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STATUS OF THE UNITED STATES NUCLEAR REGULATORY COMMISSION PRESSURIZED THERMAL SHOCK RULE RE-EVALUATION PROJECT

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ABSTRACT

The current federal regulations to ensure that nuclear reactor pressure vessels (RPVs) maintain their structural integrity when subjected to transients such as pressurized thermal shock (PTS) events were derived from computational models that were developed in the early to mid 1980s. Since that time, there have been advancements in relevant technologies associated with the physics of PTS events that impact RPV integrity assessment. Preliminary studies performed in 1999 suggested that application of the improved technology could 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 extension considerations. Based on the above, in 1999, the United States Nuclear Regulatory Commission (USNRC) initiated a comprehensive project, with the nuclear power industry as a participant, to re-evaluate the current PTS regulations within the framework established by modern probabilistic risk assessment (PRA) techniques.

During the last three years, improved computational models have evolved through interactions between experts in the relevant disciplines of thermal hydraulics, PRA, human reliability analysis (HRA), materials embrittlement effects on fracture toughness (crack initiation and arrest), fracture mechanics methodology, and fabrication-induced flaw characterization. These experts were from the NRC staff, their contractors, and representatives from the nuclear industry. These improved models have now been implemented into the FAVOR (**F**racture **A**nalysis of **V**essels: **O**ak **R**idge) computer code, which is an applications tool for performing risk-informed structural integrity evaluations of embrittled RPVs subjected to transient thermal-hydraulic loading conditions. The baseline version of FAVOR (version 1.0) was released in October 2001.

The updated risk-informed computational methodology in the FAVOR code is currently being applied to selected domestic commercial pressurized water reactors to evaluate the adequacy of the current regulations and to determine whether a technical basis can be established to support a relaxation of the current regulations. This paper provides a status report on the application of the updated computational methodology to a commercial pressurized water reactor (PWR) and discusses the results and interpretation of those results. It is anticipated that this re-evaluation effort will be completed in 2002.

1. PROBLEM DEFINITION, HISTORICAL CONTEXT, AND CURRENT REGULATIONS

Concern with PTS arises because cumulative neutron irradiation exposure makes the RPV material more brittle (with reduced ductility and fracture toughness) and, therefore, increasingly susceptible to cleavage fracture. In PWRs, transients can occur that result in overcooling (thermal shock) of the RPV concurrent with or followed by a repressurization. If an embrittled RPV is subjected to a PTS event, existing crack-like defects (flaws) on or near the inner surface of the RPV could initiate in cleavage fracture, propagating 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 at that time for PRA, thermal hydraulics (TH) analysis, vessel fracture analysis, and material embrittlement estimation methods, the NRC staff developed one of the first performance-based, risk-informed regulations to ensure the safe operation of the RPV [1]. The PTS rule [2] established an embrittlement screening criterion above which licensees would be required to demonstrate that their RPVs could be operated safely, i.e., the PTS rule established screening criteria in the form of limiting values of reference nil-ductility transition temperature, designated as RT_{NDT}, a measure of the degree of embrittlement or loss of ductility of the RPV material. The current PTS screening criteria are values of RT_{NDT} at the inner surface of the RPV of 270 ^{0}F for plates, forgings, and axial welds, and 300 ^{0}F for circumferential welds. The screening criteria were derived using both conservative deterministic analyses and risk concepts that established an acceptable probability of RPV failure.

The NRC subsequently developed Regulatory Guide (RG) 1.154 [3] regarding the format and content of analyses

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that could be used to demonstrate the continued safe operation of RPVs that would exceed the PTS screening criterion. 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. RG 1.154 provides guidance regarding an acceptable analysis methodology. Probabilistic fracture mechanics (PFM) analysis is a major element of the RG 1.154 methodology. The objective of performing a RG 1.154 analysis is to demonstrate that the frequency of RPV failure per reactor operating year, caused by PTS, is less than the current acceptance criterion of 5×10^{-6} failures per reactor year. The value of 5.0×10^{-6} RPV failures per reactor year is also currently being re-examined by NRC staff and could be modified.

The Integrated Pressurized Thermal Shock (IPTS) studies were a series of studies performed in the early 1980s as part of an NRC-organized comprehensive research project to confirm the technical bases for the PTS rule and to aid in the development of Regulatory Guide 1.154. The research project consisted of PTS analyses for three PWR plants: Calvert Cliffs Unit 1 [4], designed by Combustion Engineering; H.B. Robinson (HBR) Unit 2 [5], designed by Westinghouse; Oconee Unit 1 [6], designed by Babcock and Wilcox.

