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AUTH.NAME AUTHOR AFFILIATION
SHIFFER,J.D. Pacific Gas & Electric Co.
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SUBJECT: Forwards "Evaluation of Reactor Vessel Integrity Split Fractions for Various PTS Challenges," as part of long-term seismic program PRA.

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Pacific Gas and Electric Company

77 Beale Street
San Francisco, CA 94106
415/973-4684
TWX 910-372-6587

James D. Shiffer
Senior Vice President and
General Manager
Nuclear Power Generation

July 3, 1991

PG&E Letter No. DCL-91-172

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Long Term Seismic Program - Probabilistic Risk Assessment

Gentlemen:

In response to an oral request from the NRC Staff, enclosed is Appendix L of the Diablo Canyon Probabilistic Risk Assessment, titled "Evaluation of Reactor Vessel Integrity Split Fractions for Various Pressurized Thermal Shock Challenges."

Sincerely,

J. D. Shiffer

cc: Ann P. Hodgdon
John B. Martin
Phillip J. Morrill
Paul P. Narbut
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ENCLOSURE

PROBABILISTIC RISK ASSESSMENT APPENDIX L "EVALUATION OF REACTOR VESSEL INTEGRITY SPLIT FRACTIONS FOR VARIOUS PRESSURIZED THERMAL SHOCK CHALLENGES"

APPENDIX L. EVALUATION OF REACTOR VESSEL INTEGRITY SPLIT FRACTIONS (TOP EVENT VI) FOR VARIOUS PRESSURIZED THERMAL SHOCK CHALLENGES

L.1.1 INTRODUCTION

Pressurized thermal shock (PTS) is the term used to describe an event in a pressurized water reactor that produces a severe overcooling of the inside surface of the reactor vessel wall, concurrent with or followed by repressurization. Rapid overcooling events can produce steep temperature gradients through the vessel wall, creating high thermal stresses, which are tensile on the cool, inner surface and compressive on the outer surface. When these thermal stresses are combined with the tensile stress due to internal pressure, there is some concern that the resulting stress intensity can cause cracks to propagate from preexisting defects in the vessel base metal or welds. The ability of the pressure vessel to withstand a PTS challenge is dependent on the fracture toughness of the materials and the location, size, and orientation of any flaws that may exist in the vessel. For a given pressure vessel material, fracture toughness generally decreases with decreasing temperature and increasing neutron fluence. Because of the dependence on fluence, a severe PTS event that occurs after the vessel has experienced significant irradiation is more likely to cause unstable crack growth than if the same event occurred early in the vessel life.

The NRC has developed screening criteria (Reference L-1), herein called the PTS Rule, to identify reactor vessels that may, at some time in the plant life, have inadequate toughness to safely withstand a PTS challenge. The measure of fracture toughness used in the PTS Rule is reference temperature for nil-ductility transition (RT_{NDT}). RT_{NDT} is based on the temperature at which the material exhibits a change from ductile to brittle behavior in the dropweight and Charpy V-notch tests. A low RT_{NDT} indicates that the transition from ductile to brittle behavior occurs at a low temperature and is generally indicative of high toughness. A high RT_{NDT} indicates that the material can behave in a brittle manner even at high temperature, and generally indicates low toughness. The procedure for determining the initial (i.e., unirradiated) RT_{NDT} of the vessel base and weld metals is given in Reference L-2. The NRC PTS Rule predicts the change in RT_{NDT} as a function of fast neutron fluence (En > 1 Mev) for a material of a given composition (copper and nickel content producing accelerated embrittlement). When the reference temperature is calculated by the PTS Rule equation (which includes margin to account for uncertainties), it is called RT_{PTS} to distinguish it from a measured value of RT_{NDT}. When the predicted RT_{PTS} exceeds the screening limit, the NRC position is that (1) neutron fluence must be reduced so that the screening limit is not exceeded, or (2) additional analyses must be performed to verify that adequate fracture toughness exists through the end of life. The PTS Rule establishes upper limits on RT_{PTS} of (1) 270°F for base material and longitudinal welds and (2) 300°F for circumferential welds. A higher limit is allowed for circumferential welds because the pressure stress across the weld (i.e., the axial stress) is only about half that across a longitudinal weld (i.e., the circumferential or hoop stress) and the vessel wall bending stiffness in the axial direction is higher, providing a greater constraint on crack growth for circumferentially oriented cracks. An evaluation of the Diablo Canyon Units 1 and 2 reactor vessel materials was made at the predicted end-of-life (i.e., 40 effective full power years) fluence level (Reference L-3). As will be discussed in Section L.2, the Diablo Canyon vessel material in both vessels satisfied the PTS Rule, implying that PTS events should not be significant contributors to risk.

