurization of the RCS precludes subsequent steam generator tube failures and containment bypass. He explained that RELAP5 code solves conservation of mass, momentum and energy equations using a two-fluid (steam/water), nonequilibrium, nonhomogeneous model with a noncondensible gas phase that is tracked with the steam. Also, SCDAP models severe accident core behavior such as fuel rod heat-up, oxidation, ballooning and rupture, fission product release, melting, flow and freezing of materials, and creep rupture failure of structures. Further, SCDAP/RELAP5 is capable of predicting buoyancy-driven flows in one-dimensional geometries but lacks capabilities for modeling on a first principles basis certain multidimensional flow behavior which is pertinent for this application. To compensate for this limitation, Dr. Fletcher said that SCDAP/RELAP5 model flow coefficients are adjusted (based on experiments and CFD predictions) to match important multidimensional hot/cool steam flow effects, such as, countercurrent flow in hot legs, mixing in steam generator inlet plenum and tube bundle flows. He discussed SCDAP/RELAP5-calculated base case event sequence and the results. For situations where the operators are assumed to take no action, the event sequences which assume very small leakage paths (flow area <0.1in^2/steam generator) for steam to escape the steam generator secondary system generally do not result in containment bypass. Also, event sequences which assume RCP shaft seal leakage rates below 180gpm/pump provide a potential for containment bypass were discussed. It was said that event sequences which assume RCP shaft seal leakage rates above 180 gpm/pump generally do not result in containment bypass (exception: late increases in the leak rate to above 400 gpm/pump lead to loop seal clearing and containment bypass, regardless of other assumptions). Situations where the operators use the pre-core damage strategy (steam generator feed-and-bleed cooling at 30 minutes using TDAFW system and opening the SG PORVs) were also discussed. In summary, Dr. Fletched indicated that previous ACRS review comments have been considered and addressed in the current analysis. Steam Generator Action Plan SCDAP/RELAP5 System Analysis Thermal Hydraulic tasks are addressed in Draft NUREG/CR-6995 (to be published in late 2009).

Steam Generator Action Plan Items 3.4.h-i (Materials – Potential RCS failure locations)

RES performed scoping review to determine potential failure locations, modes and time-to-failure for non-SGT RCS components during postulated PWR severe accident event. Gene Carpenter and Jeff Hixon presented the staff work and conclusion for task items 3.4. h-i. Mr. Carpenter said that the staff conducted a three phase scoping study. Phase I involved the review of methods and models for predicting failure modes and times-to-failure. It also identified additional information needed for the study and scoped RCS components that might be "weak links." Phase II involved the development of three dimensional computer models of selected components for Westinghouse 4-Loop plant utilizing detailed mechanical and structural drawings. It also included analyses of operating history of these components. Phase III utilized RELAP5 code and CFD and an expanded high-temperature materials database to calculate the failure sequence of the selected RCS components. He discussed some of the pertinent details of each phase, such as, components selected, failure times due to rupture, and the improvements that were made to thermal hydraulic modeling. In conclusion, Mr. Carpenter indicated that NRC has improved models for determining time-to-failure of non-SGT PWR RCS components under severe accident conditions. He also said that RES has determined that RCP seals could fail prior to steam generator tubes, which could avert or mitigate containment bypass.

Steam Generator Action Plan Item 3.10 (Using laboratory data for predicting field experience, such as, crack initiation and crack growth rates)

Mr. Carpenter pointed out that this work was not based on a specific ACRS recommended action in NUREG-1740. NRC staff monitors plant operating experience through inspection process and reviews of results of licensee SGT inspections. If analysis of future operating experience or research results indicates need to revisit this area, Mr. Carpenter indicated that the staff would prioritized the issue consistent with NRC budget process.

Steam Generator Action Plan Item 3.1 (Probability Steam Generator tube failures by Steam Generator depressurization events)

Robert Palla discussed tasks item 3.1. He indicated that 3.1a – 3.1j addressed physical processes that could cause steam generator tubes to open and leak (e.g., dynamic loads, bending stress). He said that the staff concluded that loads from MSLB would not lead to additional leakage or rupture beyond that from ∆P loads alone. The ACRS concluded that the analyses of MSLB have been completed and that SGAP Task 3.1 is closed, but this did not address 3.1k. Hence, the objective of task 3.1k was to develop probability distribution for total tube leakage under ∆P loads alone. He said that the result would be used to support resolution of GSI-163 and the task item 3.5. Mr. Palla said that the need for this calculation was diminished for several reasons. First, postulated phenomena associated with depressurization did not prove to be realistic. Second, performance-based TS provide reasonable assurance that DBA leakage will be small and well within that assumed in risk studies. Third, replacement steam generators result in fewer flawed tubes left in service and fewer proposals to increase allowable leakage. He concluded his presentation by stating that the calculations planned under Task 3.1k are not needed to support closeout of GSI-163 and that task 3.1k can be closed.