In late 1989 and early 1990, the staff and the licensee for the now decommissioned Yankee Rowe PWR plant entered into an intensive evaluation of the RPV for the Yankee Rowe plant [7]. The staff had identified a high level of embrittlement for the pressure vessel and both the licensee and staff turned to Regulatory Guide 1.154 as a basis for evaluating the integrity of the pressure vessel. During the course of that evaluation, the staff and industry identified a number of shortcomings and limitations in the regulatory guide computational methodology; chief among them being the technical basis for the fabrication flaw distributions used in the PFM analyses. The Yankee Rowe evaluation, as well as the earlier evaluations that had formed the basis for the rule and regulatory guide, demonstrated that the way flaws were modeled (using 1970s non-destructive examination (NDE) data and resulting Marshall flaw distribution [8]) in the analysis dominated the uncertainty in the calculated probability of RPV failure.

In the intervening years, the staff and the industry have worked both separately and jointly to improve the data bases for the flaw-related variables and for the other key variables. Recent analyses [9] that combined the available advances to evaluate the potential impact on acceptable levels of pressure vessel embrittlement have shown that the level of conservatism in the screening criterion may be reduced without affecting the level of safety.

2. OVERVIEW OF PTS RE-EVALUATION PROJECT

The NRC has initiated the PTS Re-evaluation Project to revisit the technical bases for the PTS rule and to possibly propose a revision to that rule that will reduce any unnecessary level of conservatism that may exist in the current rule. An objective is to establish a technical basis rule within the framework established by modern probabilistic risk assessment techniques and advances in technologies associated with the physics of the PTS events. Figure 1 illustrates that elements of the improved fracture-related technology are integrated and implemented in the revised baseline version of FAVOR, which is currently being applied in the PTS re-evaluation program. In Figure 1, RVID refers to reactor vessel integrity database that is maintained at U.S. NRC on operating nuclear power plants' RPV materials.

Analogous to the IPTS studies [4-6] performed in the 1980s, the PTS Re-evaluation Project will analyze a number of commercial PWRs. An objective is to cover the range of various system designs, operational procedures, and training programs. The initial plan is to perform analyses for four commercial PWRs. Other analyses will be performed as necessary so that any revision to the PTS rule may be applied generically to all commercial PWRs.

3. APPLICATION OF UPDATED TECHNOLOGY TO PTS ASSESSMENT OF PLANT X

As discussed above, analysis of four commercial PWRs is planned as part of the PTS Re-evaluation Project. The FAVOR code was recently applied for the first time to a commercial PWR. For purposes of discussion, in this paper, the plant will simply be referred to as Plant X. A brief overview of the application of the updated technology used in PTS assessment of Plant X and the results of that analysis will be discussed. These results should be considered preliminary and possibly subject to change.

3.1 Improved Embrittlement Modeling

In the PFM analyses from which the current PTS regulations were derived [1], a single value of neutron fluence and chemistry was assumed to apply to the entire RPV beltline. FAVOR utilizes a methodology that allows the RPV beltline to be broken down into major regions such as axial welds, circumferential welds, and plate regions. The major regions may be further discretized into subregions to accommodate very detailed neutron fluence maps provided by Brookhaven National Laboratory (BNL).

Figure 2 illustrates the RPV beltline embrittlement model developed for the PFM analysis of Plant X. The model included the RPV material from one foot below the active core to one foot above the active core. Table 1 provides a breakdown of the RPV into major regions, with the respective chemistries, unirradiated values of RT_{NDT} (RT_{NDTo}), number of subregions into which the major regions are further discretized (to accommodate the detailed neutron fluence maps provided by BNL), etc. The most limiting embrittled region in the RPV beltline is the circumferential weld 1229 which, as illustrated in Table 2, has an RT_{NDT} at the inner surface of the RPV at 32 effective full power years (EFPY), of 180.7°F. The value of RT_{PTS} for this most limiting region is 238.9°F. The value of RT_{PTS} includes a margin term of 58.2°F, to account for uncertainties in the initial unirradiated value of RT_{NDTo} and the correlation used to predict the radiation-induced shift in RT_{NDT}. These values of RT_{NDT} and RT_{PTS} are calculated using the improved embrittlement correlation as discussed below in Section 3.5.

