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

April 14, 2009

MEMORANDUM TO: ACRS Members

FROM: Michael Benson, Staff Engineer ***/RA/***
Reactor Safety Branch A, ACRS

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE MEETING, MARCH 4, 2009 – ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on April 2, 2009, have been certified as the official record of the proceedings for that meeting. A copy of the certified minutes is attached.

Attachment: Certification Letter
Minutes

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

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

April 14, 2009

MEMORANDUM TO: Michael Benson, Staff Engineer
Reactor Safety Branch A, ACRS

FROM: William J. Shack, Chairman
Materials, Metallurgy, and Reactor Fuels Subcommittee

SUBJECT: CERTIFICATION OF MINUTES OF THE ACRS MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE MEETING, MARCH 4, 2009 – ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on March 4, 2009 are an accurate record of the proceedings for that meeting.

William J. Shack, Chairman
Materials, Metallurgy, & Reactor Fuels
Subcommittee

April 14, 2009
Dated

Certified by: William Shack Issued on April 14, 2009
Certified: April 14, 2009

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

MINUTES OF ACRS MATERIALS, METALLURGY, AND REACTOR FUELS SUBCOMMITTEE MEETING

MARCH 4, 2009

ROCKVILLE, MARYLAND

The Advisory Committee on Reactor Safeguards (ACRS) Materials, Metallurgy, and Reactor Fuels Subcommittee held a meeting on March 4, 2009 in Room O1G16, 11555 Rockville Pike, Rockville, MD. The purpose of this meeting was to review issues related to the proposed rule amendment to Title 10, Code of Federal Regulations (CFR) 50.61: "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." Michael Benson 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 March 4, 2009 at 1:30 p.m. and adjourned at 5:20 p.m.

Attendees:

ACRS Members
William Shack, Chairman
J. Sam Armijo
Sanjoy Banerjee
Dennis Bley
Michael Corradini
Dana Powers
Jack Sieber

ACRS Staff
Michael Benson, Designated Federal Official

NRC Staff
Meg Audrain, NRR
Stephen Dinsmore, NRR
Jen Gall, NRR
Mark Kirk, RES
Jason Lising, NRR
Matthew Mitchell, RES
Geary Mizuno, OGC
Ted Quay, NRR
Tim Reed, NRR
Stuart Richards, RES
Veronica Rodriguez, NRR
Dan Wicheritz, NRR
Jake Zimmerman, NRR

Other
William Arcieri, Information Systems Laboratories

Other members of the public attended this meeting. A complete list of attendees is in the ACRS Office File and is available upon request.

Opening Remarks and Objectives:

Dr. William Shack, Chairman of the ACRS Materials, Metallurgy, and Reactor Fuels Subcommittee, convened the meeting at 1:30 p.m. The purpose of this meeting was to review and discuss the proposed rule amendment to 10 CFR 50.61. The presenters included representatives from the Nuclear Regulatory Commission's (NRC) Offices of Nuclear Reactor Regulation (NRR) and Nuclear Regulatory Research (RES). The Subcommittee gathered information, analyzed relevant issues and facts, and formulated proposed positions and actions, as appropriate, for deliberation by the full Committee. The rules for participation in the meeting were 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.

Chairman Shack stated that concern for pressurized thermal shock (PTS) events arises from scenarios in which cold water is injected into the vessel. This scenario gives rise to high thermal stresses on the vessel surface. If the thermal stresses and vessel embrittlement are high enough, then cracks can grow and penetrate the vessel wall. Understanding this behavior requires consideration of the likelihood of sequences leading to injection into the vessel and the thermal hydraulics of that process. The likelihood of cracking is computed from a probabilistic fracture mechanics code that accounts for the thermal hydraulic challenge, the embrittlement of the vessel, and the flaw distribution in the vessel. RES has studied this problem in detail for three plants, indicating that the current regulations governing PTS are overly conservative. NRR is developing a new PTS rule incorporating the insights from the research. Ensuring that the results from the detailed study of three plants may be applicable to a broader range of plants is a major issue in developing the new rule. At the previous subcommittee meeting on this topic, the Subcommittee questioned whether it should be necessary to demonstrate that the likelihood and severity of PTS challenges for a specific plant are comparable to those from the three plants in the detailed study. The generalization of the results is the major technical focus of today's presentations.