There have been several analyses of the risk associated with PTS challenges. These analyses typically are conducted in several parts. The first part is in the form of a PRA, and evaluates the frequency of different categories of PTS challenges. Each category is characterized by a cumulative frequency (in challenges/reactor year), a representative pressure level and a representative cooldown rate and equilibrium cold temperature of the water adjacent to the vessel wall (i.e., the downcomer water). In order to evaluate the cooldown transient, thermal-hydraulic (TH) analyses must be performed for each representative scenario to develop both temperature and thermal stress distributions through the vessel wall. Next, probabilistic fracture mechanics (PFM) analyses are conducted, which address uncertainties in material properties as well as flaw size, orientation and distribution. These PFM analyses attempt to evaluate the conditional frequency of developing a through-wall crack, which is postulated to result in an excessive LOCA to the extent that core cooling cannot be assured and core damage results. The resulting PRA, TH and PFM analyses are then combined to assess the frequency of PTS caused core damage. As one can well imagine, this represents a very complex and expensive assessment requiring the integration of several rather diverse technical disciplines. Reference L-4 is such an evaluation done for a generic PWR by Westinghouse for the Westinghouse Owners Group. This work uses some of the PFM results from the NRC staff evaluation of PTS described in Reference L-5.

Since both Diablo Canyon reactor vessels satisfied the NRC PTS Rule with reasonable margin, PTS should not be a significant contributor to core melt. It was therefore decided in the DCPRA to include PTS vessel integrity questions in only a few initiating event models and to use the conditional vessel failure frequencies for Diablo Canyon specific end-of-life RT_{NDT} values using the results from Reference L-4. As will be discussed in Section L.3, we have attempted to err in a conservative direction when interpreting the results from Reference L-4.

L.2 EVALUATION OF DIABLO CANYON REACTOR VESSELS WITH REGARD TO THE NRC PTS SCREENING CRITERIA

The following discussion will summarize the evaluation made in Reference L-3.

The PTS Rule screening criteria consist of two equations which estimate the adjusted reference temperature, RT_{PTS}, as a function of the initial reference temperature (RT_{NDT}), the weight percent of copper and nickel in the material in question, and the fast neutron fluence at the inside wall of the vessel. RT_{PTS} is the lower of the values calculated by both equations.

Equation (L.1) is as follows:

$$RT_{PTS} = I + M + [-10 + 470Cu + 350CuNi] \times f^{0.27} \quad (L.1)$$

where

I = the initial reference temperature, RT_{NDT}, of the subject material measured as defined in the ASME Code (Reference L-2). When a measured value for submerged arc weld metal is not available, a generic value that is a function of submerged arc flux type must be used. For all Diablo Canyon weld metal, the generic value is -56°F.

M = the margin to be added to account for uncertainties in the values of initial RT_{NDT}, copper and nickel content, and the calculational procedure. When the RT_{NDT} is a measured value, M equals 48°F. When it is estimated, M equals 59°F. (Note: It is our understanding that this margin is intended to represent the 2σ confidence value.)

Cu and Ni = the best estimates of weight percent of copper and nickel in the material in question.

f = the best estimate of end-of-life fast neutron fluence (E_n > 1 Mev) at the clad/base metal interface, in units of 10^{19} neutrons/cm^2.

Equation (L.2) is as follows:

$$RT_{PTS} = I + M + 283\ f^{0.194} \tag{L.2}$$

Equation (L.2) is the upper bound of the Regulatory Guide 1.99, Revision 1 (Reference L-6) curve and is independent of material chemistry. For all materials in the Diablo Canyon reactor vessels, Equation (L.1) gave the lower RT_{PTS}, so all values reported here are calculated by Equation (L.1).

The PTS Rule requires that RT_{PTS} be calculated for each plate, forging, and weld in the beltline region. The beltline region is defined as the part of the vessel that directly surrounds the effective height of the active core, and other adjacent regions that experience significant neutron fluence. The beltline region includes only the intermediate and lower shell courses and the lower portion of the upper shell course (see Figure L-1). The remainder of the vessel does not experience sufficient fluence to be considered in evaluating the most limiting materials.