Detailed neutron fluence maps were provided for 32 and 40 EFPY. Neutron fluence maps for times in the operating life of the RPV later than 40 EFPY were obtained by

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extrapolating from the maps for 32 and 40 EFPY. The assumption associated with this extrapolation is that the current core (18 EFPY) refueling scheme is maintained. This assumption is also implicit in the fluence maps for 32 and 40 EFPY.

Each of the fluence maps contained 13080 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 104640 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. Each value of neutron fluence is at the inner surface of the RPV and is attenuated through the RPV wall (during the PFM analysis) using an exponential decay constant of 0.24 $inch^{-1}$.

Note that as a first approximation, each plate is modeled as a major region containing a single subregion, thus having a single value of neutron fluence. The value of neutron fluence in the entire plate region is assumed to be equal to the maximum value of neutron fluence in that region. This is obviously a conservative assumption; however, if the plates do not contribute significantly to the fracture or failure of the RPV when modeled conservatively, there is no incentive to further subdivide the plate into many subregions to accommodate the level of discretization provided in the fluence maps provided by BNL.

The modeling and procedures used in generating these neutron fluence maps were based on the guidance provided in the NRC Regulatory Guide 1.190 [10]. The calculations were performed using the DORT discrete ordinates transport code [11] and the BUGLE-93 [12] forty-seven neutron group ENDF/B-VI nuclear cross sections and fission spectra.

3.2 Improved Thermal Hydraulics Data

A comprehensive search for scenarios that are both probabilistically credible and physically significant is necessary. PRA is used to guide the choice of selected transients for thermal hydraulic analysis. There is feedback to PRA from thermal hydraulics and fracture mechanics analyses to determine not only the probability of occurrence of a given sequence but also its risk significance. Scenario screening integrates the knowledge from the three disciplines of PRA, thermal hydraulics, and fracture mechanics.

For the analysis of Plant X, Information Sciences Laboratories (ISL) provided thermal hydraulic boundary conditions for 46 transient conditions. The selection of these 46 conditions was based on not only 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 calculations were performed using the RELAP5/MOD3 code [13]. The specific transients were selected by considering the thermal hydraulic system behavior, as well as possible plant equipment failures, plant operating procedures, and human reliability issues. For each of the 46 transients, ISL provided a convective heat transfer coefficient time history, a coolant temperature time history, and a pressure time history. Each of the time histories consisted of 1000 time history pairs. The time history pairs were equally spaced by 15 seconds; therefore, the total duration of each of the transients was 250 minutes. Figure 3 illustrates the thermal hydraulic data for transient case 109, which is the most dominant transient, as will be discussed below.

3.3 Improved Scenario Identification and Frequencies

Sandia National Laboratory provided a probability distribution of the scenario frequency (events per reactor operating year) for each of the 46 transients. For each of the 46 transients, the probability distribution (histogram) has 20 bins. The SAPHIRE Version 7 [14] code was used to generate the probability distributions.

The relevant scenarios were identified by development of a PRA model for Plant X that addressed possible over-cooling transients that could occur. The identification of the possible over-cooling scenarios considered the earlier 1980s work, as well as 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 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.

As the PRA model was constructed, the latest initiator frequency estimates and equipment failure rates were applied, including means and uncertainties for these data. The human failure probabilities applied to possible operator actions in the scenarios, were assessed with a detailed consideration of the context (i.e., plant conditions and performance-shaping factors) relevant for each scenario analyzed. This was based on the ATHEANA process described in NUREG-1624 [15]. For each human error probability estimated, a mean and uncertainty distribution were assessed originally by NRC contractor staff, reviewed by Plant X staff, and revised as deemed appropriate.

The scenarios modeled in the PRA and quantified using the SAPHIRE code included a cooperative effort with Plant X staff to ensure that the model reflected current design and practices and that estimates were appropriate for the means and uncertainties for the initiator frequencies as well as equipment and human failure probabilities. Figure 4 illustrates the probability distribution for transient 109, which is the most dominant transient, as will be discussed below.

3.4 Improved Flaw Characterization Data

Perhaps the single largest improvement in the updated computational methodology is the establishment of a technical basis for the postulation of flaws. The PFM model utilized in the analyses [1] from which the current PTS regulations were derived conservatively postulated all fabrication flaws to be inner-surface breaking flaws.