Steam Generator Action Plan Item 3.4j (Develop probability distribution for rate of tube leakage for ARC applied to flaws in restricted places)

Task 3.4j involved identifying predicted flaw areas and leak rates from cracks during MSLB and severe accidents. Mr. Palla indicated that example calculations under Task 3.5 assessed various defect types, including circumferential and axial cracks at the tube support plates to address this issue. He also said that a known model (SNL/SAIC model) can be used to assess the impact for flaws in restricted places. He indicated that task item 3.4j can be closed.

Steam Generator Action Plan Item 3.4k (Integrate information provided by Tasks 3.4a – 3.4j and 3.5 to address ACRS criticisms on risk assessments for ARC)

For task item 3.4k, Mr. Palla indicated that the specific concern related to ARC that credit "indications restricted against burst." This concern was specific to South Texas Project unit with stainless steel drilled-hole support plates. In depressurization events, tube support plates might move and expose flaws. To limit displacement, tubes were expanded at various elevations. Mr. Palla indicated that staff estimated conservative leak rate of 5 gpm per burst tube within tube support plates region. Result could be included in a risk calculation but was not pursued because South Texas Project steam generators were replaced. He iterated that Steam Generator Action Plan Task 3.5, "Develop improved methods for assessing the risk associated with steam generator tubes under accident conditions" was specifically intended to address this concern. He also said that the methods and results from task 3.5 provides insights into the risk significance of C-SGTR, as well as a foundation from which risk implications of future tube integrity issues might be assessed. Mr. Palla said that although additional research related to C-SGTR is planned, the work completed has achieved the intent of SGAP Task 3.4k and that this item should also be closed.

Steam Generator Action Plan Item 3.12 (Review Insights from Task 3.5 and Assess Need for Completing Additional Regulatory Guidance)

The need for a risk-related regulatory guide on C-SGTR was identified in COMSECY-97-013 "Steam Generator Rulemaking." Mr. Palla said that NRC made a decision to endorse NEI 97-06 initiative in lieu of issuing a generic letter. The proposed regulatory guide (DG-1073) was not completed. He said that the NEI document ensures that all steam generator tubes exhibit acceptable margins against burst/rupture for DBA. Mr. Palla did say that additional guidance and tools are still needed to support future risk assessments of C-SGTR. Development of this guidance will be part of an RES User Need now in concurrence.

Steam Generator Action Plan Item 3.5 (Assessing the risk associated with SG tubes under accident conditions)

This task involved developing improved methods for assessing the risk associated with steam generator tubes under accident conditions. Dr. Selim Sancaktar provided an illustrative PRA estimate of the C-SGTR bypass frequencies for selected plants, considering internal and external events using a straightforward and easily reviewable estimation method based on CDF sequences. The method used is exercised in a prudently conservative manner due to the existence of large uncertainties and large number of sub-scenarios and conditions. Dr. Sancaktar discussed the analysis method, sequence selection, and some illustrative examples. He indicated that both the NRC and the industry have studied the potential C-SGTR events in detail for the last decade and a half, as evidenced by multiple technical reports in this area. The conditions and sequences that can lead to C-SGTR are well studied and understood. He concluded his presentation by stating that task item 3.5 of the action plan has been completed and should be closed.

Summary

The closure of the Steam Generator Action Plan does not preclude future consequential steam generator tube failure research activities. Future work activities associated with this topic will be coordinated using other agency tools such as the User Need and the Planning, Budgeting, and Performance Management processes. Staff believes that they have developed a better understanding of the induced steam generator tube rupture process.

Comments and Issues:

Dr. Banerjee's overall concern was that the severe accidents at high RCS pressure represent the most significant contributors to risk. The initial stages of core degradation involve coolant boiloff and core heatup in a steam (and potentially hydrogen) environment. He stated that it has been argued that at high pressure, naturally circulating steam can redistribute the core thermal load throughout the primary system leading to failure of the pressure boundary prior to the occurrence of large-scale core melt. The most vulnerable locations for such failure were identified as the hot leg and steam generator tubes.

The likelihood of a thermally induced creep rupture of steam generator tubes and associated containment bypass depends on several factors including the thermal-hydraulic conditions at various locations in the primary and secondary systems, which determines the temperature and pressure to which steam generator tubes are subjected as the accident progresses. Other relevant factors include the effective temperature required for creep rupture failure of the steam generator tubes and the presence of defects in the steam generator tubes, which increases the likelihood of rupture.

