Latest results of the United States Nuclear Regulatory Commission Pressurized Thermal Shock Rule Re-evaluation Project[1]

T.L. Dickson, [1)] M.T. Kirk, [2)] C.G. Santos [2)]

1) Computational Science and Engineering Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee, USA
2) Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC

ABSTRACT

The current federal regulations to insure that nuclear reactor pressure vessels (RPVs) maintain their structural integrity when subjected to transient loading conditions such as pressurized thermal shock (PTS) events were derived from computational models developed in the early-mid 1980s. Since that time, there have been advancements in relevant technologies associated with the modeling of PTS events that impact RPV integrity assessment. These updated computational models have been implemented into the FAVOR (**F**racture **A**nalysis of **V**essels: **O**ak **R**idge) computer code. An objective of the United States Nuclear Regulatory Commission (USNRC) PTS rule re-evaluation project is to determine if application of improved technology can provide a technical basis to reduce the conservatism in the current regulations while continuing to provide reasonable assurance of adequate protection to public health and safety. A relaxation of PTS regulations could have profound implications for plant license renewal considerations. As part of the re-evaluation project, the FAVOR code has been applied to three domestic commercial pressurized water reactors (PWRs). This paper discusses the application of the updated integrated computational methodology to the three PWRs and discusses the results and interpretation of those results.

PROBLEM DEFINITION, CURRENT REGULATIONS, AND PTS ANALYSIS RESULTS

The issue of pressurized thermal shock (PTS) arises because cumulative neutron irradiation exposure makes the reactor pressure vessel (RPV) more brittle (i.e., reduced ductility and fracture toughness) and, therefore, increasingly susceptible to cleavage (brittle) fracture over its operating life. The degree of embrittlement of RPV steel is quantified by changes in the reference nil-ductility transition temperature, RT_{NDT}. The radiation-induced shift in RT_{NDT} is a function of the chemical composition of the

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steel, the neutron irradiation exposure, RPV operating temperature, and the initial unirradiated transition temperature, RT_{NDTo}.

In PWRs, transients can occur that result in severe overcooling (thermal shock) of the RPV concurrent with or followed by high repressurization. Thermal transients in which the temperature of the coolant in contact with the inner surface of the RPV decreases with time results in time-dependent gradients and resulting tensile stresses across the RPV wall. Both pressure and thermal stresses are tensile stresses that tend to open existing cracks located on or near the inner surface of RPV. If an aging RPV is subjected to a PTS event, flaws on or near the inner surface could potentially initiate in cleavage fracture and propagate through the RPV wall, thus introducing the possibility of RPV failure.

In the early to mid 1980's, working from the state-of-the-art technology at that time, the nuclear industry, the NRC staff, and others performed a number of investigations aimed at assessing the risk imposed by PTS. These efforts led to the publication by the staff of SECY 82-465 [1], which provided the technical basis for what has come to be known as the PTS rule [2]. Results from these analyses, combined with the judgement that a 5×10^{-6} yearly probability of developing a through-wall crack due to PTS is acceptable, led to the establishment of the current screening limits or maximum values of RT_{NDT} permitted during the operating life of the plant. The maximum values of RT_{NDT} allowed by the current PTS rule are 270 °F for axial welds, plates, and forgings and 300 F for circumferential welds. The PTS rule requires plants that desire to operate beyond the screening criteria to submit an integrated plant-specific safety analysis to the NRC three years before the PTS screening limit is anticipated to be reached for any material in the RPV beltline. The NRC subsequently developed Regulatory Guide 1.154 [3] regarding the format and content of analyses that could be used to demonstrate the continued safe operation of RPVs that exceeding the PTS screening criterion.

Figure 1 is a curve from SECY 82-465, from which the current PTS screening criteria was derived, which established a correspondence between a value of RT_{PTS} of 270 °F and a thru-wall crack frequency of 5×10^{-6} failed RPVs per year. Here, RT_{PTS} is the metric currently used to quantify the embrittlement of the RPV and is defined as the maximum value of RT_{NDT} in the entire RPV beltline plus a margin term to account for uncertainties. Figure 1 also includes the results of PTS analyses recently performed for three commercial PWRs as part of the PTS re-evaluation study, i.e., the mean value of the thru-wall crack frequency (TWCF) as a function of increasing RT_{PTS}. These TWCF are considerably less than those generated in SECY 82-465. Specifically, figure 1 illustrates that the values of TWCF corresponding to the current screening criteria of 270 °F are approximately 7.0e-09, 7.0e-08, and 5.0e-07 for the Beaver Valley, Oconee, and Palisades plants, respectively. The results of these analyses will be discussed in more detail below.

