COVER SHEET

SUMMARY OF THE MEETING BETWEEN THE NUCLEAR REGULATORY COMMISSION (NRC) STAFF AND INDUSTRY REPRESENTATIVES ON PERFORMING RESEARCH IN SUPPORT OF THE GOAL OF RISK INFORMING 10CFR50 APPENDIX G AND 10CFR50.61

Wednesday March 29, 2006
9:00 AM - 5:00 PM
One White Flint North (Tall Building)
Room O12-B6

This packet contains the following information, arranged in order after this cover page

1. Meeting agenda
2. Attendees list
3. Minutes
4. Copies of presentation slides

Questions / enquires should be directed to

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FINAL AGENDA

MEETING BETWEEN THE NUCLEAR REGULATORY COMMISSION (NRC) STAFF AND INDUSTRY REPRESENTATIVES ON PERFORMING RESEARCH IN SUPPORT OF THE GOAL OF RISK INFORMING 10CFR50 APPENDIX G AND 10CFR50.61

Wednesday March 29, 2006
9:00 AM - 5:00 PM
One White Flint North (Tall Building)
Room O12-B6

Time	Item#	Topic	Speaker
9:30 - 9:45	1	Welcome and Introductions	M. EricksonKirk
9:45 - 10:45	2	10CFR50.61 - PTS	
	2a	**Summary of RES/NRR Interactions on Tech Basis**	M. EricksonKirk
	2b	Planned NRR Rulemaking Schedule	R. Hardies
	2c	**Industry Comments**	B. Bishop
10:45 - 11:40	**3**	**PFM work performed at ORNL**	T. Dickson
11:30 - 12:15	**4**	**PFM work performed by industry**	R. Gamble
12:15 - 1:15	5	Lunch	
1:15 – 1:30	**6**	**BWR considerations relative to Appendix G**	R. Carter
1:30 – 2:15	**7**	**Use of new NRC embrittlement trend curve for plant Evaluation, and attenuation**	R. Lott W. Server
2:15 – 3:30	**8**	**Design, procedural, physical limits on the rate at which a plant can be heated or cooled & event frequencies**	B. Bishop
3:30 – 4:30	9	General discussion, identification of program goals	M. EricksonKirk D. Weakland

Note: Agenda items appearing in **highlighted bold** have presentations associated with them. The presentation slides are attached in order after meeting minutes

ATTENDEE LIST

Name	Company	E-mail
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Jim Riley	NEI	jhr@nei.org

MEETING MINUTES

1. Mr. EricksonKirk opened the meeting, welcoming all participants. All in attendance introduced themselves and an attendance sheet was circulated. A list of those attending appears on p. 3 of this document.
2. Mr. EricksonKirk stated the following objectives for the meeting:
    a. Facilitate exchange of information between NRC and industry researchers working on projects to risk-inform 10CFR50 Appendix G. Identify areas of common interest where information sharing would be beneficial
    b. Discuss 10CFR50.61 rulemaking
3. The meeting agenda, which appears on p. 2 of this document, was finalized.
4. **Pressurized Thermal Shock (Item 2)**: Slides presented by Mr. EricksonKirk and by Mr. Bishop appear after these minutes
    a. Points from Mr. EricksonKirk's presentation
        i. Slides are attached. These slides provide information on questions asked by NRR, on changes being made to the PFM code FAVOR, and they detail dates for public comments on the PTS technical basis and for technical basis finalization.
        ii. The FAVOR 06.1 change specification would be made public as an appendix to the FAVOR 06.1 theory/users manuals.
        iii. RES's responses to NRR's fourteen questions would be made public as an Appendix to the final version of NUREG-1806, which will be issued in November 2006.
        iv. FAVOR 06.1 will be available for verification and validation work in May 2006.
        v. Mr. Gamble pointed out that the last industry V&V of FAVOR was performed on Version 04.1. In order for the industry to perform a V&V on FAVOR Version 06.1 the NRC needs to issue manuals for both versions 05.1 and 06.1. Mr. EricksonKirk committed to make both manuals available when FAVOR 06.1 is ready for V&V.
    b. Points from Mr. Hardies (NRC/NRR) presentation:
        i. In the Summer of 2005 a group of NRR staff reviewed the technical basis developed by RES. This review indicated that there is generally basis to proceed with rulemaking, but identified 14 issues on which more information and/or clarification was needed. In September 2005 NRR's Rulemaking Advisory Board authorized the staff to begin rulemaking in parallel with RES addressing NRR's 14 issues.
        ii. Rulemaking is a 3-step process.
        iii. The first step takes about half a year. During this step the NRR staff prepares a letter requesting Commission approval to go into rulemaking. It is anticipated that the letter will be sent to the Commission in the next few weeks. If the Commission approves rulemaking then Step 2 begins. The Commission could require the staff to address NRR's 14 issues before Step 2 begins.
        iv. The second step takes about a year and involves creation by NRR staff of a draft rule. Before the second step can be completed the 14 issues with the technical basis identified by NRR need to be resolved by RES. Currently, work by RES to address NRR's 14 issues is not on critical path for Step 2.