The USNRC has supported research at Pacific Northwest National Laboratory that has resulted in the postulation of fabrication flaws based on the non-destructive and destructive examination of actual RPV material. Such measurements

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have been used to characterize the number, size, and location of flaws in various types of weld and base metal used to fabricate vessels, thus providing a technical basis for the flaw data, which is critical input data into FAVOR analyses. These measurements have been supplemented by expert elicitation [16]. Separate distributions have been developed to characterize the number, size, and location (in the RPV wall) of flaws in different regions of the RPV. The regions include the main seam welds, repair welds, base metal of plates and forgings, and the cladding as applied to the inner surface of the vessel. The reader is referred to References 17-19 for the details of this research. Table 2 illustrates the application of the improved postulated flaw characterization data to Plant X. In Table 2, the flaws are segregated according to RPV major region and flaw type.

The method used to quantify the uncertainty in the flaw characterization is to include as input data multiple (1000) flaw-characterization files, each which contains different flaw densities, flaw depth and aspect ratio distributions. Weld material and plate material have distinctly different files since they contain different flaw densities and size distributions. The flaw size (depth) distributions used in the analysis of Plant X were truncated at 23% and 5% of the RPV wall thickness for flaws in weld and plate regions, respectively, i.e., the deepest flaw in weld and plate material was 23 and 5 percent of the wall thickness, respectively.

During the Monte Carlo PFM analysis, the RPV flaw-characterization data for the first stochastically-generated RPV trial is taken from the first group of file records: the first weld material flaw characterization file and the first plate material file record. The RPV flaw characterization for the second stochastically generated RPV trial is determined from the second group of file records, etc. The RPV trials cycle through the flaw-characterization file records sequentially up to 1000, and then restarts at the first file record. Table 2 illustrates the distribution of flaws in various regions of the RPV beltline from the application of one of these files to Plant X RPV beltline.

The illustration in Table 2 is for one of those files that do predict the existence of inner-surface breaking flaws. All inner-surface breaking flaws (in plate or weld regions) are postulated to be circumferentially oriented as a result of the fabrication procedures for applying the cladding on the inner surface of the RPV. Embedded flaws are assumed to be axially oriented if located in axial welds, plates, or forgings. Embedded flaws are assumed to be circumferentially oriented if located in a circumferential weld.

3.5 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 vessel. The current regulatory correlation, described in Regulatory Guide 1.99, Revision 2 [20] 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 embrittlement correlations have been recently published by researchers from Modeling and Computing Services and the University of California at Santa Barbara [21].

3.6 Improved Fracture Toughness Model

The computational model for quantification of fracture toughness uncertainty is improved in two ways: (1) the K_{Ic} and K_{Ia} databases were extended by 83 and 62 data values, respectively, relative to the databases in the EPRI report [22] and (2) the statistical representations for K_{Ic} and K_{Ia} were derived through the application of rigorous mathematical procedures. Bowman and Williams [23] 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.

3.7 Improved PFM Methodology

The FAVOR PFM model is based on the Monte Carlo technique, where deterministic fracture analyses are performed on a large number of stochastically generated RPV *trials* or *realizations*. Each RPV realization can be considered a perturbation of the *uncertain* condition of the specific RPV under analysis. The condition of the RPV is considered uncertain in the sense that a number of the vessel's fracture-toughness-related properties along with the postulated flaw population have uncertainties associated with them. These input uncertainties are described by statistical distributions. The RPV trials propagate the uncertainties with their interactions through the model, thereby determining the conditional probability of crack initiation (CPI) and the conditional probability of RPV failure (CPF) for a set of postulated PTS events at a selected time in the vessel's operating history. These values are "conditional" in the sense that the transient is assumed to occur and the postulated flaws are assumed to exist. The term "failure" is used to describe a crack that propagates to 90 percent of the wall thickness.

Current PTS regulations are based on analyses from a PFM model that produced a Boolean result for cleavage fracture initiation and RPV failure [1,4-6], i.e., the outcome for each RPV in the Monte Carlo analysis was fracture (CPI_{RPV}=1) or no fracture (CPI_{RPV}=0) and failure (CPF_{RPV}=1) or no failure (CPF_{RPV}=0). The *CPI* was calculated simply by dividing the number of RPVs predicted to experience cleavage fracture by the total number of simulated RPVs. Similarly, the *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 *CPI,* without any quantification of the uncertainty in the solution.