When a flaw is predicted to initiate in cleavage fracture, it has two possible outcomes during the duration of the transient. It either propagates through the RPV wall thickness causing RPV failure, or it experiences a stable arrest at a location in the wall. In either case, the advancement of the crack tip through the RPV wall may involve a sequence of *initiation* / *arrest* / *reinitiation* events. The failure criterion used in the analyses discussed in this paper are that a RPV is considered as failed if a flaw propagated through 90 per cent of the wall thickness or fails due to plastic instability of the remaining ligament.

OVERVIEW OF PTS RE-EVALUATION PROJECT

Preliminary studies [4] performed in 1999 suggested that the application of improved technology could reduce the conservatism in the current PTS regulations while continuing to provide reasonable assurance of adequate protection to public health and safety. Based on the above, in 1999, the USNRC initiated a comprehensive project, with the nuclear power industry as a participant, to develop an improved integrated computational methodology that eliminated known conservatisms and provides a more realistic characterization of risk associated with PTS. Figure 2 illustrates elements of the improved fracture-related technology that have been integrated into the FAVOR computer code [5].

Analogous to the Integrated Pressurized Thermal Shock studies [6-8] performed in the 1980s, the PTS Re-evaluation Project plan included the analysis of a number of commercial PWRs with the objective to cover the range of various system designs, operational procedures, and training programs. To date, analyses have been performed for the Westinghouse-designed Beaver Valley plant, the Babcock and Wilcox designed Oconee plant, and the Combustion Engineering designed Palisades plants. Additional analyses will be performed as necessary to insure that any revision to the PTS rule may be applied generically to all commercial PWRs. The PTS analysis results presented in this paper were generated by the 02.4 version of FAVOR.

DETAILED NEUTRON FLUENCE MAPS AND RVID

In the PTS analyses performed in SECY 82-465, from which the current PTS regulations were derived, a single value of neutron fluence and chemistry were assumed to apply to the entire RPV beltline, which conservatively assumes that all of the RPV materials were made of the most brittle of its constituent material.

FAVOR utilizes a methodology that allows the RPV beltline (defined to include the RPV material from one foot below the active core to one foot above the active core) to be divided into major regions such as axial welds, circumferential welds, and plate regions. These major regions may be further discretized into subregions to accommodate very detailed neutron fluence maps provided by Brookhaven National Laboratory (BNL) that include the azimuthal and axial variations in neutron fluence. Detailed neutron fluence maps were provided for each of the three plants corresponding to 32 and 40 effective full power years (EFPY).

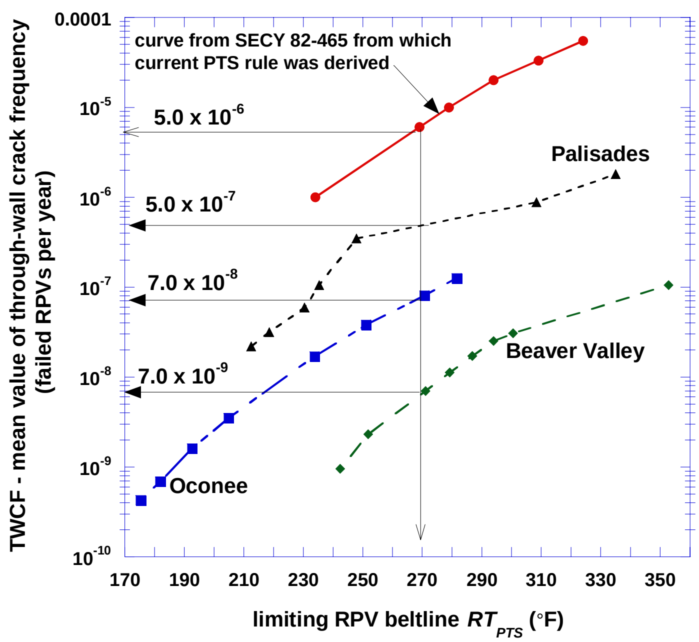

Figure 1 – Results of current PTS re-evaluation program compared to results from SECY 82-465 from which the current PTS screening criteria was derived in early 1980s

