October 11, 2000

MEMORANDUM TO: ACRS Members

/RA/

FROM: Noel Dudley, Senior Staff Engineer
ACRS\ACNW

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY CONCERNING THE PRESSURIZED THERMAL SHOCK TECHNICAL BASIS REEVALUATION PROJECT, SEPTEMBER 21, 2000 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on September 26, 2000, 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: Technical Support Branch
Operations Support Branch (3 copies)

cc via e-mail:
J. Larkins
J. Lyons
ACRS Fellows and Technical Staff
E. Barnard

Issued: September 26, 2000

CERTIFIED: October 6, 2000 **CERTIFIED** ACRS-3216

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF SUBCOMMITTEE MEETING ON
MATERIALS AND METALLURGY
PTS TECHNICAL BASIS REEVALUATION PROJECT
SEPTEMBER 21, 2000
ROCKVILLE, MARYLAND

The ACRS Subcommittee on Materials and Metallurgy met on September 21, 2000, to hold discussions with the NRC staff and its consultants concerning the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. The meeting included presentations concerning the activities associated with the initial results of thermal-hydraulic experiments and the development of a flaw distribution, fracture toughness distributions and model uncertainties, embrittlement correlations, and the FAVOR probabilistic fracture mechanics code.

The entire meeting was open to public attendance. Mr. Noel Dudley was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:35 a.m. and adjourned at 4:00 p.m.

ATTENDEES

ACRS

W. Shack, Chairman
G. Apostolakis, Member
T. Kress, Member
R. Seale, Member
N. Dudley, ACRS Staff

NRC REPRESENTATIVES

E. Hackett, RES
S. Malik, RES
D. Jackson, RES
L. Abramson, RES
M. Kirk, RES
N. Siu, RES
H. Woods, RES
D. Bessette, RES
D. Kalinousky, RES
T. Dickson, ORNL
M. Modarres, University of Maryland

There were no written comments or requests for time to make oral statements received from members of the public. Four members of the public attended the meeting. A list of meeting attendees is available in the ACRS office files.

INTRODUCTION

Dr. William Shack, Chairman of the Materials and Metallurgy Subcommittee, explained that the purpose of the meeting was to review activities related to the PTS Technical Basis Reevaluation Project. He noted that the staff had briefed the Subcommittee on the PTS Reevaluation Project activities on March 16 and April 27, 2000. He called upon Mr. Edwin Hackett, Assistant Chief of the Materials Engineering Branch, Office of Nuclear Regulatory Research (RES), to begin.

PTS TECHNICAL BASIS REEVALUATION PROJECT - Mr. Edwin Hackett, RES

Mr. Edwin Hackett, RES, provided background information concerning the PTS Reevaluation Project. He described the current status of the activities being performed by the three groups of staff and industry experts that are working in the areas of thermal-hydraulics, probabilistic risk assessment (PRA), and probabilistic fracture mechanics (PFM). Mr. Hackett noted that consideration of large, early release frequency (LERF) and containment integrity is a major departure from the current PTS framework. He stated that although initial indications pointed toward relaxation of current criterion, the final outcome is not yet clear.

The Subcommittee members and the staff discussed the evolution of probabilistic fracture mechanics codes, how uncertainties are characterized and propagated thorough the Fracture Analysis of Vessels: Oak Ridge (FAVOR) code, and the use of insights from Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the License Basis." They also discussed whether the integrity of the containment would be maintained if a PTS event resulted in a reactor vessel failure and core damage.

Dr. Kress noted that reactor vessel failures caused by PTS events result in a source term driven by air oxidation of zircaloy clad instead of steam oxidation. He questioned whether a PTS screening criterion based on 5×10^{-6} events per reactor year would adequately account for the source term resulting from air oxidation. The staff agreed to consider the effect of the air oxidation source term during its derivation of a revised PTS screening criterion.

PROBABILISTIC RISK ASSESSMENT GROUP - Mr. Hugh Woods, RES

Mr. Hugh Woods, RES, explained that the objectives of the PRA working group are to ensure that the overall process is coherent and risk-informed, to develop a screening criterion, and to update the Integrated Pressurized Thermal Shock (IPTS) PTS/PRA studies. He presented the overall framework and status of the PRA event sequence, thermal-hydraulic, and PFM analyses. Mr. Woods explained the event tree and a representative scenario for a PTS accident that would result in reactor vessel failure. He described the information that would be used in a plant-specific analysis. Mr. Woods concluded that the interactions between the three working groups are leading to improvements in the analyses.

