POLICY ISSUE
(Information)

October 5, 2001 SECY-01-0185

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 second 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 the associated Regulatory Guide 1.154, as well as to reflect the 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," and in the first periodic status report, SECY-01-0045 "Status Report - Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule." In those papers, 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 second such status report. Each of the elements of the staff's program is described below, with a summary of the current status and upcoming milestones.

CONTACTS: Shah Malik, RES
(301) 415-6007

Ed Hackett, RES
(301) 415-5650

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 cooperatively 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. The staff will review and adapt these PRAs for the PTS risk analyses.

Elements of the Reevaluation

Identify PTS Scenarios and Estimate Their Frequencies. 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 may become susceptible to brittle fracture, and (3) sustained reactor coolant system (RCS) pressure or repressurization of the RCS. Such scenarios 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 a loss of main feedwater, 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 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-1. The resulting PTS PRA model has been used to identify and estimate the frequency of roughly 180,000 individual sequences that could contribute to PTS risk. Those sequences have been organized into approximately 40 groups or "bins" for which thermal hydraulics (TH) calculations have been performed (discussed below).

---

[1] These plants are being studied because they reflect a range of designs and have agreed to provide relevant information.

[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-1.

Similarly, a PTS PRA model has been developed for Beaver Valley-1. Initial estimates of sequence frequencies have been developed, and the event sequence binning process is currently under way. Final model refinements and quantification for both Oconee-1 and Beaver Valley-1 are on schedule for completion by early October 2001.

Lessons learned from the Oconee-1 and Beaver Valley-1 analyses are being used in the staff's review (and possible modification) of PTS PRAs performed by the Palisades and Calvert Cliffs-1 licensees. Those reviews are currently being performed and are scheduled for completion in November 2001.

Thermal Hydraulics. The thermal hydraulic element provides boundary conditions for the fracture mechanics analysis of the reactor vessel. The boundary conditions are obtained from calculations of transients using the most up-to-date version of the RELAP5 code. The information supplied consists of the time history of temperature, pressure, and heat transfer in the downcomer region adjacent to the wall of the reactor vessel. The transients to be analyzed are determined from a review of prior analyses, results of probabilistic risk assessment (frequency of occurrence), human factors considerations, and operator procedures.

The staff has now analyzed approximately 75 sequences for Oconee-1. The Oconee-1 thermal hydraulic analysis is essentially complete with the exception of some work associated with uncertainty evaluation. The transients cover a wide spectrum of initiating events and multiple failures. Types of transients include a spectrum of breaks in the reactor coolant system, steam generator tube rupture, stuck-open power-operated relief valve and safety valve on the pressurizer, stuck-open secondary side valve, steam line break, and steam generator overfeed.

Thermal hydraulic analysis is in progress on Beaver Valley-1. It was necessary to develop a RELAP5 input model for this plant since none existed. This was carried out with the assistance of EPRI-funded work at Westinghouse using the RES H.B. Robinson-1 input model as a basis (the two plants are both 3-loop Westinghouse designs, but differ in many ways). Approximately 35 event sequences have been analyzed thus far. This work was completed in September 2001. Altogether, the number of events calculated with RELAP5 is expected to number approximately 50.

Two Combustion Engineering plants are also being analyzed, Palisades and Calvert Cliffs-1. As with Beaver Valley-1, a RELAP5 input model had to be developed for Palisades; the work was done using an input model provided by Framatome ANF and design information obtained from Palisades. This work is completed and preliminary calculations are under way. Calvert Cliffs-1 calculations have also begun. The analyses of these two plants are expected to be completed by mid-November 2001.

Closely associated with the analyses is an experimental program conducted in the APEX facility at Oregon State University. APEX is an integral test facility originally designed to simulate AP600. RES reconfigured the facility for PTS testing using Palisades as a basis. Thus far, 13 tests have been carried out to provide phenomenological information on PTS and data for assessing RELAP5. In addition, the RES special-purpose PTS code REMIX and the computational fluid dynamics code STAR-CD were used to analyze the fluid flow in the cold leg and downcomer. This program was reviewed recently by the ACRS Thermal Hydraulic Subcommittee, which met at Oregon State University in July 2001.

Probabilistic Fracture Mechanics (PRM). 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 thermal hydraulic 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 (Fracture Analysis of Vessels Oak Ridge) PFM computer code.

These improvements are discussed in more detail below:

Generalized Flaw Distribution. The fabrication flaw distribution and density remains a key driver in the outcome of PFM assessments for PTS. Based on the available non-destructive and destructive examination (NDE/DE) data on fabrication-induced flaws in PWR vessel weld and plate material,[3] an expert judgment process has been completed to estimate the uncertainties in flaw densities in weld metal, base-metal (plate and forgings), and cladding materials. The results of this process are being incorporated into the FAVOR code. The following items describe specific activities related to the flaw distribution:

- An analysis of the base metal flaws in PVRUF base metal has been conducted and a validated size distribution was achieved for PVRUF plate. Activities are planned for September through December 2001 to include the validation of Hope Creek Unit II and the River Bend Unit II base metal flaws.

- Cladding flaws have been shown to form on the fusion line of adjacent clad welding passes. These fusion line flaws can become long, and inspection data have shown them to be up to 10 cm in length. Experimental evidence supports both of these findings for strip clad and multi-wire clad. These data are also being reflected in the revision to the FAVOR code.

- A two-day public meeting was held at the NRC on May 9 and 10, 2001, to make a detailed presentation on NRC-funded research in developing the empirical data bases, the work in support of the expert judgment process, and to explain how the flaw distribution and density data are being integrated and transformed into input files for the FAVOR code. The workshop was useful in that it focused on this single topic and led the participants through the process systematically.