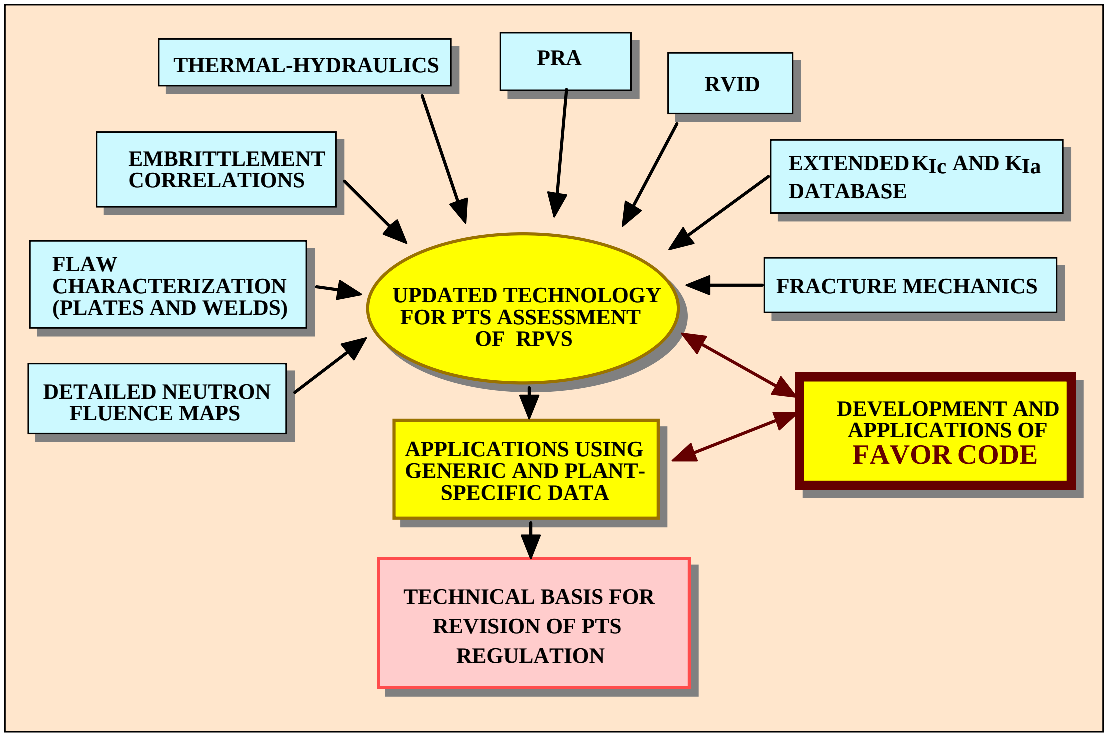

Figure 2 – Elements of improved fracture-related technology are integrated through the application of the FAVOR computer code which has been applied in the USNRC PTS re-evaluation program.

As an example, the fluence map provided by BNL for the Oconee plant contained 13,080 discrete values of neutron fluence (60 azimuthal x 218 axial) corresponding to one-eight of the RPV azimuth (45 degree sector); therefore, the entire 360 degree beltline region would have to be discretized into 104,640 subregions to accommodate the level of detail provided by the BNL. In practice, one may take advantage of symmetry to include this amount of detail with a considerably smaller number of subregions. The discretization used to model the level of detail of neutron fluence for Oconee, Beaver Valley, and Palisades was 19651, 15280, and 67076 subregions.

The modeling and procedures used in generating these neutron fluence maps were based on the guidance provided in the NRC Draft Regulatory Guide DG-1053 [9]. The calculations were performed using the DORT discrete ordinates transport code [10] and the BUGLE-93 [11] forty-seven neutron group ENDF/B-VI nuclear cross sections and fission spectra.

The Reactor Vessel Integrity Database, RVID [12], developed by the NRC, provides a key source of input data for FAVOR. RVID summarizes the properties for the RPV beltline materials for each operating commercial nuclear power plant. The chemistries (copper, nickel, and phosphorus) and values of RT_{NDTo} (initial unirradiated values of RT_{NDT}) for each of the RPV major regions used as input to FAVOR were taken from RVID.