v. The third step takes approximately a year to complete. It involves publication of the draft rule for public comment, resolution of public comments, and finalization of the rule.

c. Points from Mr. Bishop's presentation:

   i. Slides are attached. These slides concern use of information from NUREG-1806 to determine PTS acceptance limits based on fluence rather than based on reference temperature.

   ii. A tiered approach is proposed. Vendor-specific fluence limits are established at a 10^{-7}/reactor year TWCF limit. If these limits are passed then a plant could use the RT-based limits, which are established in NUREG-1806 at a 10^{-6}/reactor year TWCF limit. The benefits of this approach are seen to be as follows:

      1. Plant operators can control fluence directly. Expressing the PTS limits in terms of something that can be controlled by the operator makes the limits more understandable.
      2. Since most plants are not close to the 10^{-6}/reactor year TWCF limit, the detailed RT calculation by the plants, and review by the NRC, could be avoided in most cases.

   iii. Mr. Elliott raised the concern that the proposed tiered evaluation approach could have the undesired consequence of creating the impression that certain plants are somehow safer than others.

5. **<u>Probabilistic Fracture Mechanics Calculations to Risk-Inform Appendix G (Items 3&4)</u>**

   a. <u>Item 3</u>. Mr. Dickson presented slides (attached) summarizing recent work performed at the Oak Ridge National Laboratory to assess the risk posed by scheduled 100°F/hr cooldown and heatup transients. Major points from Mr. Dickson's presentation, and the related discussion, include the following:

      i. The majority of vessel failure probability associated with cool-down transients occurs because the ASME code now allows operators to maintain full system pressure for a long time after the beginning of the transient. In actual operating practice pressure is always ramped down along with temperature, or with a very short phase-lag relative to temperature,

      ii. Another conservatism in the current analysis of cooldown transients is that the linear cool-down rate is assumed to continue until ambient temperature conditions are achieved. In actual operating practice switchover to shutdown cooling occurs at a primary system temperature of approximately 300°F. After switchover the maximum cooling rate is limited by the capacity of the heat exchanger to maximum rates that are considerably slower than 100°F/hr.

      iii. Limited scoping calculations performed on heatup indicate that cool-down will be the limiting condition by several orders of magnitude in vessel failure probability.

         1. Mr. Bishop commented that this finding is consistent with deterministic calculations performed by Westinghouse some years ago. Additionally, Mr. Bishop and others from the industry noted that current hardware limitations make it almost impossible to achieve the maximum allowed cooling rate (100°F/hr) in practice.
         2. Mr. EricksonKirk noted, and Mr. Weakland concurred, that there may be value in developing a technical case for why cool-down

will always be limiting, thereby reducing the scope and complexity of future calculations.

b. Item 4: Mr. Gamble presented slides, attached, detailing the industry's plan for work on 10CFR50 Appendix G.
    i. This work is initiating in this fiscal year and will continue through mid 2008. The planned project includes modification of the ASME code.
    ii. The industry effort plans to use the ORNL FAVOR code as a computational tool.
    iii. The proposed approach is to replace the ASME margin of 2 on pressure with a "risk informed margin" (RIM). The value of RIM would be established such that the crack initiation frequency estimated by FAVOR would equal 10^{-6}/reactor year. Values of RIM would range from 1 to 2.