- Flaw-related inputs for the FAVOR code have been revised based on new data from recent examinations of vessel material, recommendations from the May 2001 NRC industry workshop, and improved estimation approaches developed to address issues identified during the May workshop. New data on observed flaws in plate materials have also justified reductions in flaw frequencies for base metal regions.

Neutron Fluence. Fluence estimates in the beltline region of the RPV for each of the four plants have been made using up-to-date plant fuel design and operating history information, along with

---

[3] Pressure Vessel Research User Facility (PVRUF), a vessel from a canceled nuclear power plant, was examined by Pacific Northwest National Laboratory to obtain these data.

state-of-the-art fluence calculation methods. Uncertainties associated with the calculated fluences have also been estimated using comparable methods.

Fracture Toughness Models and Embrittlement Correlations. Since March 2001, both the fracture toughness models and the irradiation embrittlement correlation that will be used in the FAVOR code have been finalized and reviewed by a working group that included both NRC and industry representatives. These models use a physical understanding of the fracture process and embrittlement mechanisms both to establish the functional form of the models and to classify uncertainty type (i.e., aleatory vs. epistemic). This understanding is applied with empirical data on the toughness and irradiation sensitivity of RPV steels to develop quantitative models. In August 2001, a program specification detailing these models for coding into FAVOR was finalized. Reports that fully detail the background on the recommended models are currently being prepared.

FAVOR Code Development. At ORNL the FAVOR code is undergoing its last major revision before being used to perform PFM calculations. Updated models of fracture toughness, irradiation effects, and flaws are all being included. Additionally, supporting documentation (theory and users manuals) is being developed. This version of FAVOR will be presented at a public meeting in October 2001.

FAVOR Code Verification and Validation (V&V). A V&V plan was developed and is being executed in conjunction with an activity by the Electric Power Research Institute and the Materials Reliability Project (EPRI/MRP) to verify and validate the FAVOR code for use in reevaluating the technical basis for the PTS Rule, 10 CFR 50.61. This activity will continue in parallel with the application of the FAVOR code in the PTS Rule technical basis reevaluation, with an understanding that significant prior peer review has been incorporated in developing various new models that are being implemented in the FAVOR code. In addition, each major portion of the code has been tested at ORNL per NRC/DOE Management Directive 11.7, Part XI, which specifies software quality assurance guidelines in accordance with NUREG/BR-0167, "Software Quality Assurance Program and Guidelines." Hence, the staff has a high level of confidence in the successful completion of the V&V effort.

Calculation of 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 through-wall crack frequency (assumed to be equivalent to RPV failure and core damage) for each of the four plants. Initial results for Oconee-1 and Beaver Valley-1 are expected to be obtained in Fall 2001, and for Palisades and Calvert Cliffs-1 in Winter 2002.

Regulatory Issues[4]

As indicated in SECY-00-0140, the PTS technical basis re-evaluation effort may lead to the development of a proposed revision to the PTS rule and the associated regulatory guide. The staff has therefore initiated an activity to identify and evaluate viable options for changes to the rule and regulatory guide. Three key elements, their current status, and upcoming milestones in this activity are as follows:

- Reassess Probabilistic Aspects of PTS Risk Acceptance 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. This reassessment involves an evaluation of the current PTS rule in light of NRC’s more recent risk-informed efforts. It will also include a scoping assessment of the impact of PTS-induced RPV failures on containment performance, since Large Early Release Frequency (LERF) is a key parameter in a number of risk-informed regulatory applications.

  The staff is developing a Commission paper that will identify and perform a preliminary evaluation of a number of options for potential changes to the form of the rule, the criteria embedded in the rule, and the regulatory guide. This Commission paper will provide a recommendation regarding the options that will be addressed in a detailed evaluation (as a part of the technical basis development for a potential revision to the PTS Rule). The paper is currently scheduled for completion in Winter 2002.

- Re-evaluate PTS Screening Criterion. RT_{PTS} is the limiting RPV reference temperature for radiation-induced embrittlement. When it is projected that a plant could exceed this limit in its operating lifetime, 10 CFR 50.61 currently requires the submission of analyses/modifications that would justify continued operation, or implementation of thermal annealing to recover RPV fracture toughness. The staff will assess the risk implications of the current RT_{PTS} values specified in the PTS Rule. This assessment will be based on the results of the four plant-specific PTS analyses and the extension of these analyses to more generic situations. For rule change options that involve proposed changes to RT_{PTS}, the staff will also develop recommendations for new values of RT_{PTS}. This element has not been initiated pending completion of the previous element. It is expected that work in this element will be performed in Winter 2002.

- 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 a revision of Regulatory Guide 1.154. When completed, this material will be provided to the Commission with a recommendation on whether or not to proceed with rulemaking.

---

[4] This section, which is concerned with identifying and evaluating options for potential changes to the structure as well as key parameters in 10 CFR 50.61 and Regulatory Guide 1.154 (plant-specific PTS analysis), reflect staff discussions on the desired output of the PTS reevaluation project that have occurred since the previous update (SECY-01-0045).

- The overall effort on reevaluation of the technical basis for the PTS Rule is currently scheduled for completion in Summer 2002.

RESOURCES:

Resources for this activity are included in the RES budget for FY2001 and FY2002. Resources are also included in the proposed FY2003 budget to address rulemaking activities. Key staff on this project are also involved in activities related to recent national events and the impact of this will be assessed in the next paper.

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 May 9 and 10, 2001, at NRC-HQ. The next public meeting is scheduled for October 9-11, 2001.

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

/RA/

William D. Travers
Executive Director
for Operations