October 26, 2005

MEMORANDUM TO: Daniel S. Collins, Acting Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

/RA/

FROM: Girija Shukla, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: SUMMARY OF MEETING HELD ON OCTOBER 11, 2005, WITH THE WESTINGHOUSE OWNERS GROUP TO DISCUSS RESUBMITTAL OF WCAP-16168-NP, "RISK-INFORMED EXTENSION OF REACTOR VESSEL INSERVICE INSPECTION INTERVAL"

On October 11, 2005, at the request of the Westinghouse Owners Group (WOG), the Nuclear Regulatory Commission (NRC) staff met with representatives of the WOG to discuss resubmittal of WOG Topical Report WCAP-16168.

The meeting began with a presentation from WOG on the background and overview of WCAP-16168, changes made to the NRC Pressurized Thermal Shock (PTS) risk reevaluation, results of the revised analysis, schedule of the planned revision and resubmittal of WCAP-16168, its plant-specific implementations, and one-refueling cycle extensions for plants with near-term needs.

The WOG explained that in revising WCAP-16168, they plan to revise the PTS transients, update the FAVOR interface, revise the PTS screening criteria from RT_{NDT} to RT_{MAX}, incorporate the latest NRC PTS risk reevaluation, and add more details as per their previous discussions with the NRC staff, including the defense-in-depth discussion reflecting Regulatory Guide 1.174 guidelines. The WOG further explained that the planned Revision 1 to WCAP-16168 will provide the technical justification for extending the reactor vessel inservice inspection (ISI) intervals from 10 years to 20 years and will provide the methodology for individual plant implementation based on plant-specific data.

After the WOG presentation, the NRC staff discussed (1) the status of the technical basis that is being prepared by the Office of Nuclear Regulatory Research (RES) for the PTS rulemaking to revise Section 50.61 of Title 10 of the *Code of Federal Regulations*, (2) the Office of Nuclear Reactor Regulation's (NRR's) review of the RES technical basis, (3) NRR's comments to RES, and (4) the expected schedule for RES's resolution of NRR's comments. These NRR comments are provided in the enclosure of this meeting summary. The NRC staff also made a comment that generic Probabilistic Risk Assessment results used in WCAP-16168 may not be applicable to the plant-specific applications. The WOG asked the NRC staff some questions to clarify NRR's comments to RES and the staff provided the WOG with an adequate response. The NRC staff stated that RES may be able to resolve NRR's comments by spring 2006, but the RES technical basis may be published earlier with NRR's comments, and a supplement

may be issued at a later date to publish RES's resolution of NRR's comments. The NRC staff noted that NRR's comments regarding the PTS technical basis may affect the results of the calculations in the WCAP-16168. The NRC staff also noted that if the WOG decides to resubmit WCAP-16168 prior to the resolution of NRR's comments by RES, the WOG would be expected to address NRR's comments as they affect the WCAP-16168 calculations. The NRC staff and the WOG also discussed the need for a fleet-wide integral inspection schedule in conjunction with the application of the WCAP-16168.

The WOG expressed its appreciation to the NRC staff for the meeting and for sharing NRR's comments on the RES technical basis with them. The WOG also stated that it will review these comments and make a decision soon regarding how WCAP-16168 should be revised and when it will be submitted to the NRC staff for review.

At the conclusion of the meeting, the NRC staff expressed its appreciation to the WOG for the meeting. An attendance list is provided in the attachment. There were no members of the public present at the meeting. The WOG slides used during the meeting are available in the Agencywide Documents Access and Management System under accession number ML052870025.

Project No. 694

Attachments: 1. Meeting Attendees
2. NRR's Comments

cc w/atts:
Mr. James A. Gresham, Manager
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MEETING NOTICE: ML052590393 **PKG.: ML052910162**
ADAMS Accession No.: ML052910148 **NRC-001**

OFFICE	PDIV-2/PM	PDIV-2/LA	EMCB/SC	PDIV-2/SC(A)
NAME	GShukla:sp	LFeizollahi	MMitchell	DCollins
DATE	10/26/05	10/26/05	10/26/05	10/26/05

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MEETING ATTENDEES

MEETING WITH THE WESTINGHOUSE OWNERS GROUP TO DISCUSS RESUBMITTAL OF WCAP-16168-NP, "RISK-INFORMED EXTENSION OF REACTOR VESSEL INSERVICE INSPECTION INTERVAL"