As mentioned earlier, the PTS Rule establishes an upper limit on RT_{PTS} of 270°F for base material and longitudinal welds and 300°F for circumferential welds. The Diablo Canyon reactor vessels are fabricated by the rolled and welded shell method shown pictorially in Figure L-1. The cylindrical portion is made from three shells (upper, intermediate, and lower). Each shell is made up of three 120° rolled sections that are welded together longitudinally. The shells are then welded together circumferentially. Only the intermediate and lower shell experience sufficient neutron fluence to be of concern (although Reference L-3 evaluated all shells). The more significant results of the assessment are summarized in Table L-1. For the Unit 1 vessel, the limiting material is the three longitudinal welds in the intermediate shell with an RT_{PTS} value of 217°F; the longitudinal welds in the intermediate shell have an RT_{PTS} of 197°F. For Unit 2, the shell base metal is most limiting; one 120° shell has an RT_{PTS} value of 228°F; another has a value of 224°F. These two shells have fairly high unirradiated RT_{NDT} values, so their fluence margin is actually higher than that for Unit 1. Both vessels satisfy the criteria. Therefore, the NRC requires neither a flux reduction program nor further analyses.

L.3 EVALUATION OF CONDITIONAL FREQUENCY OF VESSEL FAILURE (SPLIT FRACTION VI) FOR SELECTED EVENTS

Based on the results presented in Reference L-4 and on initiating event frequencies and systems analyses from the DCPRA, vessel integrity following a PTS challenge is questioned (as Top Event VI) in the secondary depressurization model (turbine trip and MSIV closure failures or unisolable steam line breaks, split fraction VI1, in bleed-and-feed scenarios (loss of steam generator cooling events, split fraction VI2), in the medium LOCA model (split fraction VI3), and in the steam generator tube rupture model (split fractions VI4 and VI5). Figure IV.3-1 of Reference L-4 shows that only LOCAs, SGTRs, and bleed and feed have any risk significance for the end-of-life RT_{NDT} values of the Diablo Canyon vessels. Secondary depressurization events were included since split fraction VI was included in the original DCPRA model.

The objective of this section is to extract mean values of the five VI split fractions (e.g., the conditional frequency of vessel failure, denoted by through-wall cracking, for the Diablo Canyon vessels) from the material presented in Reference L-4. We have intended to err in the conservative direction when there was not a one-to-one comparison between the DCPRA model and the Reference L-4 model. The discussion below will describe briefly the analysis done in Reference L-4 and then describe how these results are used to determine appropriate VI split fractions for use in the DCPRA. In the discussion, the Reference L-4 report will be referred to the Westinghouse Owners Group (WOG) study.

The WOG study evaluated seven initiating event categories that could result in a PTS challenge to the reactor vessel; these included the four selected for inclusion in the DCPRA. These categories are:

1. Secondary Depressurization (e.g., unisolable steam line breaks, excessive steaming)

2. Loss of Coolant Accident (e.g., medium LOCA)

3. Steam Generator Tube Rupture

4. Loss of Secondary Heat Sink (e.g., bleed and feed)

5. Excessive Feedwater

6. Anticipated Transient Without Scram

7. Feedline Break

Plant event models were then developed for each category with each sequence ending in a so-called PTS bin. Each bin corresponds to some representative vessel cooldown transient (i.e., the degree and rapidity of downcomer water temperature change) with some representative vessel pressure level. There are generally two different types of overcooling:

- <u>Excessive Heat Removal by the Steam Generators</u>. Events caused either by rapid secondary depressurization or by excessive (cold) feedwater addition.

- <u>Loop Stagnation with Cold Water Injection</u>. Events wherein flow through one or more reactor coolant loop becomes stagnant and cold water is injected into the loop, both cooling down the downcomer water and repressurizing the vessel.

Excessive heat removal events generally have abundant loop flow (by natural convection, which is the driving force for the cooldown transient, and possibly by reactor coolant pumps). Loop stagnation can occur when a reactor coolant pump stops and when effective natural convection flow to the steam generator ceases. Loop stagnation can occur in a single loop (i.e., asymmetric) or in all loops (i.e., symmetric), depending on the scenario.

Another important parameter addressed in the WOG study was the core decay heat rate when the event occurs. These decay heat categories were considered:

- DH1. Decay heat greater than 1%, corresponding to long-term prior operation at high power and shutdown times less than about 2 hours.

- DH2. Decay heat between 0.5% and 1%, corresponding to hot shutdown times from 2 to 20 hours following long-term high power operation, reduced power prior operation, or events occurring in the early stages of the fuel cycle.

- DH3. Decay heat less than 0.5%, corresponding to hot shutdown times greater than 20 hours following long-term high power operation or during the power ascension stage.