Final Rule Language:

Overview

Mr. Matt Mitchell, NRR, began with an overview of the proposed alternate PTS rule, 10 CFR 50.61a. It was structured to be similar to the existing rule, 10 CFR 50.61. The similarity was adopted in order to facilitate implementation and understanding of the new rule.

10 CFR 50.61a(a) – Definitions

Many definitions used in the existing rule were maintained for the alternate rule. The definition of "American Society of Engineers (ASME) Code" was broadened to include Section XI to address inservice inspection (ISI), which was not a feature of 10 CFR 50.61. Terms specific to the alternate rule were added and appropriately defined.

10 CFR 50.61a(b) – Applicability

The alternate rule will be applicable only to the existing fleet of pressurized water reactors (PWRs) and those plants with similar technology. Advanced designs (e.g., AP1000) were deemed inconsistent with the technical basis and, thus, excluded from the range of applicability of 10 CFR 50.61a. While an advanced-reactor licensee could employ the 10 CFR 50.12

exemption process to become eligible for the alternate PTS rule, PTS is not expected to be a major engineering concern for these new designs.

10 CFR 50.61a(c) – Request for Approval

This section requires submission of a license amendment application in accordance with 10 CFR 50.90. This was deemed appropriate due to the significance of the issue and the complexity of 50.61a. The application to utilize the alternate rule must be submitted at least three years prior to the date on which the applicant projects violation of 10 CFR 50.61. Each licensee requesting to apply the alternate rule must first be granted NRC staff approval. The application must include material property values compared to the screening criteria and an analysis of ISI data to demonstrate consistency with the technical basis of the rule. Surveillance capsule data must be accounted for when determining the reference temperatures for the application.

10 CFR 50.61a(d) – Subsequent Requirements

After receiving initial approval to use the alternate rule, the licensee must submit updated reference temperatures calculated according to 10 CFR 50.61a methods and updated evaluations of ISI data to ensure the vessel's flaw distribution is consistent with that of the technical basis. The evaluation of the ISI data is necessary because the major technical difference between 50.61 and 50.61a is the incorporation of a more realistic flaw distribution in the probabilistic fracture mechanics calculation. If the licensee ever determines that the limiting vessel reference temperature will exceed the screening limits, embrittlement mitigation schemes (e.g., annealing of the vessel and neutron flux reduction) defined within the rule may be employed to ensure compliance.

10 CFR 50.61a(e) – Examination and Flaw Assessment

Section (e) requires a detailed evaluation of the plant-specific flaw distribution in order to ensure consistency with the technical basis for 50.61a. This evaluation is based upon the inspections required by the ASME Code Section XI, Appendix 8, Supplements 4 and 6. According to Section (e), the results of the nondestructive evaluation (NDE) may be adjusted according to the licensee's knowledge of NDE uncertainty. Near-surface defects on the inner diameter of the vessel are not permitted to open to the surface. If the plant-specific flaw distribution is found to be inconsistent with that used in the technical basis, then the licensee is instructed to perform a plant-specific analysis that demonstrates that the through-wall cracking frequency (TWCF) is still acceptable.

10 CFR 50.61a(f) – Calculation of RT_{MAX-X} Values

This section describes the methodology for calculating the reference temperatures. This new procedure does not include a margin term, as is employed in 10 CFR 50.61, because the uncertainty in these values (and other uncertainties such as thermal hydraulic uncertainty) has been considered in the determination of the acceptance criteria. The licensee must calculate separate reference temperatures for welds, plates, and forgings. The maximum of these values is compared to the screening criteria. The reference temperature values must be based upon updated embrittlement models and surveillance-capsule data. The updated embrittlement models included use of (1) expanded databases of Charpy Impact data using surveillance-capsule samples, (2) combined use of new statistical analysis of the Charpy Impact data and a

mechanistic understanding of radiation embrittlement, and (3) incorporation of a wider range of material and environmental variables.

Updated surveillance data evaluations have been included to verify the applicability of the embrittlement models used in 50.61a. The staff is confident in the applicability of the embrittlement models, such that plant-specific data should only be used when there is clear evidence that the models are not correctly describing the vessel's behavior. Three statistical tests were developed with the help of Lee Abramson to demonstrate the validity of the embrittlement model for a specific plant. The uncertainty in the embrittlement model increases particularly at high fluence levels.