An improved PFM model provides for the calculation of probability *distributions* of CPI and CPF and thus for the quantification of uncertainty in the results. In the improved PFM model, values of CPI_{RPV} ($0 \le CPI_{RPV} \le 1.0$) and CPF_{RPV} ($0 \le CPF_{RPV} \le 1.0$) are generated for each RPV simulated in a Monte Carlo analysis. The improved PFM model also provides estimates of the uncertainties in its outputs in terms of discrete statistical distributions. By repeating the RPV trials a large number of times, the output values constitute a

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random sample from the probability distribution over the output induced by the probability distribution over the several input variables. The reader is referred to Reference 24 regarding a more detailed explanation of the improved PFM model.

The overall computational methodology for PTS integrates these probability distributions of CPI and CPF with distributions of scenario frequencies derived from plant system and human interaction considerations. Output from this process includes probability distributions for RPV fracture and failure frequencies. The reader is referred to Reference 25 regarding a more detailed explanation of the method of combining the distributions of scenario frequencies with the results of the PFM analysis to generate probability distributions for RPV fracture and failure frequencies

4. PLANT X ANALYSIS RESULTS

Table 3 is a summary of the risk-informed PTS analysis results of Plant X, which contains the mean and 95th percentile of the probability distributions for the frequency of crack initiation and the frequency of RPV failure. Analyses were performed at nine levels of embrittlement, each one in principal, corresponding to a particular point in the operating life of the RPV. The first two analysis results correspond to the neutron fluence maps of 32 and 40 EFPY, which were provided by BNL as previously discussed. The neutron fluence maps for the additional levels of embrittlement (seven times in the RPV operating life) were obtained by extrapolating beyond these two maps. Clearly, some of these extrapolations are far beyond the range of EFPY for which plant X would ever actually operate. They were performed since an objective of the analysis was to determine the level of embrittlement that corresponds to a frequency of RPV failure in the 10^{-6} range.

Figure 5 illustrates the mean and 95th percentile of the probability distributions for the frequency of crack initiation and the frequency of RPV failure plotted as a function of the most limiting value of RT_{PTS} in the RPV beltline. The value of the 95th percentile of the frequency of RPV failure does not reach 10^{-6} failures per reactor operating year when the limiting beltline value of RT_{PTS} is considerably greater than 300 F. In figure 6 the values are plotted as a function of RT_{NDT} that has been weight averaged over the beltline distribution of flaws, which is perhaps a better indicator of the embrittlement of the entire RPV beltline region.

Table 4 is a summary of the thermal hydraulic characteristics and the analysis results of the nine dominant transients, i.e., those transients that contribute most significantly to the frequency of RPV failure. The most distinguishing characteristic of these dominant transients is that they all have a minimum coolant temperature of less than 100 ^{o}F. Transient sequence 109 is the most dominant of all the transients since it contributes 28.5 percent of the total frequency of RPV failure. Another distinguishing feature of this most dominant transient, in addition to the low coolant temperature, is that it has a repressurization late in the transient. The coolant and pressure time histories of transient sequence 109 are illustrated in Figure 3. The probability distribution of the initiating frequency for this transient is illustrated in Figure 4. The combination of the transient severity and the frequency with which it is predicted to occur determines the dominance of the transient.

For transient 109, the axial welds, which are less embrittled and contain fewer flaws than the circumferential welds (due to smaller volume), contribute over 91 percent of the conditional probability of initiation and over 97 percent of the conditional probability of RPV failure. The fact that the axial welds are the principal contributors to crack initiation and failure is a result of loading caused by the repressurization. The pressure-induced hoop stress (principal stress used in the determination of the applied K_I for axially oriented flaws) is twice the magnitude of the pressure-induced axial stress (principal stress used in the determination of the applied K_I for circumferentially oriented flaws). Therefore, the applied values of K_I used in the determination of CPI and CPF for axial welds are impacted more by the repressurization than the applied values of K_I for circumferential welds.

The PFM analysis results validate the assumption discussed earlier that the plate regions do not contribute significantly to the conditional probability of crack initiation or RPV failure. Therefore, the conservative assumption of assuming each plate region to have the maximum value of fluence in the plate has no impact on the solution. Further discretization of the plate would have resulted in the same CPI and CPF. If the plate regions had contributed significantly to the solutions, additional analyses would have been performed to include additional discretization of the neutron fluences in the plate regions.

