POLICY ISSUE
INFORMATION

March 16, 2001 SECY-01-0045

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: STATUS REPORT - REEVALUATION OF THE TECHNICAL BASIS FOR THE PRESSURIZED THERMAL SHOCK RULE (10 CFR 50.61)

PURPOSE:

To provide the first periodic status report on the staff's work to revisit the technical basis of the Pressurized Thermal Shock Rule.

BACKGROUND:

The Pressurized Thermal Shock Rule, 10 CFR 50.61, was established as an adequate protection rule in 1983 in response to an issue concerning the integrity of embrittled pressurized water reactor (PWR) pressure vessels. The staff is now reevaluating the technical basis of this rule to reflect experience with application of the rule and associated regulatory guide, as well as completion of extensive research on key technical issues underlying the rule. Analyses performed as part of this research suggest that conservatism in the rule may be able to be reduced while still providing reasonable assurance of adequate protection to public health and safety.

The staff's reevaluation is described in SECY-00-0140 ("Reevaluation of the Pressurized Thermal Shock Rule (10 CFR 50.61) Screening Criterion"). In that paper, the staff indicated that it would provide updates to the Commission on the progress of its reevaluation of the rule's technical basis. This paper provides the first such status report. More specifically, each of the elements of the staff's program is described below, with a summary provided of the current status and upcoming milestones.

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DISCUSSION:

The staff's work to reevaluate the rule's technical basis includes analyses of pressurized thermal shock (PTS) risk and analyses of the rule's acceptance criterion. The technical analyses are being performed for four PWRs - Oconee-1, Beaver Valley-1, Palisades, and Calvert Cliffs-1,[1] with the principal focus being the estimation of vessel failure frequencies (from PTS-initiated through-wall cracks) and the uncertainties in these frequencies. This work is being conducted in a cooperative fashion with the nuclear industry and has been discussed in a series of public meetings. The industry portion includes, for example, providing probabilistic risk assessments (PRAs) for the four study plants, which the staff reviews and adapts for the PTS risk analyses.

Elements of the Reevaluation

Identify PTS Scenarios and Estimate Their Frequency. This element provides information on the types of scenarios that could lead to PTS-induced reactor pressure vessel (RPV) failures and the frequencies of these scenarios. Potential PTS scenarios are identified by looking for operational or accident situations that can lead to all three of the following conditions: (1) rapid cooling of the RPV, (2) continuation of such cooling to a sufficiently low temperature that parts of the vessel can become brittle, and (3) sustained high reactor coolant system (RCS) pressure or repressurization of the RCS. Such potential situations have been identified by searching operating experience, by reviewing previous PTS risk analyses,[2] and by using a team of plant systems, PRA, and human reliability experts to review plant-specific design features, emergency operating procedures, and operator training practices at the four plants being analyzed. This identification process is significantly more comprehensive than that performed to support the original rule and regulatory guide development.

Initiating event categories that have been identified for analysis in the PTS studies include, for example, a loss of main feedwater (operator actions during recovery can result in overcooling); large and small steamline breaks; loss-of-coolant accidents in which RCS pressure remains high or in which repressurization may occur; and losses of various support systems such as power buses, instrument air, and cooling or service water systems to the extent that unique scenario characteristics and/or challenges to the operators could occur. Such initiating events have been used as the starting point for developing PTS event tree/fault tree models for Oconee. The resulting PTS PRA model has been used to identify and estimate the frequency of roughly 14,500 individual sequences that could contribute to PTS risk. Those sequences have been organized into approximately 30 groups (or "bins") for which thermal hydraulics (TH) calculations have been performed (discussed below).

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[1] These plants are being studied because they reflect a range of designs and have agreed to provide relevant information. These plants do not have an identified PTS concern.

[2] Previous PRAs that have included PTS are the NRC-sponsored studies that supported the development of the original rule and regulatory guide, as well as utility-sponsored studies for Palisades and Calvert Cliffs.

Similarly, a PTS PRA model has been developed for Beaver Valley. Initial estimates of sequence frequencies have been developed, and the event sequence binning process is currently under way. Initial results were provided to the TH and probabilistic fracture mechanics (PFM) teams in January 2001.

Lessons learned from the Oconee and Beaver Valley analyses will be used in the staff's review (and possible modification) of PTS PRAs performed by the Palisades and Calvert Cliffs licensees. Those reviews are scheduled to begin in Spring 2001.[3]

Thermal Hydraulics. The TH element provides the RPV downcomer temperature and pressure boundary conditions required for the fracture mechanics analysis. The transients being analyzed have been selected using the PRA results, discussed above, and a general understanding of the TH response of a plant to a range of transients of possible PTS significance. The TH analyses are carried out in an integrated manner with the PRA and PFM analyses to define the types, probabilities, and consequences of the range of possible PTS transients.

