December 31, 2002

MEMORANDUM TO: Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

FROM: Ashok C. Thadani, Director **/RA/**
Office of Nuclear Regulatory Research

SUBJECT: TECHNICAL BASIS FOR REVISION OF THE PRESSURIZED THERMAL SHOCK (PTS) SCREENING CRITERIA IN THE PTS RULE (10 CFR 50.61)

This memorandum transmits the results of a multi-year study conducted by the Office of Nuclear Regulatory Research re-evaluating the technical basis of 10 CFR 50.61, the pressurized thermal shock (PTS) rule. This study provides a basis to support rulemaking to revise the PTS rule. Confirmatory research activities will be completed to support the rulemaking process.

A pressurized thermal shock transient involves severe cooling of the reactor pressure vessel (RPV) beltline region together with, or followed by, pressurization. Such transients can present a significant challenge to the structural integrity of the RPV, particularly in the case of an RPV that has been significantly embrittled by neutron irradiation. Compliance with the PTS rule could become a life-limiting factor for certain PWRs. Since the rule was initially promulgated, we have gained significant additional knowledge in the technical disciplines of probabilistic risk assessment (PRA), thermal-hydraulic behavior and probabilistic fracture mechanics. The staff's study has incorporated state-of-the-art advancements in each of these disciplines which has enabled a realistic, risk-informed treatment of the subject.

The study has directly supported the goals of reducing unnecessary regulatory burden and improving efficiency, effectiveness, and realism, while maintaining safety. Since PTS analyses are very complex, drawing on event characterization, event frequency assessments, thermal-hydraulic analyses, materials analyses, and risk considerations, we have also undertaken a comprehensive peer review of the study. The staff informed the Commission of the approach to this study in SECY-00-0140 and has kept the Commission informed on progress in SECY-01-0045, SECY-01-0185 and SECY-02-0092.

The technical basis for the existing PTS rule was documented in SECY-82-465. The subsequent promulgation of 10 CFR 50.61, and the associated Regulatory Guide 1.154, provided a technically defensible basis for assuring a high level of reliability for the reactor pressure vessel, consistent with its unique role in the reactor system.

Research efforts addressing RPV integrity following promulgation of the PTS rule continued to improve the underlying analysis methods and data needed to evaluate the risk due to PTS. By early 1999 it appeared there was a sufficient basis to undertake a study directed to re-evaluating the technical basis for the rule. The PTS study was undertaken consistent with Commission policy guidance on PRA and became a major risk-informed initiative which began in advance of the other Option 3 efforts, although the approach used is consistent with the Option 3 framework. In a user need memorandum (NRR-2002-025) you requested that related activities be completed.

After consulting with your staff at the inception of this study, RES staff approached EPRI's Materials Reliability Program (MRP) to seek their cooperation in developing the plant-specific information that would be needed to carry out the analyses. The industry group agreed to support the effort and budgeted resources to interact with the RES staff in reviewing the technology to be implemented and in providing significant plant-specific information for the Beaver Valley-2, Oconee-1, Palisades, and Calvert Cliffs-1 Pressurized Water Reactors. It should be emphasized that these plants participated in the study voluntarily and that their participation does not connote any unique risk associated with PTS. The industry's active participation was instrumental in developing models and collecting the plant-specific information needed to complete the study.

The approach used in this study was to assess the risk associated with pressurized thermal shock of an embrittled RPV. This study is significantly more comprehensive than the previous study conducted in the early 1980s. As noted above, the study has incorporated state-of-the-art advancements as well as developed new data and approaches in all three major technical disciplines associated with the study. For example, the plant PRAs included extensive modeling of operator actions which, in part, was based on simulator exercises. PTS- specific thermal-hydraulic (T/H) tests were conducted at the Oregon State University which provided first-of-a-kind data to assess T/H codes. Expert elicitation and destructive and non-destructive examinations of an unused vessel were carried out to provide more robust flaw distribution for use in the probabilistic fracture mechanics analyses. These developments were key to the success of this study. The study also explicitly addressed uncertainties and has produced estimates of the mean value and full probability distribution of through-wall cracking frequencies for the RPV.

Based on the detailed evaluations for the Beaver Valley, Oconee, and Palisades plants, a sufficient technical basis exists to undertake a significant revision to 10 CFR 50.61. For the three plants evaluated thus far, the mean frequency of through-wall cracking due to PTS has remained significantly below 1×10^{-6}/reactor year. This indicates that, even with consideration of uncertainties, there would be a significant margin to RPV failure from a PTS event. In terms of assessing the risk due to PTS, even if we assume a through-wall crack results in core damage and large early release, the resulting CDF and LERF values would be very low. The study includes a scoping level review of the margin between RPV failure, core damage and large early release.

