April 3, 2002

NOTE TO: Dr. Peter Ford, Chairman
Materials & Metallurgy Subcommittee

FROM: August Cronenberg, Senior Staff Engineer

SUBJECT: ANALYSIS OF EDO RESPONSE TO ACRS COMMENTS ON REEVALUATION OF THE TECHNICAL BASIS FOR PRESSURIZED THERMAL SHOCK

The purpose of this note is to forward my analysis of the EDO's March 22, 2002 response to the ACRS letter dated February 14, 2002, concerning staff efforts regarding a reevaluation of the technical basis for assurance of reactor vessel integrity under pressurized thermal shock (PTS) conditions. The EDO's response centered on the primary concerns of the committee, namely:

a) that additional information is needed on operator response to dynamic events for main steamline break (MSLB) scenarios,

b) committee concerns regarding the variance narrowing associated with histogram sampling, and

c) the impact of reactor power level and fuel burnup rates.

The EDO response to Item-(a) states that human reliability analysis (HRA) was considered in plant operator response to MSLB events, using ATHEANA guidelines. The example cited were distractions that might delay mitigative operator response to shut-off of flow to the faulted steam generator for an MSLB event, e.g. blowdown audible distractions, nuisance alarms, equipment outages, and the like. The EDO's letter also noted plant operator training on mitigative procedures. Results of such HRA indicate a high probability of operator success, so that the staff concluded that inadequate operator response to dynamic events can be screened from the PRA considerations of event sequences that could lead to PTS.

The EDO response to Item-(b) states that the staff is assessing the degree to which calculational procedures may contribute to variance narrowing and that alternate numerical techniques will be pursued if variance is underestimated in event frequencies.

With regards to the impact of changes in reactor power and potential changes in fuel burnup rates (Item-c), the EDO response states that these were not explicitly treated in the PTS re-evaluation project. However, the EDO's response notes that an increase in power would be manifested by an increase in neutron flux at the vessel wall and an associated increase in vessel embrittlement. Power uprates approvals require assessments of vessel embrittlement using approved embrittlement correlations. The EDO's response regarding burnup is that its impact would largely be felt a shift in neutron energy spectrum and flux, which are again treated through embrittlement correlations.

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From my view the EDO's response should be acceptable to the Committee. The response adequately addresses the principal concerns of the ACRS raised in its initial review of staff progress related to reevaluation of the technical basis for the PTS screening criteria.

Attachments: EDO response dated March 22, 2002
ACRS Letter dated February 14, 2002

cc: ACRS Members
J. Larkins
S. Bahadur
S. Duriswamy
H. Larson

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 14, 2002

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: REEVALUATION OF THE TECHNICAL BASIS FOR THE PRESSURIZED THERMAL SHOCK RULE

During the 489th meeting of the Advisory Committee on Reactor Safeguards, February 7-8, 2002, we reviewed the methodology and initial results of the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. Our Subcommittee on Materials and Metallurgy also reviewed this matter on January 15-16, 2002. During our reviews, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The PTS Reevaluation Project is extensive and appears to be technically sound.

2. The preliminary results of the analysis of the Oconee Unit 1 reactor pressure vessel indicate that when the current PTS screening criterion is reached, the frequency of throughwall cracking of the vessel would be approximately two orders of magnitude below the acceptance criteria for vessel failure given in Regulatory Guide (RG) 1.154. If the ongoing work demonstrates that such results are characteristic of the fleet of pressurized water reactors (PWRs), then the current PTS screening criterion may be overly conservative.

3. When the factors that have large impacts on the failure frequency of the reactor vessel have been identified, they should be scrutinized appropriately.

BACKGROUND

The PTS Rule, 10 CFR 50.61, was established as an adequate protection rule in 1985 in response to a longstanding design-basis issue concerning the integrity of irradiation embrittled PWR pressure vessels during scenarios in which there is a thermal transient in conjunction with the maintenance of system pressure. The rule specifies numerical values of an end-of-life material toughness parameter (RT_{PTS}). Licensees are required to demonstrate that the material toughness (RT_{NDT}) in their pressure vessels is less than the PTS screening criterion, which depends on the orientation of the crack. The analyses that defined the screening criterion included a number of assumptions that may make the criterion overly conservative. The staff is now reevaluating the degree of conservatism in the technical basis for the screening criterion in the Rule and the associated RG 1.154 acceptance criteria.