Public Comments:

Overview

Ms. Veronica Rodriguez, NRR, stated that the proposed rule was issued for public comment on October 3, 2007, with the comment period closing on December 17, 2007. NRC staff received comment letters from five groups: the PWR Owners Group (PWROG), Electric Power Research Institute (EPRI), Nuclear Energy Institute, Duke Energy, and Strategic Teaming and Resource Sharing. The supplemental proposed rule was issued for public comment on August 11, 2008, with the comment period closing on September 10, 2008. The staff received comment letters from the PWROG, EPRI, and FirstEnergy Nuclear Operating Company.

NRC Approach

The public comments were binned into categories. The categories for the proposed rule were Updated Embrittlement Trend Curves and Fluence Maps, Surveillance Data Evaluation, Flaw Limits and Flaw Density Determinations, and Miscellaneous. The categories for the supplemental proposed rule were Adjustments of ISI Volumetric Examination and Surveillance Data Evaluation. Mr. Mitchell stated that the discussion of the public comments is not exhaustive, due to the large number of comments received.

Updated Embrittlement Trend Curves and Fluence Maps

Mr. Mitchell stated that the public asked for removal of the embrittlement model, requiring instead the use of an NRC-approved methodology. The staff disagreed with this comment, stating that the model ensured consistency with the technical basis and regulatory certainty.

Surveillance Data Evaluation (Proposed Rule)

The public requested removal of the requirement to assess surveillance data, since variability is included in the model derivation. The staff rejected this comment on the grounds that the surveillance-data evaluations ensure that the embrittlement models remain conservative, especially at high neutron fluence.

Flaw Limits and Flaw Density Determinations

The comments in this category were varied, requesting certain improvements or clarifications to Section (e). NRC staff accepted some of these comments and incorporated them accordingly into the final rule language.

Miscellaneous

One example from the Miscellaneous category was concerned with evaluation of steam generator overfeed events. The staff reevaluated the event sequence and found that the screening criteria in 50.61a were not impacted.

Adjustments to ISI Volumetric Examination (Supplemental Proposed Rule)

The public agreed with the NRC staff that potential consideration of NDE uncertainty was appropriate for the rule. The staff responded by implementing this aspect into the final rule.

Surveillance Data Evaluation (Supplemental Proposed Rule)

The staff received comments requesting that the slope test for surveillance data be eliminated from the rule. The staff disagreed with this comment on the basis that the slope test provides assurance that the embrittlement model is still appropriate at high neutron fluence levels.

Generalization Studies:

Plants Studied (Detailed Study)

Mr. Mark Kirk, RES, stated that three PWRs were studied in detail to develop the technical basis of 10 CFR 50.61a. Two of these plants were close to the screening criteria of the current PTS rule.

Transient Classes Modeled (Detailed Study)

Three classes of transients represented 99.99% of the TWCF in the detailed study of three plants: medium- and large-break loss-of-coolant accidents (LOCAs) with pipe breaks 5 in. or larger, stuck-open primary valves, and main steam line breaks (MSLBs). The pipe-break scenarios are characterized by rapid depressurization and cooling of the vessel. The effects of these transients are determined solely by the thermal conductivity of the vessel, so other details of the circumstances surrounding the transient are insignificant. This situation supports the idea that the results of the technical basis may be generalized to the entire fleet of PWRs.

Stuck-open primary valves, which are similar to small-break LOCAs, become significant because of the late-stage repressurization combined with the thermal stresses and cooler temperature. The contribution of MSLBs is minor because the primary temperature cannot decrease as low as it can for a LOCA.

At low embrittlement levels, LOCAs and stuck-open valves contributed roughly equal amounts to the TWCF, although the absolute value of the TWCF at low embrittlement levels is very small. As vessel embrittlement increases, LOCAs begin to dominate the TWCF, the effect of stuck-open valves decreases, and the MSLB transients begin to have a noticeable impact. Other transients, such as feed-and-bleed scenarios and the steam generator overfeed event mentioned by Duke Energy, result in a low challenge to vessel integrity.

Mr. Kirk stated that the staff wanted to know why the three transient classes were dominating the TWCF in order to determine whether generalization was possible. For stuck-open primary valves, the temperature of the primary water approaches the injection-water temperature, which is similar across the fleet of PWRs. Also, the late-stage repressurization, which is a necessary

feature in order to have significant challenge to vessel integrity for these events, increases the pressure to the safety-valve setpoint, which is similar across the entire fleet. Crediting operator actions to mitigate the late-stage pressurization does not significantly impact the screening limits. Significant impact from operator actions would have made generalizing the rule more difficult.