Predictions of a plant's thermal-hydraulic response and conditions are the key to evaluating the failure times of specific reactor system components and calculation of conditional SG tube failure probability. Thermal-hydraulics analyses were performed using the SCDAP/RELAP5 systems analysis code aided by FLUENT CFD simulations that provided local three-dimensional details of the thermal conditions at specific locations of interest. The staff's thermal-hydraulic evaluation focused on severe accident scenarios that resulted from SBO events in

Westinghouse four-loop PWRs. The scenarios that challenged the steam generator tube's integrity involved either a counter-current natural circulation flow pattern or full-loop natural circulation flows (which are possible if the water in the loop seal and lower downcomer is cleared). During the latter scenario, the mass flow and heat-transfer rates, along with the amount of mixing and entrainment between the hottest and cooler flows, are significant factors in determining the timing of the RCS failures. If loop seals are cleared and full loop natural circulation is established, the challenge to the steam generator tubes increases because hot steam from the reactor vessel enters the steam generator tubes without the benefit of counter-flow cooling.

Dr. Banerjee identified the following issues associated with evaluating induced failure of steam generator during severe accidents:

1. Loop Seal Clearing: Steam passing through the steam generators may blow the water in the crossover legs out to either break or vessel. It is not clear whether RELAP has the capability to predict loop seal clearing.
2. Proper Modeling of Vessel T-H in CFD Calculations: The simplified representation of vessel for coupling of SCDAP/RELAP5 to CFD calculations of hot leg and steam generator should be further evaluated. The issue of mixing and potential for stratification in the vessel should also be addressed as part of this evaluation.
3. Sensitivity of CFD Results to Nodalization. The sensitivity of CFD results to the number of grids has not been addressed. Establishing grid independence to resolve the shear (mixing) layers is very important.

Chairman Powers stated that as a part of closing items of the Steam Generator Action Plan, the staff should prepare a single document that describes the current state of understanding of this issue and the technical issues that remain to be resolved for specific plant applications. In response to the comment, staff indicated that a Regulatory Information Letter has been prepared which makes an attempt to summarize the massive amount of documents on this issue.

Member Shack and Chairman Powers asked for who is the PRA method intended to be used by? In response, Dr. Sancaktar indicated that the staff will use the PRA to conduct rough estimates. Dr. Sancaktar discussed some example scenarios.

Chairman Powers noted that Bypass accidents and the risks associated with them were first identified in WASH-1400, ten years before the NUREG-1150.

SUBCOMMITTEE DECISIONS AND ACTIONS:

Following the presentations and discussions, Chairman Powers asked if anyone had any further questions, thanked everyone for their presentations, and then adjourned the meeting at 12:00 pm.

BACKGROUND MATERIALS PROVIDED TO THE SUBCOMMITTEE PRIOR TO THIS MEETING:

1. Memorandum from T. McGinty, Director, Division of Policy and Rulemaking, NRR, to E. M Hackett, Executive Director, ACRS, "Steam Generator Action Plan Items, 3.4, 3.5, and 3.12," 09/03/2009 (ML092310761)