IMPROVED FLAW CHARACTERIZATION

The single largest improvement in the updated computational methodology is the establishment of a technical basis for the postulation of flaws. The probabilistic fracture mechanics (PFM) model utilized in SECY 82-465 conservatively postulated all fabrication flaws to be inner-surface breaking flaws. Specifically, the model assumed the RPV had six axially oriented infinite length inner-surface breaking flaws with their size distributed according to the OCTAVIA distribution.

The USNRC-sponsored research at Pacific Northwest National Laboratory (PNNL) has resulted in the postulation of fabrication flaws based on the non-destructive and destructive examination of actual RPV material. Such measurements have been used to develop characterizations of the number, size, and location of flaws in various types of weld and base metal used to fabricate vessels. This has provided a technical basis for the flaw data which is critical input data into FAVOR. These measurements have been supplemented by expert elicitation [13]. Separate distributions have been developed to characterize the number and size of flaws in weld and plate regions. The reader is referred to References 14-16 for the details of this research. The flaw depth distributions used in these analyses were truncated at 23 per cent and 5 per cent of the RPV wall thickness for flaws in weld and plate regions, respectively.

A major result of the PNNL flaw characterization research is that RPV material has a much higher density of flaws than was postulated in the model utilized in the PFM analyses from which the current PTS regulations were derived; however, all of the flaws detected thus far have been embedded. Application of the improved PNNL flaw characterization results in an average of 7937, 4848, and 5772 postulated flaws for Oconee, Beaver Valley, and Palisades, respectively, in each stochastically generated RPV modeled in the Monte Carlo PFM analysis. Nearly all of the postulated flaws are embedded flaws distributed uniformly through the first 3/8 of the RPV wall thickness.

During the Monte Carlo PFM analysis, the fracture response of each postulated flaw in a RPV is assumed to be physically and statistically independent of all other flaws postulated in that RPV.

IMPROVED EMBRITTLEMENT CORRELATION

Embrittlement correlations are used to predict the neutron-irradiation-induced increase in embrittlement, as characterized by a shift in the nil-ductility transition temperature (ΔRT_{NDT}), over the operating life of the RPV. The current regulatory correlation, described in Regulatory Guide 1.99, Revision 2 [17] is based on analysis of Charpy V-notch impact-energy test data available in 1984. Since then, a significantly larger body of Charpy surveillance data has become available, and the understanding of embrittlement mechanisms has advanced. The result is that improved embrittlement correlations have recently been developed by research sponsored by the NRC. These improved correlations have been published by researchers from Modeling and Computing Services and the University of California at Santa Barbara [18].

IMPROVED THERMAL HYDRAULICS AND PRA

The PTS analyses performed in SECY 82-465 were based on a highly simplified treatment of plant transients (very coarse groupings of many operational sequences), necessitated by limitations in computational resources required to perform multiple thermal hydraulic and PFM analyses. Also, no significant credit was given for operator action.

For the current PTS analyses, a comprehensive search was performed for each plant for transients that are both probabilistically credible and physically significant. The base case transients were identified by development of a PRA model for each plant that addressed possible over-cooling transients that could occur. The identification of the possible over-cooling scenarios for each plant consisted of a review of the current plant design, recent operating history, latest procedures, present-day operator training, and feedback from the ongoing thermal-hydraulic analyses. The PRA model development for each plant involved two visits to the plant, where plant staff input was obtained and plant staff comments on the PRA, including the human reliability assessments, were received and incorporated. Additionally, during the first visit, over-cooling events were simulated on the plant simulator to gain insights about operator responses to such events.

Sandia National Laboratory generated a probability distribution of the transient frequency (events per reactor operating year) for each of the base case transients. The SAPHIRE Version 7 [19] code was used to generate the probability distributions. Hence, the scenarios modeled in the PRA and quantified using the SAPHIRE code, included a cooperative effort with plant staff to ensure that the model reflected current design and practices, and appropriate estimates for the means and uncertainties for the initiator frequencies as well as equipment and human failure probabilities.

Information Sciences Laboratories (ISL) provided thermal hydraulic boundary conditions for the 55, 61, and 29 base case transients for Oconee, Beaver Valley, and Palisades, respectively. The selection of these transients was based not only on thermal hydraulic and anticipated fracture mechanics considerations, but also on PRA input regarding scenarios of sufficient likelihood to be of potential concern to PTS. The thermal-hydraulic calculations were performed using the RELAP5/MOD3 code [20].