For excessive heat removal scenarios, decay heat affects the lower (equilibrium) temperature level and (probably) has a minor effect on the cooldown rate. For loop stagnation events, decay heat affects both the likelihood and rapidity of stagnation conditions (because decay heat provides the driving force for convection flow). In both cases, lower decay heat levels result in more severe PTS challenges. DH1, DH2, and DH3 conditions were estimated in the WOG study to be about 75%, 15%, and 10%, respectively. The DCPRA does not explicitly address differences in decay heat levels, since it is primarily assessing core damage events rather than overcooling events.

Other variables included in the WOG study are equivalent break sizes (for secondary depressurization events and for LOCAs) and the rapidity of certain operator actions. These will be discussed further for each of the initiating event categories analyzed.

The next step in the WOG PTS analysis was a thermal-hydraulics evaluation of each of the overcooling bins used to terminate each sequence from the event tree analysis. Each cooldown transient was characterized by a so-called stylized exponential cooldown transient by two parameters:

- FTEMP - the final temperature (°F)
- BETA - an exponential time constant (1/minutes)

Thus, the stylized downcomer fluid temperature (T_{DC}) varies with time (t, in minutes) as

$$T_{DC} = FTEMP + (T_i - FTEMP) \times e^{-BETA \times t}$$

where T_i is the initial downcomer water temperature, taken as 550°F.

The detailed thermal-hydraulic analyses were conducted with three proprietary Westinghouse computer programs:

- LOFTRAN, used for all non-LOCA transients.

- NOTRUMP, used for the analysis of small LOCAs.

- MXGCUP, a so-called "mixing cup" model used during intervals when SI injection into stagnant loop(s) is taking place. The LOFTRAN or NOTRUMP programs supply initial and boundary conditions to MXGCUP.

The MXGCUP program was validated based upon the results of several tests conducted by CREARE Corporation for EPRI.

The next step in the evaluation was the probabilistic fracture mechanics (PFM), which quantifies the conditional probability of vessel failure for a given stylized transient (as denoted by FTEMP and BETA), a pressure level (CPRESS), and the RT_{NDT} value at the inside surface of the vessel. The fast neutron fluence (F) attenuation through the wall decreases exponentially as

$$F(a) = F(ID) \times e^{-0.33 \times a}$$

where

a = distance into the wall (inches).

F(ID) = the fluence at the inside wall.

The reactor vessel modeled has a wall thickness and mean radius of 9 inches and 90 inches, respectively (the corresponding Diablo Canyon numbers are 8.5 and 86.5 inches, which are close and conservative). A Monte Carlo evaluation is made to evaluate the probability of a through-wall crack flaw extension in a longitudinal weld. The analysis treats as random, normally distributed variables the initial (longitudinally oriented) crack depth, the initial (unirradiated) RT_{NDT} value, copper content, fluence and the critical stress intensity factor. The NRC PFM assumes an infinitely long (i.e., continuous) flaw for both crack initiation and arrest. Subsequent WOG analyses assume a 6:1 finite flaw for crack initiation. No benefit was claimed for warm prestressing or for flaw detection by nondestructive examination. About 10^6 Monte Carlo trials were run per case, and the conditional probability of weld failure was calculated simply by dividing the number of trials which failed by the total number of trials. It was generally assumed that each vessel had six longitudinal welds, so the conditional probability of vessel failure was six times that for an individual weld. A large number of cases were run for five values of RT_{NDT}, three values of FTEMP (150°F, 225°F, and 300°F), three values of BETA (.05, 0.15, and 0.5 min^{-1}), and five values of CPRESS (0, 500, 1,000, 1,500, and 2,000 psi).

The WOG analysis is for cracks initially in longitudinal welds. Circumferential welds should not be a problem as noted earlier, and Reference L-3 indicates that the circumferential weld material used in the Diablo Canyon vessels is good. As noted earlier and in Table L-1, the Diablo Canyon specific Unit 1 reactor vessel data indicates longitudal welds to have the highest RT_{PTS} value. However, the Unit 2 vessel is limited by two (of three) intermediate shell plates (i.e., the base metal). The following statement is made on page 166 of Reference L-4: "For vessels limited by base plates, the factor of 6 should be conservative because the probability of flaw existence is expected to be lower in shell plates and forgings than in welds." Thus, we will use the same approach for both the Unit 1 and 2 vessels.