The Subcommittee members and the staff discussed the need to develop PTS event trees since most PRAs have screened out these events and the use of A Technique for Human Event Analysis (ATHEANA) in developing the event trees. They also discussed the treatment of aleatory and epistemic uncertainties presented in the paper by Mr. F. Li, University of Maryland, titled "K_{Ic} / K_{Ia} Uncertainty Characterization." The staff agreed to brief the Subcommittee at a future meeting on the treatment of uncertainties in the overall PTS Reevaluation Project as described in the white paper by Mr. Nathan Siu.

THERMAL-HYDRAULIC GROUP - Mr. David Bessette, RES

Mr. David Bessette, RES, explained that the objective of the thermal-hydraulic working group is to update the thermal-hydraulic inputs developed at the time of the IPTS studies if these inputs are no longer operative. He identified the four participating plants and the thermal-hydraulic issues

being evaluated. He described the results of the initial RELAP Code analysis for Oconee and presented initial conclusions [see attached slide #1]. Mr. Bessette described how the results of the RELAP code will be validated and how the thermal-hydraulic uncertainties will be evaluated. He presented the status of experiments being conducted at the APEX facility and the schedule for completing similar thermal-hydraulic calculations for the Beaver Valley, Calvert Cliffs, and Palisades reactors.

The Subcommittee members and the staff discussed the treatment of uncertainties, the stability of the RELAP code during the two hours of transient time that was evaluated, use of the results of the REMAX code, and the differences between the boundary conditions used in the IPTS studies and the more recent RELAP code results.

GENERALIZED FLAW DISTRIBUTION REPORT - Ms. Deborah Jackson and Mr. Lee Abramson, RES

Ms. Deborah Jackson, RES, presented the process used to determine the flaw distribution in reactor pressure vessels. She explained how an expert panel was used to resolve specific technical issues for which there is significant scientific uncertainty. Ms. Jackson described the flaw distribution data obtained from detailed examination of the Pressure Vessel Research User Facility (PVRUF) vessel and the Shoreham vessel, and the comparison of this data to the Marshall flaw data distribution used for previous PTS studies. Ms. Jackson provided details of the expert judgement process and presented the panel's conclusions [see attached slides #2 and #3). She concluded that the staff identified fabrication process factors that are important in considering the introduction of flaws into the reactor pressure vessel.

Mr. Lee Abramson, RES, presented the generalized flaw distribution methodology, the PVRUF flaw distribution, and the Complementary Cumulative Distribution Function for flaws. He described how the flaw distribution would be used as an input to the FAVOR code. Mr. Abramson explained that the generalized flaw distribution combines the densities of flaws, crack depth distributions, and plant specific volumes and areas. He reiterated that the generalized flaw distributions being developed would be based on all available relevant data and that expert judgement would be used where necessary to account for factors not addressed in the data.

The Subcommittee and the staff discussed the following:

- weld processes used to construct boiling water reactor pressure vessels,
- the objective of the expert judgement process,
- clarification of which portions of the flaw distribution are derived from data and which are derived from expert judgement,
- validation of the flaw distribution,
- treatment of overlapping issues,
- how the opinion of each expect is weighted, and
- agreement of flaw distribution with results of the PRODIGAL model.

PROBABILISTIC FRACTURE MECHANICS GROUP - Mr. Shah Malik, RES

Mr. Shah Malik, RES, presented the major activities of the PFM working group. He described the objective, responsible staff, NRC contractors, and status of the following activities for developing:

- a fabrication flaw distribution,
- fracture toughness curves,
- embrittlement correlations,
- material chemistry distributions,
- beltline neutron fluence maps, and
- a revised FAVOR computer code.

Mr. Malik explained that several analysis models are being finalized, the PRA and thermal-hydraulic analyses of Oconee have started, and rigorous uncertainty models are being developed for key variables.

The Subcommittee members and the staff discussed uncertainty models, revising the margin term in Regulatory Guide 1.99, “Radiation Embrittlement of Reactor Vessel Materials,” changes to parameters as a result of power uprates, and how preliminary results match the results from the IPTS studies.