IMPROVED FRACTURE TOUGHNESS MODEL

The computational model for quantification of fracture toughness uncertainty is improved in two ways: (1) the fracture initiation toughness (K_{Ic}) and crack arrest (K_{Ia}) databases were extended by 83 and 62 data values, respectively, relative to the databases in the EPRI report [21] and (2) the statistical representations for K_{Ic} and K_{Ia} were derived through the application of rigorous mathematical procedures. Bowman and Williams [22] provide details regarding the data and mathematical procedures. A Weibull distribution, in which the parameters were calculated by the *Method of Moments* point-estimation technique, forms the basis for the new K_{Ic} statistical model. The new K_{Ia} model is based on a log-normal distribution.

IMPROVED FRACTURE MECHANICS

One particularly significant improvement in fracture mechanics modeling capability is the addition of the phenomena known as warm-prestress (WPS) which is now available in FAVOR as a user-option. The concept of the WPS effect is that a crack tip cannot initiate in cleavage fracture in a stress field that is decreasing with respect to time. The results reported in this paper for Oconee, Beaver Valley, and Palisades included warm-prestress. The analyses performed in SECY 82-465, from which the current PTS screening criteria was derived, did not include WPS. A limited number of sensitivity analyses were performed for Oconee and Beaver Valley to quantify the impact of WPS on TWCF. The impact of the inclusion of WPS was to reduce the TWCF by between a factor of 2 and 5. The impact decreases over the operating life of the plant.

IMPROVED PFM METHODOLOGY

The PFM model in SECY 82-465 [1] utilized a methodology that produced a boolean result for cleavage fracture initiation and RPV failure, i.e., the outcome for each RPV in the Monte Carlo analysis was fracture or no fracture and failure or no failure. The conditional probability of crack initiation (*CPI)* was calculated simply by dividing the number of RPVs predicted to experience cleavage fracture by the total number of simulated RPVs. Similarly, the conditional probability of RPV failure *CPF* was calculated by dividing the number of RPVs predicted to fail by the total number of simulated RPVs. The final results were discrete values for *CPI* and *CPF,* without any quantification of the uncertainty in the solution. The improved PFM model provides for the calculation of probability *distributions* of *CPI* and *CPF* and thus for the quantification of uncertainty in the results. The reader is referred to reference 23 for a more detailed explanation of the improved PFM model.

The overall PRA methodology for PTS integrates these probability distributions of *CPI* and *CPF* with distributions of transient initiating frequencies (events per reactor year) derived from plant system and human interaction considerations. Output from this process includes probability distributions for CIF and TWCF. It is the mean of these distributions that are illustrated in Figures 1 and 3.

PTS ANALYSIS RESULTS, TRENDS, AND INSIGHTS

As previously discussed, Figure 1 illustrates the mean value of TWCF for each of the three US commercial PWRs plotted as a function of RT_{PTS}. The improved embrittlement correlation discussed above was used in the calculation of RT_{PTS} for the three plants. The neutron fluence maps utilized for the higher levels of embrittlement were obtained by extrapolating beyond the two maps provided by BNL corresponding to 32 and 40 EFPY. Some of these extrapolations are far beyond the range of EFPY for which these plants would ever actually operate. These analyses were performed at the higher levels of embrittlement to establish the relationship between TWCF and values of RT_{PTS} at and above the current screening criteria.

Figure 3 illustrates the TWCF for each of the three PWRs plotted as a function of operating life. Tables 1 and 2 contain results that would be relevant as these three PWRs approach the end of their 40-year operating licenses and consider requesting a 20 year license renewal.