The WOG report defines for each initiating event category the frequency of significant flaw extension of a single longitudinal weld (based upon the NRC continuous flaw model) for several PTS bins for RT_{NDT} values of 200°F, 250°F, 300°F, and 350°F. Also given are the frequency associated with each PTS bin and the cumulative flaw extension values for the entire vessel (i.e., six times the per weld value) base on both the NRC continuous flaw and the WOG finite flaw models. In this conservative assessment, we are interpreting significant flaw extension to be through-wall cracking. Based upon a telephone conference call with knowledgeable Westinghouse and PGandE people (Reference L-7), significant flaw extension implies that the crack propogates through at least 75% of the wall thickness, but it does not necessarily imply vessel rupture. It was stated by a Westinghouse participant that the frequency of vessel rupture could be 2 or 3 orders of magnitude less likely than that termed "significant flaw extension (SFE)." This observation is based upon the Westinghouse analysis of the H. B. Robinson plant and on similar analyses done at Pacific Northwest Laboratories. It was also stated it would be very difficult to quantify the conservation. Therefore, as stated earlier, it will be conservatively assumed in this analysis that SFE constitutes vessel failure. We will use the Westinghouse 6:1 initial flaw geometry, since we judge that an infinitely long flaw is too conservative.

The Diablo Canyon end-of-life RT_{PTS} values shown in Table L-1 are based on (1) the initial, unirradiated RT_{NDT} value; (2) the amount of copper and nickel in the weld; (3) the end-of-life fluence; and (4) a margin mandated by the NRC to account for uncertainties in these values. Since we are evaluating the mean split fractions for Top Event VI, we do not want to include this margin in the point estimate quantification (uncertainties will be accounted for in those PTS scenarios that are risk significant). Thus, the end-of-life RT_{NDT} value is evaluated as

$$RT_{NDT}, EOL = RT_{PTS} - M$$

For the Unit 1 vessel, the limiting RT_{NDT} value is (217 - 59 =) 158°F, and for the Unit 2 vessel, the limiting value is (228 - 48 =) 180°F. Note that both RT_{PTS} values are below the NRC screening criteria and both RT_{NDT} values are below the lowest value analyzed in the WOG study; i.e., 200°F. The approach that will be used in this analysis will be as follows.

1. Select a representative scenario (i.e., PTS bin) for each Diablo Canyon PRA initiating event category whose plant model has a VI top event (for the SGTR event, different split fractions will be evaluated based upon the timeliness of operator actions to terminate safety injection as directed in procedures).

2. Evaluate the conditional frequency of significant flaw extension (Note: this is per longitudinal weld) at 200°F and 250°F RT_{NDT} values by dividing the tabulated value by the bin frequency and multiplying by the ratio of the cumulative NRC to Westinghouse frequencies to account for the finite flaw model.

3. Extrapolate the conditional frequencies of SFE in step 2 down to the limiting RT_{NDT} values for the Diablo Canyon Units 1 and 2 reactor vessels. This will be a linear extrapolation of the log of frequency with temperature evaluated as follows:

$$CF_{W,X} = 10^{[\alpha \cdot \log_{10} CF_{W,200°F} - (\alpha - 1) \times \log_{10} CF_{W,250°F}]}$$

where

$CF_{W,X}$ = conditional frequency of significant flaw extension per weld at an RT_{NDT} temperature of X°F

α = temperature extrapolation factor

= (250 - X)/(250 - 200)

= 1.84 for X = 158°F (RT_{NDT} for vessel 1)

= 1.40 for X = 180°F (RT_{NDT} for vessel 2)

4. Obtain the conditional frequency of vessel failure at temperature X ($CF_{V,X}$) by multiplying the per weld value by six welds as was done in the WOG analysis, or

$$CF_{V,X} = 6 \times CF_{W,X}$$

For the Unit 1 vessel, which is limited by longitudinal welds (see Table L-1), the six multiplier is probably too large by a factor of almost 2 since the fluence level in the intermediate shell welds is substantially lower. This benefit will be neglected for this evaluation. For the Unit 2 vessel, two of the three intermediate shell plates are limiting; the factor of 6 will be used, since we really do not have a basis to do otherwise. Let us now evaluate the VI split fractions for the different initiating event categories.