6. **<u>BWR Considerations (Item 6)</u>**: Mr. Carter presented the attached slide. He pointed out that the major issue for BWRs is not heatup or cooldown rates, but rather the potential need to heat the RCS water to perform a leak test. Mr. Carter reminded those in attendance that 10CFR50 Appendix G addresses a broader scope of issues than just heat up and cool-down limits.
7. **<u>Use of the new NRC embrittlement trend curve for plant evaluations, and attenuation (Item 7)</u>**: Messrs. Lott and Server made a presentation (slides attached). Major points from the presentation, and related discussion, were as follows:
    a. New NRC embrittlement trend curve (Eason 1-06) increases RT_{NDT} at EOL by approximately 25°F relative to Reg Guide 1.99 Revision 2 values.
    b. Sensitivity studies show that RT_{NDT} prediction is most sensitive to changes in Cu and Ni values. Changes in all other input values (P, Mn, fluence, coolant temperature) have much smaller effects on RT_{NDT}.
    c. Preliminary results from on-going (IAEA) attenuation study indicate that the Reg. Guide 1.99 Revision 2 attenuation function is a conservative (under) predictor of attenuation.
8. **<u>Design Limits on heatup and cooldown (Item 8)</u>**: Mr. Bishop presented slides concerning a study performed by Westinghouse under EPRI sponsorship. This study examined both the structural and operational flexibility of relaxing Appendix G limits on heat up and cool down. The outcome of this study is documented in EPRI MRP Report # 143, which is available from Mr. Spanner. Major points from the presentation, and related discussion, were as follows:
    a. A cool-down rate of up to 200°F/hr was achieved using the Westinghouse simulator, but the ability to maintain control over plant chemistry during such a rapid cool down remains to be demonstrated.
    b. An event frequency of 1 to 2 cooldown events per year was estimated.
    c. Both realistic and hypothetical transients were modeled and used as input to FAVOR. The plant analyzed was Beaver Valley Unit 1 at 60 EFPY, for which the maximum RT_{NDT} is 332°F (this value includes a RG1.99R2 margin term of 42°F). These transients represented the following conditions:
        i. Initial delays between start of cooling and start of depressurization of 20, 50, and 80 minutes. It was noted that while any delay between cooling and depressurization constituted a departure from the design basis it was, nevertheless, a short-term occurrence at a few Westinghouse plants.
        ii. Cooling rates of up to 200°F/hr between the operating temperature and temperature (i.e., 350°F) at which switchover must be made to the residual heat recovery (RHR) system to continue cooling.
        iii. Hold times at 350°F of at least 30 minutes.

iv. Cooling rates between 350°F and ambient temperature in excess of the design limit of 100°F/hr.

These analyses demonstrated that in order to get non-zero estimates of the yearly probability of crack initiation, conditions that are either unlikely or impossible to achieve in service had to be assumed. Specifically, an 80 minute delay between the start of cooling and start of depressurization coupled with a cooling rate below 350°F and ambient temperature in excess of the design limit were needed.

9. **General Discussion / Actions (Item 9)**: The following items were discussed:
    a. Mr. Kirk committed to release both the FAVOR 05.1 and FAVOR 06.1 manuals together to facilitate industry V&V of the FAVOR code. Additionally, Mr. Kirk promised that these manuals would include information detailing the changes made to these versions of the FAVOR code relative to previous versions.
    b. Mr. Kirk commented that current schedules suggest that the NUREG/CR documenting the technical basis for the NRC's new embrittlement trend curve (that was presented at the ASTM E10.02 meeting held in February 2006 in Phoenix Arizona) will be released for public comment sometime in May 2006.
    c. Mr. Kirk stated that the FAVOR heatup code would be placed in the public domain once theory and users manuals were completed and reviewed by ORNL.
    d. All agreed that there appeared to be a technical basis for treating cool-down transients as limiting, thereby eliminating the need for performing further detailed heat-up calculations. Mr. Weakland and Mr. Kirk agreed that they would investigate the possibility of including such work in their on-going projects.
    e. Mr. Carter committed to provide Mr. Kirk with a copy of the BWRVIP report documenting the Appendix G studies performed by SARTREX/ATI.
    f. Mr. Spanner committed to provide Mr. Kirk with a copy of the EPRI/MRP program plan for Appendix G work. This plan will be issued on 31st March 2006 as a NEPO report.
    g. Mr. Kirk committed to provide Mr. Spanner with an updated / detailed plan for NRC/ORNL work on Appendix G.
    h. The date and venue for the next meeting was established as follows: 30th August 2006 at USNRC Headquarters in Rockville, Maryland. Mr. Kirk committed to get a bigger meeting room for the 30th August meeting.