5. SUMMARY

As part of the United States Nuclear Regulatory Commission PTS Rule Re-evaluation Project, an updated risk-informed computational methodology has evolved through interactions between experts from the NRC staff, their contractors, the general public, and representatives from the nuclear industry. An objective of the PTS Re-evaluation Project is to establish a technical basis rule within the framework established by modern probabilistic risk assessment techniques and advances in technologies associated with the physics of the PTS events. A relaxation of the current PTS regulations could have profound implications for plant license extension considerations.

The updated risk-informed computational methodology has been integrated into the FAVOR code, which was recently applied to an actual PWR vessel. The value of the 95th percentile of the frequency of RPV failure does not reach 10^{-6} failures per reactor operating year when the limiting beltline value of RT_{PTS} is considerably greater than 300 ^{o}F. While it is premature to draw a general conclusion, the preliminary result of the PTS analysis of the Plant X is encouraging that the application of the state-of-the-art technology will establish a technical basis for a potential relaxation of the current PTS regulations for commercial PWRs.

Work is continuing on PTS evaluation of selected other PWR plants. It is anticipated that this work will be completed in 2002.

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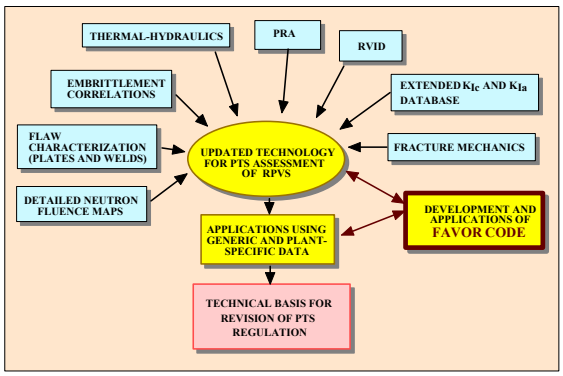

Figure 1 – Elements of improved fracture-related technology are integrated through the application of the FAVOR code, which is currently being applied in the USNRC PTS re-evaluation program.

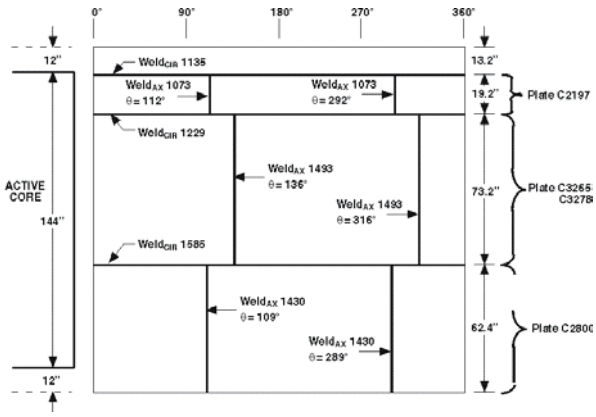

Figure 2 – Breakdown of RPV beltline for Plant X into major regions

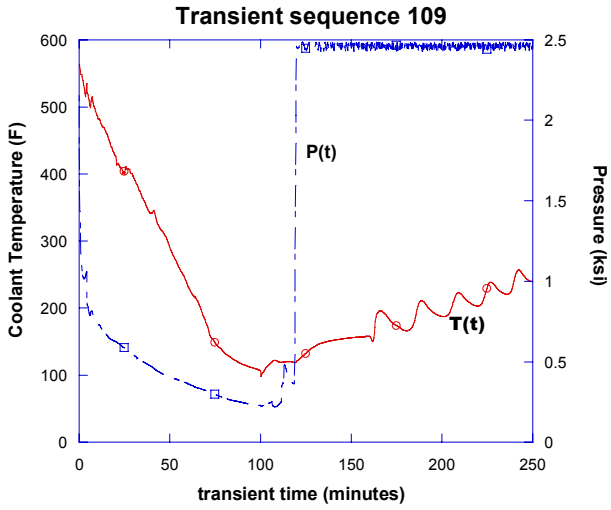

Figure 3 – Transient thermal hydraulic boundary condition imposed on RPV inner surface for transient sequence 109