L.4 SECONDARY DEPRESSURIZATION EVENTS (SPLIT FRACTION VI1)

From the WOG study, 28 secondary depressurization categories were analyzed for three different secondary side break sizes (< 0.11 ft^2, 0.11 to 0.33 ft^2, and > 0.33 ft^2), for three decay heat levels (> 1%, 0.5 to 1%, and < 0.5%), and for three different times for terminating auxiliary feedwater flow to the faulted steam generator(s); i.e., 10 minutes, 20 minutes, and 60 minutes. The results of these analyses are shown in Table IV.3-1 of Reference L-4. There is negligible PTS risk if the core decay heat is greater than 1%. The largest conditional vessel failure frequency occurs for very low decay heat levels (< 0.5%), is generally insensitive to the time feedwater flow is terminated, and of course occurs for the largest equivalent unisolated leak area. The DCPRA transient event tree models the potential PTS challenge when the reactor trips, but the turbine fails to trip and the MSIVs fail to close; i.e, the most severe secondary depressurization event. As a bounding case, the event is assumed to occur when the reactor has not been at power long. Thus, the following case (low decay heat, large leak) will be analyzed:

Scenario	CF_W for RT_{NDT} (°F) Value:			
	158°F	180°F	200°F	250°F
SD-DH3-S3-L2-OA1	3.2×10^{-6}	1.9×10^{-5}	$3 \times 10^{-4} \times 0.31$	$.01 \times 0.51$

The first numbers under the 200°F and 250°F headings are the conditional frequencies using the NRC continuous flaw model. The second number is the correction to account for the WOG finite flaw model. Correcting for six longitudinal welds results in:

$CF_{V,SD,180°F} = 1.1 \times 10^{-4}$/challenge (Unit 2)

$CF_{V,SD,158°F} = 1.9 \times 10^{-5}$/challenge (Unit 1)

L.5 LOSS OF SECONDARY HEAT SINK (SPLIT FRACTION VI2)

From Table IV.3-4 of the WOG study, only one loss of secondary heat sink scenario was evaluated. These scenarios involve eventual steam generator dryout, with the operators establishing bleed and feed cooling with essentially stagnant loops (no natural convection or reactor coolant pump flow). The conditional frequencies of significant flaw extension for the entire vessel using the Westinghouse finite crack model are 5.6 x 10^{-2} and 0.6 for 200°F and 250°F RT_{NDT} values, respectively. Extrapolating down to the 158°F and 180°F temperatures gives

$CF_{V,LOHS,180°F}$ = 2.2 x 10^{-2}/challenge (Unit 2)
$CF_{V,LOHS,158°F}$ = 7.6 x 10^{-3}/challenge (Unit 1)

L.6 LOSS OF COOLANT ACCIDENT (SPLIT FRACTION VI3)

The WOG analysis of LOCA events considers three break size ranges:

- LOCA-S1 < 1.5" diameter
- LOCA-S2 1.5" < diameter < 6"
- LOCA-S3 diameter > 6"

Referring to Table IV.3-2 of the WOG Study, the highest PTS risk comes from LOCAs in the 1.5-inch to 6-inch diameter range. This is to be expected since the rate of cooldown for smaller breaks is slower and the repressurization level for the larger breaks was low, so the worst condition is in the intermediate, or medium LOCA range. Analyses were done for three decay heat levels; the PTS risk increases with decreasing decay heat because loop stagnation occurs sooner. We will use the intermediate decay heat value; i.e., between 0.5% and 1%, corresponding to the event occurring between 2 and 20 hours after shutdown, or conversely, shortly after the reactor has started up. The conditional frequency of significant flaw extension for an individual longitudinal weld from the NRC analyses and the Westinghouse correction factors are noted below:

Scenario	CF_W for RT_{NDT} Value:			
	158°F	180°F	200°F	250°F
LOCA-DH2-S2-LI-OP1	7.9×10^{-5}	3.4×10^{-4}	0.30 x .042	0.196 x 0.174

Correcting for six longitudinal welds results in:

$CF_{V,LOHS,180°F}$ = 2.0 x 10-3/challenge (Unit 2)

$CF_{V,LOHS,158°F}$ = 4.7 x 10-4/challenge (Unit 1)

L.7 STEAM GENERATOR TUBE RUPTURE (SPLIT FRACTIONS VI4 AND VI5)

A few aspects of a SGTR event are worth mentioning in order to put the overall analysis in perspective.

First of all, a steam generator tube failure by itself will not result in a rapid cooldown of the primary system or in an excessively high reactor coolant system pressure. Furthermore, natural circulation will develop in all primary loops and mix with incoming safety injection flow to preclude local temperature depressions even if the RCPs are stopped. Analyses presented in the WOG study indicate that if tube leakage is less than about 650 gpm (which is indicative of several tube breaks and larger than the five SGTR events that have occurred prior to December 1982), the RCPs will not be tripped if the operators follow the ERG guidelines. However, if the RCPs are tripped, the subsequent operator actions to terminate primary-to-secondary system leakage may rapidly cool the reactor coolant system for short periods and may stagnate the faulted loop. In that case, local temperature depressions resulting from continued safety injection flow may occur.