Summary of Findings (Detailed Study)

Primary-side faults were found to dominate the risk, since the temperatures can fall considerably below 212°F. Very severe secondary faults, such as MSLBs, do have a minor contribution to the TWCF. Other classes of transients were found to be insignificant to the risk.

Stuck-open primary valves dominated the risk at low embrittlement levels. The factors that lead to failures during these transients are the low reactor coolant temperatures at the time of repressurization and the repressurization to the safety-valve setpoint. These factors are similar across the entire PWR fleet. While rapid operator action can be important during this scenario, removing credit for operator action does not impact the screening criteria. A transient during full power is believed to be more severe than one at hot zero power. The fact that operator actions were found to have minimal impact on the screening criteria suggests that generalization is possible.

Medium and large LOCAs dominated the risk at higher embrittlement levels. Transient severity depends upon the steel thermal conductivity and the vessel geometry. Operator actions were not relevant for this class of transient.

The aim of the generalization study was to determine if the screening criteria, which were developed from the detailed study of three PWRs, are applicable to all PWRs. For generalization, the question to be answered is: Are the probabilistic risk assessment- (PRA-) and thermal hydraulic- (TH-) related plant characteristics that affect the screening criteria similar when comparing the three plants studied in detail and the PWR fleet?

Methodology

If there were differences in the frequency or severity of PTS challenges, then the effects of these differences would be most apparent at an embrittled plant. Therefore, five high-embrittlement plants were chosen to study the possibility of generalizing the rule to all PWRs.

Frequency of PTS Challenge

Mr. Steve Dinsmore, NRR, stated that the generalization study considered the possible changes to sequence frequencies and TH characteristics, such as the temperature in the refueling water storage tank. An increase in the frequency of a dominant sequence would affect the results. Furthermore, identifying any previously unimportant sequences (as determined from the three-plant study) that become significant is necessary for generalization.

Severity of PTS Challenge

The severity of PTS challenges must be considered, as well. The important transients in the three-plant study should have a similar severity at the five plants of the generalization study. The results of this study are documented in “Generalization of Plant-Specific Pressurized

Thermal Shock Risk Results to Additional Plants," dated December 14, 2004. This information is also summarized in Section 9.3 of the main report.

Probabilistic Risk Assessment Scenarios

The investigators identified five general PRA event scenarios and evaluated them for differences between the plants. These scenarios were related to secondary breaches, secondary overfeed, medium and large LOCAs, power-operated relief valves (PROV) and safety/relief valves (S/RVs), and feed-and-bleed conditions. These events do not represent core-damage sequences. The sequences stop when either the PTS event occurs or is avoided. Small LOCAs were not considered, since a repressurization using high-pressure injection at the PORV setpoints are covered by feed-and-bleed scenarios.

Secondary breaches include MSLBs and valve failures, each of which results in the uncontrolled release of steam. The investigators asked the five plants how operators identified faulted steam generators (S/Gs), isolated feed from faulted S/Gs, and properly steamed operating S/Gs. They concluded that the lack of auxiliary feedwater isolation at some units may increase the frequency of excessive cooldown, but that operators had multiple opportunities to identify and isolate faulted S/Gs. The investigators determined that these events will not become important at any operating PWR.

The secondary overfeed events involved the operators severely overfeeding an S/G. The medium and large LOCAs became dominant at high embrittlement levels. Operator actions have negligible affect on these scenarios; therefore, either a PTS challenge occurs or it does not.

The PORV- and S/RV-related events were most important at low embrittlement levels. These scenarios included failure of a system or operators to avoid excessive primary cooldowns or cold repressurization. The investigators considered the number and sizes of PORVs and S/RVs, the capability to identify stuck-open valves, and procedures for coping with stuck-open valves and valve reclosure. They concluded that differences in the capability to identify stuck-open valves may increase the scenario frequency but not enough to cause the event to be important at high embrittlement. The significance of these events is not expected to change for any PWR.

The feed-and-bleed events tend to be unimportant contributors regardless of embrittlement level. They result from an initiating transient followed by loss of secondary-side cooling. The investigators considered the capacity of secondary feed systems, procedures directing the implementation of feed-and-bleed, the number of PORVs and S/RVs for bleed, and the number of high-pressure injection systems for feed. They concluded that the likelihood of losing secondary feed may be different between plants but not enough to make the scenario important at any PWR.