EMBRITTLEMENT CORRELATIONS - Mr. Mark Kirk, RES

Mr. Mark Kirk, RES, explained how the revised embrittlement curves have application to the PTS rule and Regulatory Guide 1.99. He presented modeling considerations and provided examples of how specific variables were modeled. He presented the gating criteria used to determine which variables would be modeled in the FAVOR Code. Mr. Kirk showed the relative changes between the present Regulatory Guide 1.99 curves and the new trend curves. He outlined a framework for treating uncertainties and suggested potential applications of the revised embrittlement curves.

The Subcommittee members and the staff discussed the relationship between the embrittlement curves and the J_R curve correlation proposed by Prof. Eason, the effect of the revised embrittlement curves on reactor pressure vessel heatup and cooldown curves, and aging effects that are independent of neutron embrittlement.

FRACTURE TOUGHNESS DISTRIBUTIONS AND UNCERTAINTIES - Mr. Mark Kirk, RES

Mr. Mark Kirk, RES, explained that the goal of this activity is to characterize toughness using all available data in a way that is consistent with PRA insights. He presented how NRC contractors collected and assembled data for evaluating fracture toughness and how they developed interim fracture toughness curves. He described how uncertainties associated with the fracture toughness curves were established. Both the interim curves and the associated uncertainties will be inputs to the FAVOR code.

Mr. Kirk introduced root cause diagrams and explained how they could be used to display a complex process in a logical format and help provide a more rigorous treatment of uncertainties. He demonstrated how the root cause diagrams were used to develop more robust procedures for

selecting fracture toughness and uncertainty values for a specific reactor pressure vessel. He described a model that predicts the shift in reactor pressure vessel reference temperatures (RT_{NDT}) due to irradiation embrittlement. Mr. Kirk stated that on-going activities included full implementation of uncertainty models in the FAVOR code, determining RT_{NDT} bias correction functions and modeling procedures, and assembling input data to run the models.

The Subcommittee members and the staff discussed how the uncertainties associated with RT_{NDT} should be accounted for in the characterization of the fracture toughness distribution.

STATUS OF FAVOR CODE DEVELOPMENT - Mr. Terry Dickson, Oak Ridge National Laboratory

Mr. Terry Dickson, ORNL, presented how the FAVOR code is applied in the PTS Technical Basis Reevaluation Project. He explained that the FAVOR code was intended to combine the best features of earlier PFM codes that had been used in the development of the original PTS rule. He described how elements of updated technology, such as enhanced neutron flux maps and new statistical models for static initiation and arrest fracture toughness, are being integrated into the FAVOR Code.

Mr. Dickson presented the FAVOR code structure and described the overall PRA methodology. He described the load generator, PFM, and post processor modules of the FAVOR code. He explained the analyses performed by each module and how the final probability of reactor pressure vessel failure is calculated. He concluded the presentation by providing the near term schedule for continued development of the FAVOR code.

The Subcommittee members and the staff discussed what elements of the updated technology results would provide the greatest relaxation to the PTS screening criterion and what the form would be of the thermal-hydraulic inputs to the FAVOR Code. They held an extended discussion on how fracture toughness curves were used to arrive at a probability of reactor vessel failure and the statistical treatment of the data associated with the curves.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

Dr. George Apostolakis provided the following recommendations concerning the draft report on the results of the expert judgement process:

- add sections to the report that clearly state the objectives and describes the methodology of the expert judgement process;
- show the calculations that were used to derive the flaw distribution from the experts' inputs;
- explain how the experts' inputs are equally weighted: for example plot the flaw distribution for each expert and take the mean value at different points; and

- provide more context for the increased flaw density.

Dr. William Shack provided the following recommendations concerning the draft report on the results of the expert judgement process:

- explicitly separate which portions of the flaw distribution were derived from data and which were derived from expert judgement,
- add the Pacific Northwest National Laboratory table of flaw data, and
- add the bead size for welds associated with the flaw distribution.

Dr. Thomas Kress suggested that the staff needed to consider an acceptance criterion applicable to PTS events before the PTS Reevaluation Project could be completed. Dr. Kress explained that PTS events result in air-oxidation of the fuel instead of steam-oxidation. As a consequence, the source term may be higher and hence the vessel failure frequency may need to be lowered to ensure the same level of risk for PTS events as the current LERF assessment guideline provides for events where a source term for steam-oxidation is appropriate. Dr. Kress noted that such considerations may markedly lower the acceptance criterion for RPV failure frequency below 1 X 10^{-6} per year.