The two VI split fractions to be evaluated for the SGTR event are (1) VI4 for the case when the operator quickly (i.e., within 3 minutes) terminates SI as directed by procedures and (2) VI5 for the delayed termination case.

From Table IV.3-3 of the WOG study, 12 SGTR scenarios are evaluated. The first 6 have the reactor coolant pumps continue operating and result in negligible PTS risk, since the loop flow mixes the cold injected water. For cases wherein the RCPs stop and the operator follows procedures on isolating the secondary side of the faulted steam generator, WOG analyses predict loop stagnation even for a single tube failure. The last 6 cases have the RCPs off and consider variations in core decay heat levels and whether or not the operators terminate SI (as directed in the procedures) in a timely manner. The conditional frequency of vessel failure (CF_V) increases with decreasing decay heat levels. We will use the intermediate decay heat values (between 0.5 and 1% of rated power, corresponding to shutdown times between about 2 to 20 hours). The conditional frequency of longitudinal weld failure (CF_W) values using the NRC continuous flaw model are presented below, along with the multiplier to account for the Westinghouse finite (6:1 aspect ratio) flaw:

Scenario	CF_W for RT_{NDT} Value:			
	158°F	180°F	200°F	250°F
SGTR-DH2-S1-OSI1-OR2	5.1^{-8}	3.0^{-7}	1^{-5} x .147	4.1^{-4} x .197
SGTR-DH2-S1-OSI2-OR2	7.5^{-4}	1.5^{-3}	.019 x .147	.068 x .197

Note: Exponential notation is indicated in abbreviated form, i.e., $5.1^{-8} = 5.1 \times 10^{-8}$.

The first scenario has the RCPs tripped, intermediate decay heat, and the operators terminating SI within 3 minutes after meeting the termination criteria. The second scenario is the same except for delayed (i.e., within 1 hour) SI termination. Whether or not SI termination is performed in a timely manner is tracked by Top Event OP in the event tree model for steam generator tube ruptures. Extrapolating down to the lower RT_{NDT} values and correcting for the six welds results in the following case for timely SI termination (VI4):

$CF_{V,SGTR,180°F,timely} = 1.8 \times 10^{-6}$/challenge (Unit 2)

$CF_{V,SGTR,158°F,delayed} = 3.1 \times 10^{-7}$/challenge (Unit 1)

For delayed SI termination (VI5):

$CF_{V,SGTR,180°F,delayed} = 9.0 \times 10^{-3}$/challenge (Unit 2)

$CF_{V,SGTR,158°F,delayed} = 4.5 \times 10^{-3}$/challenge (Unit 1)

L.8 REFERENCES

L-1. U.S. Nuclear Regulatory Commission, "Analysis of Potential Pressurized Thermal Shock Events," 10CFR Part 50, Final Rule, Federal Register, Vol. 50, No. 141, pp. 29937-29945, July 23, 1985.

L-2. ASME Boiler and Pressure Vessel Code, Section III, Paragraph NB-2331.

L-3. Sullivan, M. D., "Evaluation of Diablo Canyon Power Plant Reactor Vessel Materials by the NRC Pressurized Thermal Shock Screening Criteria," PGandE Report 420-85.687, January 13, 1986.

L-4. Cheung, A. C., K. R. Balkey, D. S. Ackerson, T. S. Andreychak, D. R. Sharp, and V. V. Subramaniam, "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, from Pressurized Thermal Shock of Reactor Vessels of Westinghouse Nuclear Power Plants," performed by Westinghouse for the Westinghouse Owners Group, WCAP-10319, December 1983.

L-5. U.S. Nuclear Regulatory Commission, "NRC Staff Evaluation of Pressurized Thermal Shock," SECY-82-465, Enclosure A, November 1982.

L-6. U.S. Nuclear Regulatory Commission, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Regulatory Guide 1.99, Revision 1, April 1977.

L-7. Letter from R. Thierry (PGandE) to File No. 1245, 1243.2.6, Summary of Conference Call on December 8, 1987, between D. Wakefield and D. Buttemer (PLG), M. Sullivan and R. Thierry (PGandE), and K. Balkey and T. Meyers (Westinghouse), dated February 25, 1988.