Thermal Hydraulic Scenarios

Four general transient classes were identified: large LOCAs, medium LOCAs, stuck-open primary valves that reclose and cause repressurization, and overcooling of the primary by the secondary. In the previous section, the investigators were concerned with how the frequencies of these scenarios changed from plant to plant. Here, they were concerned with flow rates and coolant temperature.

Mr. William Arcieri, Information Systems Laboratories, stated that one subset of LOCAs includes scenarios where the pumps transitioned through critical flow very quickly. The high-pressure and low-pressure injection systems discharge and cool the primary. The pumps operate at runout or near-runout conditions. The probabilistic fracture mechanics calculations show that temperature-induced failures occur within 10-15 minutes. These well-defined scenarios occurred during pipe breaks of 8 in. or greater.

For pipe breaks of less than 5.7 in, the accumulated discharge is determined by how much flow can pass through the break. Under these conditions, system volume and power level, which are generally directly related, can have an impact on PTS event severity. The five plants considered in the generalization study operate at a higher power level than the three plants, and changes in temperature and pressure should occur more slowly in the five plants.

The events related to stuck-open primary valves that reclose involve a long cooldown transient followed by repressurization to the relief valve setpoint via high-pressure injection, once the valve recloses. The differences noted here involve valve sizes, since plants at a higher power require a higher valve capacity.

The final category was overcooling of the secondary. Issues of consideration include size and location of the secondary break, available flow restrictions, and operator action to isolate secondary systems. These scenarios are bounded by an MSLB, where the primary cools down. As long as some control is maintained over the auxiliary feedwater system, the secondary temperature does not decrease below the saturation temperature of water, 212°F. So, the cooling rate of the primary system is limited.

Mr. Dinsmore stated that the staff made some changes to the content of 50.61a because of the results of the generalization study. Mr. Arcieri said that the Fort Calhoun plant is smaller in terms of power than the other plants considered in the generalization study, but it has larger relief valves. When the investigators discovered that the TWCF increased for this particular situation, they appropriately modified the rule.

Mr. Dinsmore summarized by saying that the PTS technical basis appropriately models the challenge type, frequency, and severity in the study plants. The generalization study showed that the study plants represented the operating fleet well. PRA and TH evaluations were determined to be unnecessary to implement 50.61a.

Member Comments:

Final Rule Language

Member Powers asked whether the staff performed a trade study to determine effective language strategies in developing the rule. Mr. Mitchell stated that no detailed assessment was involved. The decision to structure the rule similar to 10 CFR 50.61 was a regulatory judgment call.

Member Powers expressed concern that the two rules may be confusing to licensees. Mr. Mitchell stated that the steps required to enter under 50.61a will require a cognizant, positive decision-making process to use the alternative rule.

Member Powers asked whether applicants not exceeding the criteria of 10 CFR 50.61 are required to submit an application. Mr. Mitchell stated that they would be required to submit an

application. Licensees that meet the current rule's screening criteria may still apply for the new rule.

Chairman Shack asked whether the schedule of the ASME inspections would be augmented. Mr. Mitchell stated that most licensees would continue doing inspections at the typical 10-year intervals. However, some may request to extend that interval to 20 years, based upon the technical basis of the 50.61a rulemaking project. Licensees seeking this extension must perform a flaw-distribution check.

Chairman Shack, Member Sieber, and Member Armijo expressed concern about detection of flaw sizes that fall below the ASME threshold and about assumptions made in processing ISI data. Mr. Mitchell stated that, while there is a bias toward oversizing flaws, licensees should be able to perform the requested measurements and analyses. In addition, provisions for dealing with NDE uncertainty are included in the rule. Member Sieber asked whether ISI data is of sufficient quality to support the analyses being requested. Mr. Mitchell stated that RES and EPRI's NDE Center have confirmed that current ISI data is adequate for demonstrating compliance with 50.61a. Mr. Kirk stated that experiments involving typical ISI measurements confirmed by destructive metallurgical examinations would be beneficial. Chairman Shack expressed concern about the degree of confidence the staff has in ISI results for accepting applications.

Chairman Shack asked whether the capabilities of ISI would be documented in a NUREG/CR or topical report. Mr. Mitchell stated that such a report has not been requested but is possible.