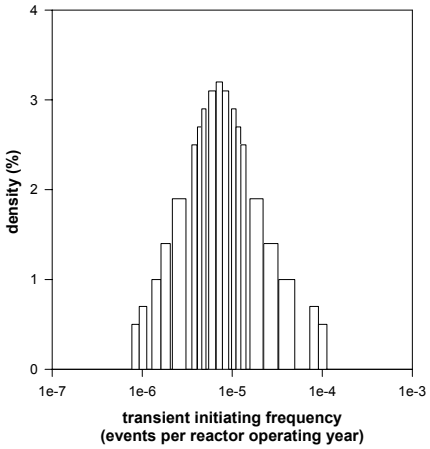

Figure 4 – Probability distribution for the initiating frequency for transient sequence 109

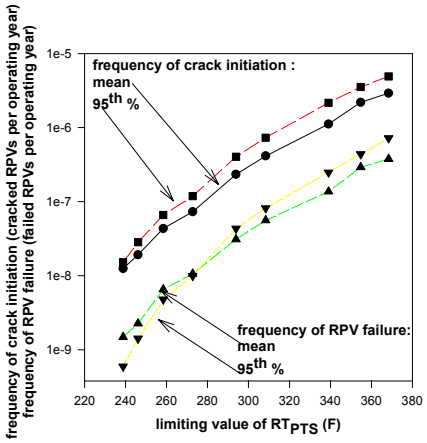

Figure 5 – Risk Informed PTS analysis results plotted as a function of RT_{PTS} of the most embrittled RPV beltline region

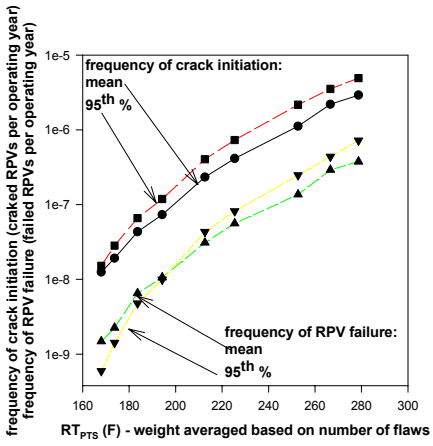

Figure 6 – Risk Informed PTS analysis results plotted as a function of weight averaged RT_{PTS} of the RPV beltline region (weight averaging based on number of flaws)

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Table 1 – Plant X RPV Beltline Major Region Embrittlement Data

Major RPV Region Number	Major RPV Region Description	Number of RPV subregions	% Cu	% Ni	% Phos	RT_{NDTo} (°F)	Max neutron fluence @ 32 EFPY 10^{19} n/cm^2	RT_{NDT} (max) @ 32 EFPY (°F)	RT_{PTS} max) @ 32 EFPY (°F)	Volume (ft^3)
1	ax weld 1430	78	.19	.57	.017	-8	0.72	131.6	189.8	1.00
2	ax weld 1493	87	.19	.57	.017	-8	0.68	130.2	188.4	1.17
3	ax weld 1073	55	.21	.64	.025	-8	0.61	155.1	213.3	0.30
4	cir weld 1585	480	.22	.54	.016	-8	0.84	147.5	205.7	4.36
5	cir weld 1229	480	.23	.59	.021	10	0.76	180.7	238.9	4.36
6	cir weld 1135	480	.23	.52	.011	-8	0.34	120.2	178.4	4.36
7	plate c2800	1	.11	.63	.012	0	0.79	75.5	127.2	171.81
8	plate c3265	1	.10	.50	.015	0	0.79	68.5	120.2	100.77
9	plate c3278	1	.12	.60	.010	0	0.79	77.4	129.1	100.77
10	plate c2197	1	.15	.50	.008	0	0.72	82.3	134.0	52.87

Note: $RT_{NDT} = RT_{NDTo} + \Delta RT_{NDT}$
where: RT_{NDTo} = initial unirradiated value of RT_{NDT}
ΔRT_{NDT} = shift in RT_{NDT} caused by radiation damage
$RT_{PTS} = RT_{NDT}$ + margin; where margin = $2\,[\sigma_{RTNDTo} + \sigma_{\Delta RTNDT}]^{1/2}$
where: σ_{RTNDTo} = 17 F for weld and plate material
$\sigma_{\Delta RTNDT}$ = 23.6 F for weld material and 19.5 F for plate material