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TABLE L-1. EVALUATION OF END-OF-LIFE RTPTS FOR UNITS 1 AND 2

Vessel Material	Initial RT_{NDT} (°F)	Margin M (°F)	Composition (%) (Cu)	Composition (%) (Ni)	EOL Fluence ($n/cm^2 \times 10^{19}$)	RT_{PTS} (°F)	Fluence* Margin (%)
Unit 1							
o Lower Shell, Longitudinal Welds	-56**	59	0.21	0.98	2.9	217	56
o Intermediate Shell, Longitudinal Welds	-56**	59	0.21	0.98	2.0	197	69
o All Others						$<$ 166	$>$ 90
Unit 2							
o Intermediate Shell Plates:							
- 1	+52	48	0.15	0.62	2.9	224	69
- 2	+67	48	0.14	0.59	2.9	228	69
- 3	+33	48	0.15	0.62	2.9	205	79
o All Others						$<$ 185	$>$ 74

*Fluence margin (%) = $\frac{\text{Fluence to reach PTS limit - EOL fluence}}{\text{EOL fluence}} \times 100$.

**Generic value from PTS rule.

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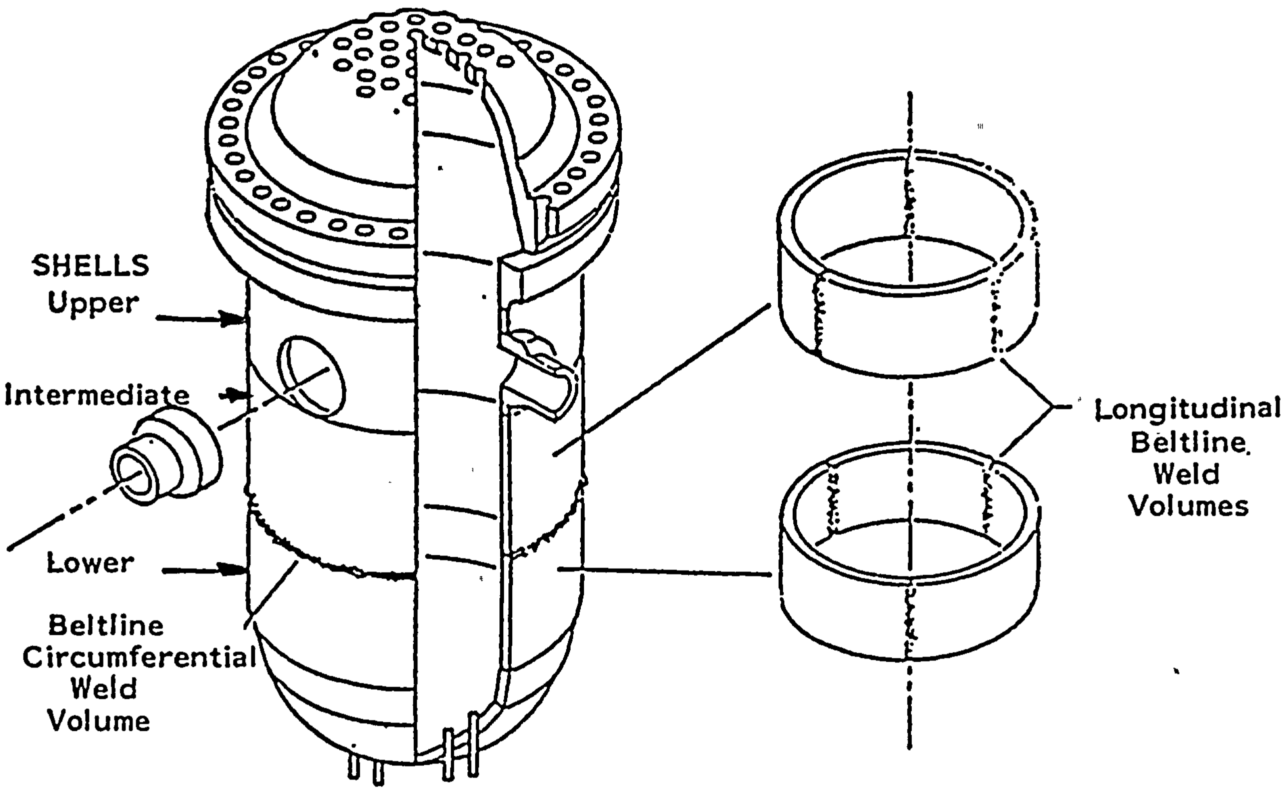

FIGURE L-1. DIABLO CANYON REACTOR VESSEL FABRICATION