Member Sieber stated that the size of flaws was more important than the number of flaws. Mr. Kirk stated that both aspects are important because, as the flaw density increases, the probability that a flaw resides in an embrittled portion of the vessel increases.

Chairman Shack suggested that the rule language should specify that the flaw size be determined in inches before the instruction to "divide by 1,000 inches" found in Section (e).

Chairman Shack asked whether staff planned to write a Regulatory Guide on the analysis of flaw distribution. Mr. Mitchell stated that, while staff believes that the rule may be successfully implemented without it, they are aware that some aspects of the rule may benefit from additional guidance.

Member Armijo requested clarification on the analysis required for comparing plate and weld reference temperatures. Mr. Kirk stated that first the fluence value is found for that particular azimuthal location, which is an important distinction from the requirements of the original rule. The transition temperature shifts for the weld and two adjacent plates and the sum of the unirradiated reference temperature and the reference-temperature shifts are calculated for the two plates and the weld. The maximum of these values is then noted. This process is repeated for every axial weld, and the maximum of all these values is chosen as the limiting value.

Chairman Shack asked when the surveillance samples would be consumed. Mr. Mitchell stated that industry has been requested to obtain data at higher fluence levels. Also, staff and industry has discussed strategies for making the best use of the remaining samples for extending the database to high fluence levels.

Member Powers asked whether there is protection against potential surprises in embrittlement trends that may occur at higher fluence levels. Mr. Mitchell stated that there is evidence that

new embrittlement mechanisms may become significant at high fluences. The surveillance evaluations are in place to identify whether new mechanisms are becoming active. Member Powers expressed concern about a unique vessel, the surveillance capsule of which has not been tested. Mr. Mitchell stated that the surveillance measurement is required before reaching high fluence levels. The new statistical tests are developed to help ensure that data sets indeed are consistent with the embrittlement model of the technical basis.

Member Armijo asked whether the surveillance results are biased by the fact that the radiation damage is occurring at a higher rate than what is being experienced by the vessel. Mr. Mitchell stated that this concern could be an issue. The surveillance capsules are irradiated at a rate that is three to five times faster than the vessel wall. Generally, it is believed that these increases are not enough to significantly impact the microstructural evolution. Some vessels have capsules with lead factors close to 1.

Public Comments

Chairman Shack pointed out that the staff could ensure consistency with the technical basis by issuing a Regulatory Guide, as opposed to including the embrittlement model in the rule itself. Mr. Mitchell stated that the technical, regulatory, and legal perspectives all agreed on including the embrittlement model in the rule as a means of providing clarity. Member Armijo observed that the embrittlement model cannot be changed without another rulemaking action. Mr. Mitchell stated that a licensee could apply to use a different model through the 50.12 exemption process. Mr. Geary Mizuno, Office of the General Council, stated that the approach has value because it provides consistency and predictability. A licensee may also submit a petition for rulemaking if the stakeholder feels that the rule should be further developed. Chairman Shack expressed concern that this situation could lead to inconsistency in how the rule is enforced, with several embrittlement models and/or Regulatory Guides being employed. Mr. Mizuno agreed with this comment but stated that the Commission adopted the current approach as a matter of policy.

Member Sieber asked whether any comments on the flaw limits and density changed the basic structure of the rule. Mr. Mitchell stated that the neutron fluence map was moved to Section (e)(6).

In response to a comment from Member Powers, Mr. Mitchell stated that data sets with meaningful information may not be identified if the slope check was not incorporated into the rule.

Member Armijo asked whether the number of measurements from extracting a surveillance capsule was statistically significant. Mr. Mitchell stated that eight or ten Charpy specimens would be tested, but that this procedure results in only one value of the reference temperature shift due to irradiation. Each licensee is expected to have three to five shift values as they approach higher fluence levels, and the statistical tests are designed for that range of observations. Mr. Mitchell stated that a licensee must develop a proposal for calculating the reference temperatures when the Charpy data shows deviation from the embrittlement model. Member Powers observed that this approach requires sacrificing regulatory certainty.

In response to a question from Member Powers, Mr. Mitchell stated that between eight and twelve plants are projected to exceed the screening limits of 50.61, but that number could increase due to power uprates or life beyond 60 years.