STAFF AND INDUSTRY COMMITMENTS

The staff agreed to provide Dr. Apostolakis with a copy of NUREG/CR-5505, “RR-Rodical - A Model for Estimating the Probabilities of Defects in Reactor Pressure Vessel Welds.” [received 9/22/2000]

The staff agreed to provide Dr. Apostolakis a copy of NUREG/CR-6471, “Characterization of Flaws in U.S. Reactor Pressure Vessels.” [Volumes 1 and 3 received 9/22/2000] [Volume 2 received 9/26/2000]

The staff agreed to consider the effect of the air-oxidation source term during its derivation of the PTS screening criterion.

The staff agreed to brief the Subcommittee on the treatment of uncertainties in the FAVOR Code at a future meeting, including the information presented in the white paper by Mr. Siu and in the paper by Mr. F. Li, University of Maryland, titled “K_{Ic} / K_{Ia} Uncertainty Characterization.”

SUBCOMMITTEE DECISIONS

The Subcommittee requested that the staff brief the Committee on the status of the development of the FAVOR Code and on the fracture toughness distributions and uncertainties at the October 5-7, 2000 ACRS meeting.

The Subcommittee requested that the staff brief it on the development of a criterion for deriving the PTS screening criterion.

FOLLOW-UP ACTIONS

None

PRESENTATION SLIDES AND HANDOUTS PROVIDED DURING THE MEETING

The presentation slides and handouts used during the meeting are available in the ACRS office files or as attachments to the transcript.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE:

1. Dickson, T., Oak Ridge National Laboratory, and Malik, S., U.S. Nuclear Regulatory Commission, "An Updated Probabilistic Fracture Mechanics Methodology for Application to Pressurized Thermal Shock," issued in the Proceedings from the IAEA Specialists' Meeting, "Methodology and Supporting Research for the Pressurized Thermal Shock Evaluation," Rockville, MD, July 2000.
2. Jackson, D., and Abramson, L., U.S. Nuclear Regulatory Commission, "Draft Report on the Results of the Expert Judgement Process for the Generalized Flaw Size and Density Distribution for Domestic Reactor Pressure Vessels," received September 7, 2000.
3. Eason, D., and Wright, J., Modeling & Computing Services, "Draft Report on Updated Transition Temperature Shift Model," dated July 28, 2000.
4. Bowman, K., and Williams, P., Oak Ridge National Laboratory, "Technical Basis for Statistical Models of Extended K_{Ic} and K_{Ia} Fracture Toughness Databases for RPV Steels," dated February 2000.
5. Li, F., et. al., University of Maryland, " K_{Ic} / K_{Ia} Uncertainty Characterization," dated June 23, 2000.

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20006, (202) 634-3274, or can be purchased from Ann Riley & Associates, LTD., 1025 Connecticut Ave., NW, Suite 1041, Washington, D.C. 20036, (202) 842-0034.

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RELAP Analysis of Oconee Conclusions

- Useful interchange between PRA and thermal hydraulics. Good progress made on Oconee-1

- The break spectrum that was calculated for Oconee-1 covers the range of interest from where the primary system pressure stays near the steam generator secondary pressure to where the primary system pressure drops below accumulator pressure and further to low pressure injection

- The results are sensitive to the trip criteria for the reactor coolant pumps on loss of subcooling. Once subcooling is lost in a small break LOCA it will generally not be reestablished unless the break can be isolated.

- Stuck open valves on the secondary combined with a small break on the primary help to maintain subcooling and, therefore, the reactor coolant pumps are not tripped

- When the reactor coolant pumps are tripped, stagnation beginsand the downcomer cools in response to high pressure injection

- Cold leg breaks slightly less severe than hot leg breaks (similar to IPTS results)

Some Conclusions of the Expert Judgment Process

- A generalized flaw distribution for domestic RPVs can be developed but with a wide range of uncertainty
- The flaw density of base metal is substantially less than that for weld metal
- Discontinuities in the cladding that do not affect the clad base metal interface may not be of concern for RPV integrity

Some Conclusions of the Expert Judgment Process (con't)

- Large flaws are usually caused by loss of control of the welding process and would be detected by NDE
- Variations in weld quality are greatly affected by welder skill and inspector skill
- Weld processes are an important factor in the introduction of flaws

#3