Table 2 – Plant X RPV Beltline Flaw Distribution

Major RPV Region Number	Description	RT_{PTS} (max) @ 32 EFPY (°F)	Volume (ft^3)	Number of category 1 flaws(1)	Number of category 2 flaws(2)	Number of category 3 flaws(3)	Total number of flaws	% of Total number of flaws
1	ax weld 1430	189.8	1.00	.0100	90	180	270.01	3.36
2	ax weld 1493	188.4	1.17	.0170	105.3	210.6	316.07	3.93
3	ax weld 1073	213.3	0.30	.0030	27.0	54.0	81.00	1.00
4	cir weld 1585	205.7	4.36	.0436	392.4	784.8	1177.24	14.65
5	cir weld 1229	238.9	4.36	.0436	392.4	784.8	1177.24	14.65
6	cir weld 1135	178.4	4.36	.0436	392.4	784.8	1177.24	14.65
7	plate c2800	127.2	171.81	1.718	515.43	1030.86	1548.00	19.25
8	plate c3265	120.2	100.77	1.007	302.31	604.62	907.94	11.29
9	plate c3278	129.1	100.77	1.007	302.31	604.62	907.94	11.29
10	plate c2197	134.0	52.87	.5287	158.61	317.22	476.36	5.93
		totals	441.77	4.4185	2678.16	5356.32	8039.04	100.00

(1) Category 1 flaws are inner surface breaking flaws
(2) Category 2 flaws are planar embedded flaws in which the inner crack tip resides between the clad base interface and t/8 where t is the RPV wall thickness
(3) Category 3 flaws are planar embedded flaws in which the inner crack tip resides between t/8 and 3t/8

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Table 3 – Results of risk-informed PTS analysis for Plant X

Max RT_{PTS} of circ weld (^oF)	Max RT_{PTS} of axial weld or plate (^oF)	Weight average RT_{PTS} (^oF) (based on flaws)[3]	Mean frequency of crack initiation[1]	95th percentile of frequency of crack initiation (1)	Mean frequency of RPV failure[2]	95th percentile of frequency of RPV failure(2)
238.8	213.2	168.0	1.25e-8	1.52e-8	1.48e-9	5.95e-10
246.1	221.2	173.7	1.92e-8	2.83e-8	2.26e-9	1.42e-9
258.4	234.5	183.6	4.34e-8	6.60e-8	6.54e-9	4.78e-9
272.8	250.0	194.2	7.32e-8	1.19e-7	1.06e-8	9.94e-9
293.9	272.2	212.6	2.34e-7	4.05e-7	3.10e-8	4.33e-8
308.4	287.2	225.4	4.14e-7	7.32e-7	5.60e-8	8.17e-8
339.1	318.8	252.7	1.12e-6	2.17e-6	1.37e-7	2.48e-7
354.8	334.9	266.6	2.20e-6	3.54e-6	2.92e-7	4.39e-7
368.4	348.9	278.7	2.92e-6	4.93e-6	3.764e-7	7.22e-7

(1) frequency of crack initiation expressed in cracked RPVs per reactor operating year
(2) frequency of RPV failure expressed in RPV failures per reactor operating year
(3) $RT_{PTS} = \sum_{I=1} (RT_{PTS})$ (fraction of total flaws in region)

Table 4 – Dominant transients for Plant X evaluated at 60 EFPY

Transient sequence number	Minimum coolant temp (^oF)	Last coolant temp (^oF)	Min pressure (ksi)	Last pressure (ksi)	CPI_{mean}	CPF_{mean}	Mean value of transient initiating frequency (events per year)	% of total frequency of crack initiation	% of total frequency of RPV failure
109	98.18	239.71	0.220	2.478	1.5715e-4	1.4766E-4	1.023e-5	4.50	28.52
113	98.33	289.70	0.218	0.219	6.4865e-5	5.4933E-5	1.461e-5	3.55	19.56
53	70.09	70.16	0.062	0.084	1.7424e-3	1.0994E-4	8.358e-6	38.00	16.08
141	71.42	71.42	0.176	0.208	1.1781E-4	1.3875E-5	4.926e-5	13.52	10.41
112	98.33	288.56	0.218	0.219	1.2842e-5	1.7193E-6	4.318e-4	16.19	8.67
52	70.59	70.59	0.124	0.143	7.1976e-4	5.2041e-5	7.151e-6	15.34	8.50
144	70.95	70.95	0.179	0.207	2.8095E-5	3.5369E-6	6.295e-5	4.12	3.49
55	73.63	73.75	0.186	0.213	5.2733e-5	6.7948E-6	8.343e-6	1.04	1.23
146	92.91	91.91	0.310	0.390	4.5227E-6	7.5406E-7	1.044e-4	1.02	1.01

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