The staff has analyzed over 40 transient scenarios for Oconee using the most current version of the RELAP5/MOD3 code. The calculations are more extensive and more complete than previous PTS work because of significant improvements in NRC's TH analysis tools and capabilities. The scenarios calculated included small-break loss-of-coolant accidents, secondary system breaks, stuck-open valves, steam generator overfeed, and combinations of failures. The frequencies and TH characteristics of these bins were provided to the PFM team in December 2000 for their use in estimating the frequency of through-wall cracks due to PTS.

The second plant to be analyzed is Beaver Valley. Since this plant was not previously analyzed in the context of PTS accidents, a new RELAP5 input model has been prepared. The analyses for this plant are expected to be completed in September 2001.

The third and fourth plants to be analyzed will be Palisades and Calvert Cliffs, two Combustion Engineering plants. The preparation of the Palisades RELAP5 input model began in February, and the set of transient analyses is expected to be complete in Fall 2001. The additional TH analyses that may be necessary for Calvert Cliffs will be evaluated based on design and operations differences between the two plants. Additional TH analyses that may be necessary for Calvert Cliffs are planned to be performed by the end of 2001.

To support the TH analyses, an experimental program is being conducted in the APEX-CE facility at Oregon State University. This facility is scaled to relate the experiments being performed to PTS transients in Combustion Engineering designs. The test program has three

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[3] As discussed in the resources section, the staff's work to date to assess PTS accident scenario frequencies and containment performance has been performed at the Idaho National Engineering and Environmental Laboratory (INEEL). Because of the potential for conflict of interest with the new operating organization at INEEL, this work is being terminated there, and further work will be performed by another contractor. This change will result in additional costs and could result in schedule slippages of up to six months beyond the dates discussed here.

purposes: (1) understand fluid mixing phenomena in a plant geometry, (2) identify plant conditions which would lead to the primary loop stagnation and this would lead to the worst PTS conditions, and (3) provide integral system test data to validate and assess RELAP5 and other codes for PTS analyses.

Probabilistic Fracture Mechanics. The PFM element of the staff's work provides estimates of the probabilities of through-wall cracks for each of the sets of PTS scenarios and TH conditions identified in previous elements.

Significant progress has been made in developing key inputs to the PFM analyses. Notable among these are development of generalized flaw distributions, neutron irradiation embrittlement data bases and embrittlement correlations, assessment of neutron fluences, updated assessment of cleavage fracture initiation (K_{Ic}) and arrest (K_{Ia}) toughness in the ductile-to-brittle fracture-mode transition regime, and refinements in the "FAVOR" PFM computer code. Each of these improvements is discussed in more detail below.

Based on the available non-destructive and destructive examination (NDE/DE) data on fabrication-induced flaws in PWR vessel weld and plate material,[4] an expert judgment process has recently been completed to estimate the uncertainties in flaw densities in weld metal, base-metal (plate and forgings), and cladding materials.

Fluence estimates for each of the four plants have been made using up-to-date plant fuel design and operating history information, along with state-of-the-art fluence calculation methods.

Since the PTS rule was established, a large body of additional embrittlement data has become available, and the understanding of embrittlement mechanisms has advanced. The resulting improved embrittlement correlations have recently been published as a result of the staff's research efforts and will be reflected in the ongoing PTS analyses.

Additional cleavage crack initiation toughness, K_{Ic}, and crack-arrest toughness, K_{Ia}, data (including uncertainties), beyond what has been available in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, K_{Ic} and K_{Ia} curves, have recently been incorporated into this project's analysis.

These advances represent a significant improvement in PFM analyses and a more realistic assessment of the uncertainty in the analyses.

Calculate PTS Through-Wall Crack Frequency. The frequency of a through-wall crack will be estimated in this element. That is, for each plant, distributions of cleavage fracture initiation and RPV failure probabilities (from probabilistic fracture analyses in FAVOR) are integrated with distributions of transient event frequencies (estimated in the PRA and TH elements described above). This approach combines key uncertainties in each part into an assessment of overall uncertainty. The key output from these analyses will be probability distribution functions for

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[4] Pressure Vessels Research User Facility, a vessel from a canceled nuclear power plant, was examined by Pacific Northwest National Laboratory to obtain these data.

through-wall crack frequency (assumed to be equivalent to RPV failure and core damage) for each of the four plants. Initial results for Oconee are expected to be obtained in Summer 2001, for Beaver Valley at the end of 2001, and for Palisades and Calvert Cliffs in the first half of 2002.