Generalization Studies

Member Bley expressed concern about a conditional probability, given a large-break LOCA, that approached 1. Mr. Dinsmore stated that the values close to 1 involved full power years (fpy) well beyond 60 years. The conditional probability of failure at 60 fpy is 7×10^{-5}.

Member Bley stated that multiplying the conditional probability for a through-wall crack by the initiating event frequency would create a convincing argument for whether a plant is protected against a PTS event. Mr. Dinsmore stated that, instead of taking that approach, the staff is determining a reference temperature at the acceptance criterion of TWCF = 1×10^{-6} per year. Based on current fluxes and embrittlement trends, the acceptance criterion would not be met until the vessel has been exposed to radiation for several hundred years.

In response to a concern raised by Member Bley about the PRA, Mr. Kirk stated that operator actions do not affect the results for LOCAs, which account for 75% of the overall TWCF. Only a small portion of the stuck-open valve scenarios (those that initiated at hot zero power) included operator-action credit. In response, Member Bley suggested removing all operator-action credit and observing the effects on the analysis.

Member Armijo asked whether a fast transient consisting of a small amount of depressurization was found. Mr. Kirk stated that the only transient that retains pressure in the primary system is a secondary-side break. The temperature in the primary system does not decrease enough to cause significant embrittlement of the vessel.

Member Banerjee asked about the assertion that the severity of a medium or large LOCA depends only on the vessel conductivity and dimensions. Considerable discussion ensued concerning how well-mixed the coolant is as it enters the downcomer and flows up the vessel. Mr. Kirk pointed out that the thermal plumes only increase the axial stresses, which can influence only the less threatening circumferential flaws. Member Banerjee stated that, if the vessel conductivity is really the important factor that drives LOCA severity, then generalization is straight forward.

Member Banerjee asked why foreign regulators were concerned about thermal plumes. Mr. Kirk stated that their failure metric is initiation of crack growth, as opposed to TWCF. The axial stresses will initiate the growth of the circumferential flaws, but they will not penetrate the surface. Hence, thermal plumes that generate axial stresses are significant to foreign regulators.

Member Bley observed that results of PRAs where two plants with a few differences were compared did not align very well.

Member Bley asked whether the generalization study involved actual calculations or roundtable discussion. Mr. Dinsmore stated that the study involved a questionnaire, where the investigators made careful judgments about what was important for generalization.

Member Bley asked whether the degree of embrittlement of the five plants of the generalization study was included in considering the severity of PTS challenges. Mr. Dinsmore stated that embrittlement was not included, since the conditional probability of failure was calculated at 60 fpy for each plant.

Member Bley asked whether seismic events were ever considered as initiators to medium-break LOCAs. Mr. Dinsmore stated that a report addressing all external events for the five plants of the generalization study was available. Member Bley stated that aspects such as the fragility of pipe supports can be unique from plant to plant. Mr. Dinsmore stated that the investigators performed a bounding analysis based on seismic hazard curves for Diablo Canyon. Member Bley expressed concern that the bounding analysis was not convincing, without making specific observations of plant layouts. Chairman Shack noted that IPEEE guidance allowed seismically-initiated large and medium LOCAs to be screened from the analysis, indicating they were unlikely. Mr. Dinsmore stated that, if the seismic-induced frequencies were the same as the internal-event frequencies, then the investigators would have sufficient reason not to observe each plant. Member Banerjee asked if any PTS events, besides the LOCA, can occur as a result of the earthquake. Mr. Kirk stated that core damage is more likely than fracture of the vessel.

Member Bley observed that very few plants have undergone a fragility analysis. Member Powers said that many fragility guidelines were questioned as a result of a Japanese earthquake.

Member Bley expressed concern that the surveys used to perform the generalization study did not adequately address the manner in which different plants mount valves, especially within the context of a seismic event.

Member Powers asked whether the stuck-open valve scenario led to the exclusion of AP1000 from using 50.61a. Mr. Mitchell stated that the work of ensuring that AP1000 characteristics were consistent with the technical basis of the new rule has not been done. Mr. Kirk and Mr. Mitchell pointed out that, with the new ring-forged vessel with low copper content, the AP1000 would not be in danger of a PTS event. Member Powers pointed out that the vent valve system in the AP1000 design would cause caution in considering the use of 50.61a.

Member Bley asked whether any plant exhibited an increased capability to overfeed than the others. Mr. Arcieri stated that they assumed that operator action protected against overfeeding.