Elements of the reevaluation include: (1) a probabilistic risk assessment (PRA) to identify the event sequences that could lead to PTS and then estimate their frequencies; (2) thermal-hydraulic calculations of the pressure, temperature, and heat transfer coefficient in the coolant adjacent to the pressure vessel wall following the various event sequences; and (3) probabilistic fracture mechanics (PFM) estimates of the probabilities of initiating, propagating, and arresting a crack in the pressure vessel for the sets of plant operational and thermal-hydraulic conditions identified in the previous elements. The PFM estimates are calculated using the Fracture Analysis of Vessels - Oak Ridge (FAVOR) code, which is based on earlier Oak Ridge National Laboratory codes; these, in turn, had their foundation in fracture experiments on prototypical pressure vessels started in the 1970s. The current version of the FAVOR code (v01.0) incorporates the probabilistic aspects of the inputs, such as, PRA analysis of operational scenarios and thermal hydraulic, material, and stress conditions, with the output being a calculated distribution of the frequency of throughwall cracking of the vessel. The PTS Reevaluation Project involves the application of this integrated analytical process to four PWRs that reflect a range of designs: Oconee Unit 1, Beaver Valley Unit 1, Palisades, and Calvert Cliffs Unit 1.

In this letter, we comment on the technical progress to date. We do not comment on issues such as external events, containment integrity, and source terms, which are pertinent to potential changes to the throughwall cracking frequency criteria given in RG 1.154 or the PTS screening criterion. These topics will be examined in the future.

DISCUSSION

The PTS Reevaluation Project involves integration of tasks involving PRA, thermal-hydraulics, and PFM including an integrated, quantitative treatment of uncertainty. Overall, the analytical logic and the approach to the physical reality of the technical basis appear to be sound.

The staff has committed to provide us with additional information concerning: how the dynamic events associated with a main steamline break will affect the assumed responses of the operators and the plant; the variance narrowing associated with histogram sampling; and the sensitivity of results to changes in reactor operating power and fuel burnup.

An important aspect of this reevaluation is providing explicit credit for mitigative actions by the operators. The Oconee Unit 1 analysis indicates that some of these actions may have a large impact on the vessel failure frequency. The probabilities of operator failure are evaluated by assessing the relevant performance shaping factors and employing expert judgment. Due to the potential significance of these actions, detailed scrutiny of these probability estimates, including sensitivity studies, alternative human reliability analysis models, and independent peer reviews, should be performed.

There appear to be other factors, such as the spatial and size distribution of flaws, that have a significant impact on the results but have a relatively weak empirical basis. Like the modeling of human error probabilities, these factors should also receive appropriate scrutiny. Prior to completing this Project, it is important to document the validation bases of the relevant codes and databases. We look forward to reviewing further progress.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Kirk, M., NRC, and Williams, P., ORNL, "Recommended Method to Account for Uncertainty in the Fracture Toughness Characterization Used to Re-Evaluate the Pressurized Thermal Shock (PTS) Screening Criterion," revised draft dated October 3, 2001(Draft Predecisional).
2. Williams, P. T. and Dickson, T. L., ORNL, NUREG/CR-xxxx, ORNL/TM-2001-xx, "Fracture Analysis of Vessels - Oak Ridge FAVOR, v01.0, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations," revised draft dated October 15, 2001 (Draft Predecisional).
3. Dickson, T. L. and Williams, P.T., ORNL, NUREG/CR-xxxx, ORNL/TM-2001-55, "Fracture Analysis of Vessels - Oak Ridge FAVOR, v01.0: Computer Code: User's Guide," revised draft dated October 15, 2001 (Draft Predecisional).
4. SECY-01-0185, "Status Report - Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule (10 CFR 50.61)," dated October 5, 2001.
5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," issued January 1987.