OCTOBER 11, 2005

Westinghouse Owners Group

K. Hoffman	Constellation	
M. Richter	Constellation	
M. Pyne	Duke	
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C. Boggess	Westinghouse	
J. Molkenthin	Westinghouse	
N. Palm	Westinghouse	
J. Andrachek	Westinghouse	

NRC

G. Shukla	NRC Project Manager for the WOG/DLPM/NRR
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M. Mitchell	Section Chief, EMCB/DE/NRR
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S. Dinsmore	SPSB/DSSA/NRR
S. Long	SPSB/DSSA/NRR
M. Kirk	MEB/DET/RES

OFFICE OF NUCLEAR REACTOR REGULATION STAFF'S COMMENTS
ON THE TECHNICAL BASIS FOR THE PTS RULEMAKING TO REVISE 10 CFR 50.61
PREPARED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH
AS DISCUSSED IN THE MEETING WITH WOG ON RESUBMITTAL OF WCAP-16168
ON OCTOBER 11, 2005

1. The Office of Nuclear Regulatory Research (RES) documentation of the risk assessments for three plants indicates that the risk spans a range of about a factor of three above and below the average of the three individual plant values at the same value of the nil-ductility transition temperature. The RES documentation attempts to address this by introducing additional factors (the α factors in equations 11-1 and 11-2 of NUREG-1806) that make the computed risk result conservative for all three of the example plants. This approach may not provide reasonable assurance that the risk of pressurized thermal shock at each of the other 67 plants in the industry will be bounded by these equations.

   An alternative approach would be to consider the statistical basis for the application of the RES results to the entire industry. In order to obtain a confidence level of 90 percent or 95 percent that a nil-ductility transition temperature limit based on the results of the three plants bounds the risk for each of the 70 plants, one could represent the probability distribution of the plant risks by a mean and standard deviation derived from the three-plant sample. In simple terms, a point above the mean on the risk probability distribution could be determined such that the probability of drawing 70 samples from the full distribution is 0.90 or 0.95, so that there is 90 percent or 95 percent confidence that all 70 of the real plants will fall below that level of risk at that value of the nil-ductility transition temperature. Please provide an estimate of the effect of this alternative analysis methodology on the screening criteria proposed in the technical basis documents.

2. The Fracture Analysis of Vessels - Oak Ridge (FAVOR) Code uses a Reference Temperature for Nil Ductility Transition (RT_{NDT}) fracture toughness indexing parameter and a Master Curve Approach fracture toughness indexing parameter (T_0) to estimate material toughness properties. The sampling of the RT_{NDT}- T_0 correction parameter in the Monte Carlo process (used in the FAVOR Code), may have an effect on the variation that is seen in the results for the example plants. Currently the correction is sampled inside the flaw loop so that each flaw is potentially assigned a different correction. It is probably more appropriate to sample the correction outside of the flaw loop so that the correction is sampled once for each material for each vessel simulation. There is not much, if any, effect expected on the mean value of the results when all of the plants are averaged. Please provide an estimate of the change in the Through Wall Cracking Frequency (TWCF) distributions when the RT_{NDT}- T_0 correction is re-sampled for each flaw and when it is only re-sampled once per vessel for each material. Please discuss whether the differences in the TWCF distributions, if any, between the two sampling schemes have any effect on the screening criteria proposed in the technical basis documents. Please also evaluate the effect of the differences, if any, with respect to the alternative analysis methodology described in Question 1A.

3. FAVOR Code samples a very uncertain relationship between a plant's current estimate of RT_{NDT} and the T_0 value that is now considered to be the most appropriate way to relate nil-ductility to temperature. Although available data indicates that the relationship is about 65° F conservative on average, the same data demonstrates that individual RT_{NDT} values may be as much as 19° F non-conservative as well as up to 174° F conservative, compared to T_0. What is not known is how a particular plant TWCF would change from the mean prediction used in the technical basis document if that plant had T_0s that were all known to be relatively close to RT_{NDT}. FAVOR samples a number of distributions that may raise similar questions.