Member Bley expressed a concern regarding the belief that advanced designs will not be susceptible to PTS challenges. Mr. Mitchell responded by saying that the new vessels will consist of single-piece ring forgings with no axial welds and with very low copper content.

Member Armijo asked about how the staff came to agree with the conclusions of the generalization study. Mr. Mitchell stated that the results of the FAVOR calculations convinced the staff that medium and large LOCAs were the dominant transient. Previously, the most threatening LOCA-type PTS event was considered to be a stuck-open valve, since that included repressurization. The more realistic flaw distribution requires more severe loading transients to initiate a crack. Mr. Kirk pointed out that some time is required to have a rulemaking action from technical basis work, since many people in the agency have to become comfortable about the idea. This fact is especially true when the conclusions are different than what was previously widely believed. Ms. Rodriguez said that the rulemaking phase of this project began in 2006-2007, with the Commission authorizing rulemaking in 2007.

Subcommittee Decisions and Actions:

Member Powers asked about the letter for the PTS rulemaking. Chairman Shack stated that the three-plant study has been approved many times by the Committee, but the main issue of

concern presently is whether the generalization study is convincing. Particularly, is it clear that a plant-specific consideration of event frequencies or TH characteristics is not necessary?

Member Bley expressed concern about the generalization study. The narrowness of the important events is promising for generalization. The project hinges on the reliability of the probabilistic fracture mechanics code. On seismic issues, the possibility for odd mountings of pipes and valves was not considered. The consideration of only five plants may not have been sufficient for generalization. Mr. Kirk stated that adopting an approach of developing a very conservative model essentially determines the final result, with the vessel being less resistant to PTS events. Developing a realistic model made generalization easier, since many issues that were once believed to be important were not significant. Mr. Dinsmore stated that the Indian Point plant was successfully evaluated against the generalization study to extend their reactor vessel. Mr. Mitchell stated that no vessel will likely approach the screening limits developed in the rule.

The Members offered suggestions to the staff on preparing for the forthcoming meeting with the full Committee. Following the full Committee meeting, the Committee will write a letter formally stating their position on the PTS rulemaking action. Chairman Shack adjourned the meeting at 5:20 pm.

Background Materials Provided to the Subcommittee Prior to the Meeting:

1. Letter Report, "Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk," December 14, 2004 (ML042880476).
2. NUREG-1809, "Thermal-Hydraulic Evaluations of Pressurized Thermal Shock," (ML050390012).
3. NUREG/CR-6856, "Final Report for the OSU APEX-CE Integral Test Facility," (ML043570405).
4. NUREG-1795, "FAVOR Code Version 2.4 and 3.1 Verification and Validation Summary Report," (ML072820045).
5. **PROPRIETARY** MRP-90, "Materials Reliability Program: Validation and Verification of FAVOR v02.4, PFM Computational Algorithms and Associated Sampled Variables," (ML083080415).
6. MRP-125, "Materials Reliability Program: Validation and Verification of FAVOR, v03.1, PFM Computational Algorithms and Associated Sampled Variables," (www.epri.com).
7. MRP-171, "Materials Reliability Program: Validation and Verification of FAVOR, v04.1, PFM Computational Algorithms and Associated Sampled Variables," (www.epri.com).
8. Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock Risk Results to Additional Plants," December 14, 2004 (ML042880482).
9. NUREG/CR-6859, "PRA Procedures and Uncertainty for PTS Analysis," December 2004 (ML051790414).
10. NUREG-1806, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule," August 2007 (ML072830076).
11. K. Wallin, "The Scatter in K_{Ic} Results," *Engineering Fracture Mechanics*, vol. 19, pp. 1085-1093, 1984.

12. Letter from Graham B. Wallis to Luis A. Reyes, “Pressurized Thermal Shock (PTS) Reevaluation Project: Technical Basis for Revision of the PTS Screening Criterion in the PTS Rule,” March 11, 2005 (ML050730177).

13. Memorandum from Timothy A. Reed to Edwin M. Hackett, ”Advisory Committee on Reactor Safeguards Review of Final Rule Regarding the Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events (10 CFR 50.61a),” February 2, 2009 (ML090330688).

****

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/2009/ or purchase from Neal R. Gross and Co., Inc., (Court Reporters and Transcribers) 1323 Rhode Island Avenue, NW, Washington, DC 20005 (202) 234-4433.