The staff's work also includes three elements to address associated regulatory issues. These elements, their current status, and upcoming milestones, are:

Regulatory Issues

Reassess Probabilistic Aspects of PTS Screening Criterion. In parallel with the development of revised technical information on PTS events and their frequencies and consequences, the staff is reassessing the basis for the "acceptable" frequency of such events. A scoping analysis of the impact of such RPV failures on containment performance is also being performed as part of this element.

Recent work in this element has focused on the issue of containment performance during PTS accidents. Two scoping studies have been initiated at the Idaho National Engineering and Environmental Laboratory (INEEL) and at the University of California, Santa Barbara (UCSB).

- The INEEL study[5] is examining the set of issues related to containment performance during a PTS accident using a PRA Level 2 event tree analysis. This analysis systematically identifies the key issues in containment performance, such as the size of the vessel crack, extent of piping movement caused by the accident and its impact on penetration integrity, and availability of pressure suppression systems such as containment sprays. Probabilities will be assigned to the issues, with the overall outcome being an estimate of the probability of containment failure, given a PTS accident.

- The UCSB study is examining in detail one of the key issues in the INEEL study, which is the interrelated effects of pressure-driven RCS blowdown and growth of the initial through-wall crack. That is, once a through-wall crack occurs, the subsequent growth of the crack depends, among other things, on the (decreasing) pressure in the RCS. The final size of the crack is a critical factor in determining to what extent the emergency core cooling systems can accomplish their function.

The next steps in this element are to complete the containment performance analysis and to examine its implications on the acceptable frequency of PTS accidents. This is expected to be completed in Summer 2001.

Re-evaluate PTS Screening Criterion. The staff will develop recommendations for new values of RT_{PTS} (the limiting RPV reference temperature for radiation-induced embrittlement), using the results of the four plant-specific PTS analyses, extension of these analyses to more generic conclusions, and the reassessment of the probabilistic aspects of the screening criterion.

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[5] As discussed in Footnote 3 and the resources section, this work is being terminated at INEEL, with followup work to be performed by another contractor.

This element has not yet been initiated, pending completion of the previous element. It is expected that work in this element will be performed in FY2002.

Propose Technical Basis for Revision to 10 CFR 50.61. The information created and assembled in previous tasks will be integrated into a form that will support, if appropriate, a new version of the rule and regulatory guide. When completed, this material will be provided to the Commission with a recommendation on whether or not to proceed with rulemaking.

This element has not yet been initiated, pending completion of the previous elements. It is expected that work in this element will be performed in FY2002.

It was also noted in SECY-00-0140 that revisiting the technical basis of the PTS Rule provides an early test of the staff's framework for risk-informed changes to Part 50 (documented in SECY-00-0198) in the context of possible modifications of an adequate protection rule. This test of the framework will be initiated in the next several months, in the context of the reevaluation of the screening criterion.

RESOURCES:

The resources for this activity are partially included in the RES budget for FY 2001. However, the staff's work to perform the four plant-specific TH and PFM analyses has required more resources than originally planned. This, and the associated need for additional risk analyses, will be included in the RES FY 2001 midyear request for additional funding. If midyear funds are not approved, the program will be extended in accordance with available resources.

Additional resources are included in the FY 2002 President's Budget, and will be included in the proposed FY 2003 budget. In addition, if rulemaking ensues as a result of this endeavor, RES funds to provide technical support for the rulemaking activities will be included in the FY 2003 budget. Ultimately, if additional resources beyond amounts requested are determined to be needed for FY 2002 or FY 2003, they will be prioritized and requested through the Planning, Budgeting, and Program Management process.

It must also be noted that parts of the staff's work in this project (the estimates of accident scenario frequencies and containment performance) have been performed at INEEL. In January 2001, it was concluded that this work must be terminated because of the potential for conflict of interest introduced by Bechtel's recent assumption of laboratory management. The staff is now proceeding to terminate the project at INEEL and restart the work at another contractor. The cost of this change will be included in the FY2001 mid-year request for additional funds noted above. The overall schedule impact of the change of contractors is not yet clear, but could result in a delay in the project's completion of up to six months. This impact is not expected to have an adverse impact on PTS-related licensing decisions.

COORDINATION:

As noted above, this work is being coordinated with nuclear industry work on related technical subjects. The latest in a series of public meetings on the work was held on February 22 and 23, 2001, at NRC-HQ.

The Office of the Chief Financial Officer has reviewed this paper and has no objections. The staff is providing the Advisory Committee on Reactor Safeguards with periodic briefings on the overall program to revisit the PTS Rule technical basis and the approach being taken with respect to the staff's reassessment of the screening criterion.

/RA/

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SUBJECT: STATUS REPORT - REEVALUATION OF THE TECHNICAL BASIS FOR THE PRESSURIZED THERMAL SHOCK RULE (10 CFR 50.61)

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