    For each of the input parameters that are sampled from a distribution during the FAVOR Monte Carlo analysis, please categorize the distributions as being generic or plant-specific. Please discuss whether the treatment of any parameter should be modified to facilitate the development of the plant-to-plant variability distribution that is needed to obtain the confidence bounds discussed in Question 1.A.

    For parameters whose treatment is determined to be in need of modification, implement necessary changes (if possible).

    If there are any parameters where change is desired but not currently feasible, describe why the change is not feasible and perform a qualitative assessment of the impact on TWCF.

    Use the information derived from the steps above when developing the confidence bounds.

4. There appears to be sufficient information from the integrated test facility results to demonstrate that thermal plumes below the cold-leg nozzles are no stronger than 20° F less than the average downcomer temperature. The effect of a 20° F plume does not seem to have been quantified with respect to its potential effect on TWCF. Please discuss and, to the degree possible, quantify the effect on TWCF of a 20° F plume. Also, if the effect of the plumes is determined to be significant, please indicate, if known, whether any of the more embrittlement-sensitive plants (e.g., the top twenty plants in terms of predicted TWCF) have cold-legs located directly above more than a single axial weld.

5. The comparisons of the Reactor Excursion and Leak Analysis Program (RELAP) results to the data obtained in the thermal-hydraulic test facilities have been made on the basis of the mean difference between the predicted and measured temperatures and the standard deviation of those differences, evaluated over rather long time intervals. This may not always be a valid basis for evaluating the differences in the thermally-induced stresses that would be computed from the RELAP results and the test data.

    Based on review of only the few comparisons of RELAP results to test data that are contained in NUREG-1806, the Multi-Loop Integral System Test (MIST) Test 4100B2 shown in Figure 6.20 (shown on next page) on page 6-123 seems like a possible candidate for significant under-prediction of the thermally-induced

stress. It shows an over-prediction of temperature during a small time segment at a temperature of approximately 170° F. The maximum temperature difference is about 58° F, and it occurs at the point where the time derivative of the temperature is most under-predicted. Consequently, it seems like the stresses predicted from the RELAP results are likely to be less than the stresses predicted from the MIST data at time when the metal temperatures are low enough to create a non-zero probability of brittle fracture. The NUREG-1806 text does note this, but ends with a statement that RELAP over-predicted the downcomer fluid temperature by an average of 0.67° F over the entire test period. Nothing is said about the significance of the period when the temperature is over-predicted and its time derivative is significantly under-predicted. Please discuss, and quantify to the degree possible, the effects on TWCF of the absolute difference in temperature and the difference in the rate of change of temperature between experimental observations and RELAP calculations. Please perform this evaluation for the range of available test data rather than just for this example of MIST test data.