2. Memorandum from T. McGinty, Director, Division of Policy and Rulemaking, NRR, to E. M. Hackett, Executive Director, ACRS, “Steam Generator Action Plan Items, 3.1k, 3.4j,and 3.4k,” 08/04/2009 (ML092110041)
3. Memorandum from T. McGinty, Director, Division of Policy and Rulemaking, NRR, to E. M. Hackett, Executive Director, ACRS, “Steam Generator Action Plan Item 3.10,” 06/05/2009 (ML091260536)
4. Memorandum from T. McGinty, Director, Division of Policy and Rulemaking, NRR, to E. M. Hackett, Executive Director, ACRS, “Steam Generator Action Plan Item 3.5 Closure,” 04/14/2009 (ML090980572)
5. Memorandum from E. M. Hacket, Executive Director, ACRS, to R. W. Borchardt, Executive Director for Operations, NRC, “ACRS Review of Steam Generator Action Plan Items,” 05/18/2009 (ML091320054)
6. Memorandum from M. Evans, Director, Division of Component Integrity, NRR, to E. M. Hackett, Executive Director, ACRS, “Proposed Closeout Package - Generic Safety Issue 163, ‘Multiple Steam Generator Tube Leakage,’ ” 03/09/2009 (ML090690074)
7. Letter Report from M. V. Bonaca, Chairman, ACRS, to G.B. Jaczko, Chairman, NRC, “Proposed Resolution of Generic Safety Issue-163, ‘Multiple Steam Generator Tube Leakage,’ ” 05/20/2009 (ML091320055)
8. Draft NUREG-1922, “Computational Fluid Dynamics Analysis of Natural Circulation Flows in a Pressurized-Water Reactor Loop Under Severe Accident Conditions,” 07/2009 (ML092230132)
9. Memorandum from B. Sheron, Director, RES to E. J. Leeds, Director, NRR and M. R. Johnson, Director, Office of New Reactors, Research Information Letter 09-003: Consequential Steam Generator Tube Rupture Work Performed in the Office of Nuclear Regulatory Research,” 08/21/2009 (ML092150157 and ML092150382)
10. Draft NUREG/CR-6995, “SCDAP/RELAP5 Thermal-Hydraulic Evaluations of the Potential for Containment Bypass during Extended Station Blackout Severe Accident Sequences in a Westinghouse Four-Loop PWR,” 06/2009 (ML091660110)
11. Memorandum from M. A. Cunningham, Director, Division of Risk Assessment, NRR, to J. A. Grobe, Associate Director, Engineering and Safety Systems, NRR, “Closure of Steam Generator Action Plan Task 3.1k,” 06/18/2009 (ML091120480)
12. A Risk Assessment of Consequential Steam Generator Tube Ruptures Final Report, RES 03/20/2009 (ML083540412)
13. “Behavior of PWR Reactor Coolant System Components, Other than Steam Generator Tubes, Under Severe Accident Conditions, Prepared by RES” 11/2008 (ML082900620)
14. Letter Report, JCN Y6486, “Severe Accident Initiated Steam Generator Tube Ruptures Leading to Containment Bypass – Integrated Risk Assessment,” prepared for the Office of Nuclear Regulatory Research by Sandia National Laboratories and Science Applications International Corp., 02/2008 (ML080500084)
15. Memorandum from J. T. Larkins, Executive Director, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, “Resolution of Generic Safety Issue 188, ‘Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steamline or Feedwater Line Breaches,’ ” 03/17/2006 (ML060870089)
16. Memorandum from J. T. Larkins, Executive Director, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, “Draft Final Generic Letter 2005-XX, Steam Generator Tube Integrity and Associated Technical Specifications,” 11/09/2005 (ML053170008)
17. Letter from M. V. Bonaca, Chairman, ACRS, to L. A. Reyes, Executive Director for Operations, NRC, “Resolution of Certain Items Identified by the ACRS in NUREG-1740, ‘Voltage-Based Alternative Repair Criteria,’ ” 11/17/2004 (ML043220681)
18. Letter from L. A. Reyes, EDO, NRC, to M. V. Bonaca, Chairman, ACRS, “Resolution of Certain Items Identified by the Advisory Committee on Reactor Safeguards in NUREG-

1740, 'Voltage-Based Alternative Repair Criteria,' " 08/25/2004 (ML0421902671)
19. Letter from M. V. Bonaca, Chairman, ACRS, to W. D. Travers, Executive Director for Operations, NRC, "Resolution of Certain Items Identified by the ACRS in NUREG-1740, Voltage-Based Alternative Repair Criteria," dated May 21, 2004 (ML041420237)
20. NUREG-1740, "Voltage-Based Alternative Repair Criteria," prepared by Advisory Committee on Reactor Safeguards, 03/2001 (ML0107503151)
21. Letter from D. A. Powers, Ad Hoc Subcommittee Chairman, ACRS, to W. D. Travers, Executive Director for Operations, NRC, "Differing Professional Opinion on Steam Generator Tube Integrity," 02/01/2001 (ML010780125)
22. Letter Report from G. E. Apostolakis, Chairman, ACRS, to R. A. Meserve, Chairman, NRC, "NRC Action Plan to Address the Differing Professional Opinion Issues on Steam Generator Tube Integrity," 10/18/2001 (ML012920749)
23. Letter Report from G. E. Apostolakis, Chairman, ACRS, to R. A. Meserve, Chairman, NRC, "Proposed Steam Generator Program Guidelines and Associated Generic License Change Package," 12/14/2001 (ML013540630)
24. Letter Report from G. E. Apostolakis, Chairman, ACRS, to R. A. Meserve, Chairman, NRC, "Response to Your May 7, 2001 Memorandum Regarding Differing Professional Opinion on Steam Generator Tube Issues," 06/14/2001 (ML011700613)
25. Memorandum from B. Sheron, Associate Director for Projects, Licensing and Technical Analysis, and J. Johnson, Associate Director for Inspection and Programs, to S. Collins, Director, NRR, "Steam Generator Action Plan," 11/16/2000 (ML003770259)
26. Memorandum from D. A. Powers, Chairman, ACRS, to W. D. Travers, Executive Director for Operations, NRC, "Differing Professional Opinion on Steam Generator Tube Integrity Issues," 09/11/2000 (ML091070471)

****

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/reading-rm/doc-collections/acrs/tr/subcommittee/2008/ or purchase from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers) 1323 Rhode Island Avenue, NW, Washington, DC 20005 (202) 234-4433.