As discussed below, In terms of finalizing this study, there are four confirmatory efforts that will be completed to support rulemaking activities:

(1) Generalization of Results

As noted, we have analyzed three of the four plants that participated in the effort. We are using the Calvert Cliffs evaluation to provide additional confirmation of the generalization of the results. While we expect the results will be similar, completing this detailed study will provide an important consistency check on the methodology.

Additionally, we are evaluating system similarities from other selected PWRs to determine if there are any plant-specific characteristics that could create a more severe overcooling transient than those considered in the other four plant studies. In this way, we are developing a basis for generalizing the results of four detailed studies to the entire fleet of PWRs.

(2) Post-Vessel Rupture

The detailed analyses summarized in the report estimate the frequency of through-wall cracking to be well below 1×10^{-6}/reactor year. Phenomenological analyses to extend this result to CDF, LERF, and the potential for air oxidation of the core in the unlikely event of a catastrophic failure of the RPV, are extremely complicated and highly uncertain. The ACRS has specifically raised concerns about the potential change in source term if large-scale air oxidation were to occur following RPV failure. In the attached draft report we have qualitatively addressed each of these issues. Due to the low estimates of RPV failure frequency from PTS, we believe that the consequent CDF and LERF risks are sufficiently low to preclude the need for extensive analyses of containment performance.

(3) Peer Review

The PTS analyses are complex, drawing on event characterization, event frequency assessments, thermal-hydraulic analyses, materials analyses, and risk considerations. We have benefitted considerably from regular interactions with the ACRS, acting in a *de facto* peer review role. Additionally, as we began the study, we interacted with industry representatives in a series of public meetings to evaluate various models and identify appropriate input information. The industry has performed a limited peer review of the probabilistic fracture mechanics code used in the study. However, to preclude conflict of interest issues, we have avoided a substantive involvement with the industry as we finalized the models and undertook the specific analyses. The exception is that the PRA analysts from the individual plants were involved in reviewing the PRA models we used in this study.

An important confirmatory step in finalizing the study is to implement a formal peer review of the models, data, and results. We will initiate this peer review in early CY 2003. We also will be interacting with the international community to describe our study and the results, and to gain their insights and comments. We also anticipate discussing the results of this study with the PRA Steering Committee.

(4) Sensitivity Studies

In addition to the above, the staff is currently evaluating the need for additional or enhanced sensitivity studies to assess the importance of potential variations in key input parameters and modeling assumptions. For example, while RES has performed significant experimental and predictive work in the area of RPV weld flaw distributions, questions remain regarding extrapolating this work to cover all U.S. reactor vessels. This is an area where the staff will be evaluating the sensitivity of the overall PTS risk assessment to flaw inputs other than the generalized flaw distribution.

In closing, I believe that the attached report reflects a comprehensive state-of-the-art study of RPV integrity when subjected to PTS transients and provides the technical basis to support a significant change to the PTS rule that would effectively relax the PTS screening criteria in 10 CFR 50.61 while still maintaining the potential for pressure vessel failure at a very low likelihood level. Such a change could be implemented without necessitating new or additional material characterization testing. However, any such change should be made at the same time the embrittlement estimation procedures of Regulatory Guide 1.99 are revised. This will avoid the regulatory conflicts that were introduced when earlier changes to the embrittlement criteria were implemented without consistent changes in the regulation. I also note that a major change in 10 CFR 50.61 could influence implementation of the pressure-temperature limit requirements in 10 CFR 50, Appendix G. This is an issue that has previously been identified by staff in RES and NRR, and is part of NRR user request NRR-2002-025.

During the course of this study, we have regularly involved your staff. We also have routinely briefed the Advisory Committee on Reactor Safety (ACRS) and the cognizant subcommittees. We have benefitted from staff and ACRS input. We look forward to interacting with you and your staff as you undertake rulemaking to change 10 CFR 50.61. We would welcome comments on the attached draft report. Such comments would be most useful if received by February 28, 2003. The technical contact for this study is Mark Kirk of the Materials Engineering Branch, Division of Engineering Technology at 415-6015, e-mail: mtk@nrc.gov.

Attachment: As stated

cc: W. Travers
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