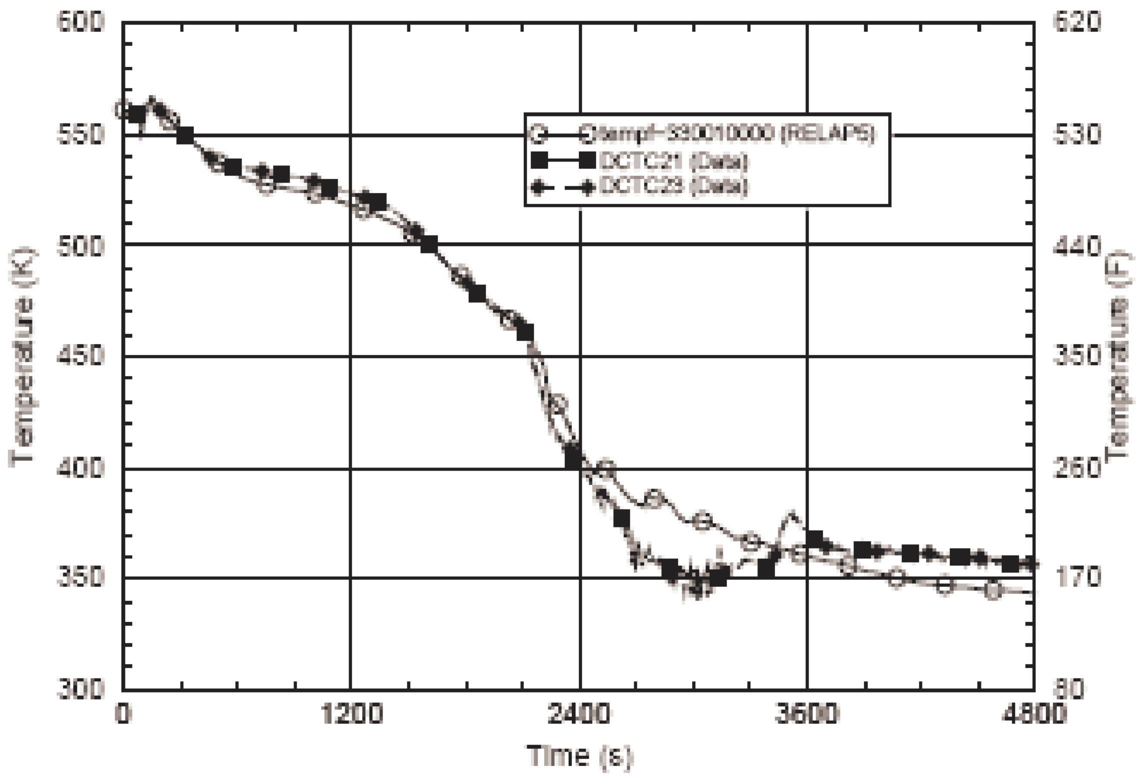

Figure 6.20. Reactor Vessel Downcomer Fluid Temperatures - MIST Test 4100B2.

6. Gamma heating in the vessel wall can affect the shape of the thermal gradient in the vessel during a transient. What are the effects of gamma heating on TWCF?

7. Repair welds seem to have a higher density of larger flaws than Submerged-Arc Welding (SAW) welds. To develop the sample flaw distributions as input to the FAVOR code, Pacific Northwest National Laboratory (PNNL) assumed that 2 percent of the

volume of weld seams consisted of repair welds. The repair welds were assumed to be uniformly distributed through the SAW weld thickness. Since repairs typically intersect the surface, it is possible that flaws associated with repairs would be preferentially located adjacent to surfaces. The extra flaws associated with repairs are typically located at the deepest point of the repair. Examination of the repairs detailed in Section 5.7 of NUREG /CR-6471, Volume 2 indicates the deepest part of the excavation cavity would be more often associated with the surface (or within 2 inches of the surface) than with the interior regions of the plate or weld. Accordingly, it seems reasonable to increase the proportion of the flaw distribution that should be attributed to weld repairs from the current 2 percent to some higher value. The higher value should be associated with the typical area density of weld repair along weld seams. The current approach uses a 2 percent contribution, which was chosen so that it would be a bound to the observed 1.5 percent proportion of weld repair in the Pressure Vessel Research Users Facility vessel. The 1.5 percent seems to have been calculated on a volume basis.

1. What is the proportion of weld repair associated with the weld seams on the PVRUF vessel near the Inside Diameter surface of the vessel on an area rather than a volume basis?

2. What is the expected or calculated effect of this change in the assumptions regarding repair flaw distributions on the TWCFs?

8. Very shallow flaws were created on some forged vessels by underbead cracking that occurred during or following the cladding process. What is the effect of underclad cracks on TWCF?

9. There is a subset of plants that do not have Power Operated Relief Valves (PORVs). In order to ensure the generalization work is applicable to the non-PORV plants, it is important to understand whether there are significant thermal hydraulic differences in the dominant sequences for plants that do not have PORVs. Has this class of plants been analyzed? Do the analyses that have been completed envelope the transients that would occur in a plant that did not have PORVs?

10. The result of the current analysis indicates that a low density of flaws are a primary factor in keeping the risk low. Our state of knowledge of the flaw densities in the 70 individual Pressurized-Water Reactor plants now in service is based primarily on detailed destructive examinations of a small number of welds and plates from 4 vessels (but mostly from 2 vessels), coupled with expert elicitation and physical modeling. Another potential source of information on flaw density is the in-service inspections (ISIs) performed at 10-year intervals on each operating vessel. It would be very helpful if those inspections could provide evidence to support the assumptions in the current analysis. In order to assess the ability of ISI to produce information that could be used to partially validate the flaw distributions used in FAVOR, please provide the following two types of information:

    1. The probability of detection for flaws, parameterized by the depth and length dimensions of the flaw and the depth of the inner edge of the flaw below the vessel surface; and

    2. The importance of a flaw to the FAVOR analysis, parameterized in the same manner.