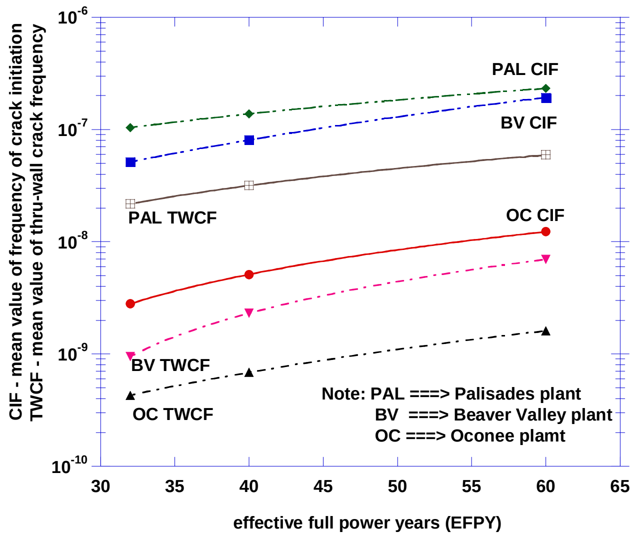

Figure 3 – Results of current PTS re-evaluation program as a function of reactor operating life

Tables 1 and 2 contain results that would be relevant as these three PWRs approach the end of their 40-year operating licenses and consider requesting a 20 year license renewal. Table 1 expresses the relationship between operating life and the current embrittlement metric RT_{PTS}. Based on these results, Oconee and Palisades are not expected to exceed the current PTS screening criteria within 60 EFPY. While Beaver Valley is projected to exceed the current PTS screening criteria, at 60 EFPY, the integrated risk analysis illustrates that TWCF at 60 years is considerably less than the currently acceptable 5.0×10^{-6}. In addition to illustrating that these three PWRs satisfy the current PTS criteria for a license extension from 40 to 60 years, the results of the analysis provide a technical basis for a relaxation to the current PTS screening criteria.

Table 1 – Relationship between RPV operating life and RT_{PTS}

EFPY	Oconee RT_{PTS} (°F)	Beaver Valley RT_{PTS} (°F)	Palisades RT_{PTS} (°F)
32	175.3	242.4	212.5
40	181.9	251.8	218.5
60	192.6	271.1	230.5

Table 2 – Results from PTS analyses relevant to license extension considerations

EFPY	Oconee CIF(1)	Beaver Valley CIF(1)	Palisades CIF(1)	Oconee TWCF(2)	Beaver Valley TWCF(2)	Palisades TWCF(2)
32	2.80E-09	5.15E-08	1.04E-07	4.29E-10	9.49E-10	2.18E-08
40	5.12E-09	8.05E-08	1.38E-07	6.86E-10	2.31E-09	3.17E-08
60	1.23E-08	1.93E-07	2.33E-07	1.61E-09	6.98E-09	5.94E-08

(1) mean value of probability distribution of crack initiation frequency (cracked RPVs per EFPY)
(2) mean value of probability distribution of thru-wall crack frequency (failed RPVs per EFPY)

Operationally, the events that dominate TWCF are all associated with primary coolant system failures. These include both loss of coolant accidents (LOCAs) and stuck open safety relief valves that produce a sudden repressurization when the valve does close. By far the greatest contribution to PTS risk arises from loss-of-coolant accidents (LOCAs).

In each of the three PWRs analyzed, the dominant contributing material to PTS-induced risk of failure was axial welds. Postulated flaws in plate regions contributed one per cent or less to the PTS-induced risk of failure owing to the smaller flaw size distribution of plate flaws relative to weld flaws.

For Oconee, a circumferential weld is the most embrittled material, i.e., has the highest value of RT_{PTS}. The most embrittled material for Beaver Valley is a plate and for Palisades is an axial weld. The fact that the dominant contributing material to PTS-induced risk of failure was axial welds, which are not necessarily the materials with the highest values of RT_{PTS}, suggests that an improved embrittlement metric can possibly be defined that more directly correlates with the factors affecting the likelihood of PTS-induced failure.

SUMMARY

In 1999, the USNRC initiated a comprehensive project to establish an updated technical basis for PTS regulation and to determine if it would lead to a relaxation in the current regulations. An updated technical basis has been established within the framework established by modern probabilistic risk assessment techniques and the advances in technologies associated with the physics of the PTS. During this project, an improved risk-informed computational methodology was developed that provides a more realistic characterization of PTS risk. This updated methodology has been applied to three commercial PWRs. The results of these analyses provide encouragement that a technical basis can be established to reduce the conservatism in the current regulations while continuing to provide reasonable assurance of adequate protection to public health and safety. Work is continuing to insure that that any proposed revision to the PTS rule may be applied generically to all commercial PWRs.

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