Summary of RES / NRR Interactions on Tech Basis for PTS Rule Revision

Mark EricksonKirk

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Materials Engineering Directorate
Division of Fuel, Engineering, and Radiological Research
Office of Nuclear Regulatory Research

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Public Meeting
29th March 2006 – Rockville, MD

History

- **30 Jun 05: RES provides tech basis to NRR for review**
- **12 Oct 05: NRR completes review, asks 14 questions, requests that tech basis reports be made available for public comment**
- **Imminent (1-ish Apr 06)**
  - **RES replies to NRR questions**
  - **RES specifies changes to FAVOR to address NRR questions**

Future – FAVOR Re-Runs

- A = Apr-May 06: FAVOR 06.1 available for V&V
- A + 2 months: V&V complete
- A + 3 months: FAVOR revised per V&V, re-runs begin
  - Baseline analysis of Oconee, Beaver Valley, Palisades
  - Sensitivity analysis of sub-clad cracks
- Mid-August 06: ORNL provides FAVOR results to RES
- Sep 06: RES modifies tech basis reports based on new FAVOR results & public comments, provides to NRR
- Nov 06: RES finalizes revised tech basis reports

Future – Public Comment on PTS Reports

- **Apr 06: Federal Register Notice (FRN) places tech basis documents into public domain, 60 day public comment period opens**
- **Jun 06: 60 day public comment period closes**
- **Jul – Aug 06: Public comments resolved**

14 NRR Questions

A. Statistical process for bounding TWCF results to develop RT-screening limits

B. Move sampling of RT_{NDT} epistemic uncertainty outside of flaw loop (inside vessel loop)

C. Sampling of $RT_{NDT(u)}$ uncertainty and Cu SD moved outside of flaw loop

D. Clarification of the insignificant effect of the small magnitude plumes that may exist on TWCF

E. Clarification of insignificant effect of TH uncertainty on TWCF

F. Clarification of the insignificant effect of gamma heating on TWCF

G. Change of repair weld model to account for higher probability of repair weld flaws occurring close to the ID or OD

H. Estimate of effect of sub-clad flaws on TWCF

14 NRR Questions

I. **Clarification that existing work addresses adequately non-PORV plants**

J. **Demonstration that the flaws that contribute to TWCF are detectable by NDE performed to ASME SC VIII Supplement 4 requirements**

K. **Demonstration that analysis of benign sequences cannot artificially lower TWCF**

L. **Clarification of explanation of mixing / natural circulation**

M. **Clarification of how dye/saltwater experiments reconcile with more recent integral system test results**

N. **Clarification that the interaction of axial and circumferential cracks does not need to be considered**

Other FAVOR Changes

Code changes

- New embrittlement trend curve
  - Mn uncertainty and sampling protocols
- Error in data basis for $RT_{NDT(u)}$ epistemic uncertainty corrected
- Coefficients in upper shelf model updated to reflect new data
- Output enhanced
- Temperature dependent thermal-elastic properties

Input changes

- LOCA break frequencies consistent with information in NUREG-1829
- Analysis of sub-clad flaws to establish RT-screening criteria for forged vessels
- Plant specific Mn values from RPVDATA

MRP-ITG Meeting with NRC

MRP ITG Thoughts on an Alternative PTS Rule Form

Bruce Bishop

Ted Meyer

(On behalf of the MRP RPVI ITG)

MRP ITG Thoughts on an Alternative PTS Rule Form

- Background
- Alternative PTS Rule Philosophy
- Alternative Criteria for the PTS Rule
  - Calculation of TWCF
- Why Change the PTS Rule Form
- Alternative Screening Criterion Fluence Limits Used for PTS Rule Comparison
- Technical Basis for Screening Criteria
  - Maximum RT_{NDT} at 40 Years
  - TWCF Results at 60 Years

Background

- Past ITG Chairman requested straw-man and side-by-side comparison of PTS rules by a Task Group at ITG meeting February 9-11, 2005
- Task Group met on March 17, 2005, proposed new concept and requested consistency with on-going NRC Research work
- ACRS slides and draft of SMIRT paper on latest proposed PTS screening criteria were used for consistency
- Summary of concept prepared for May 25th ITG meeting with NRC, but not presented because the topic was not on the public meeting notice
- NRR review of RES basis for PTS rule is ongoing, so results of Task Group action items are being presented to NRC now.

Alternative PTS Rule Philosophy

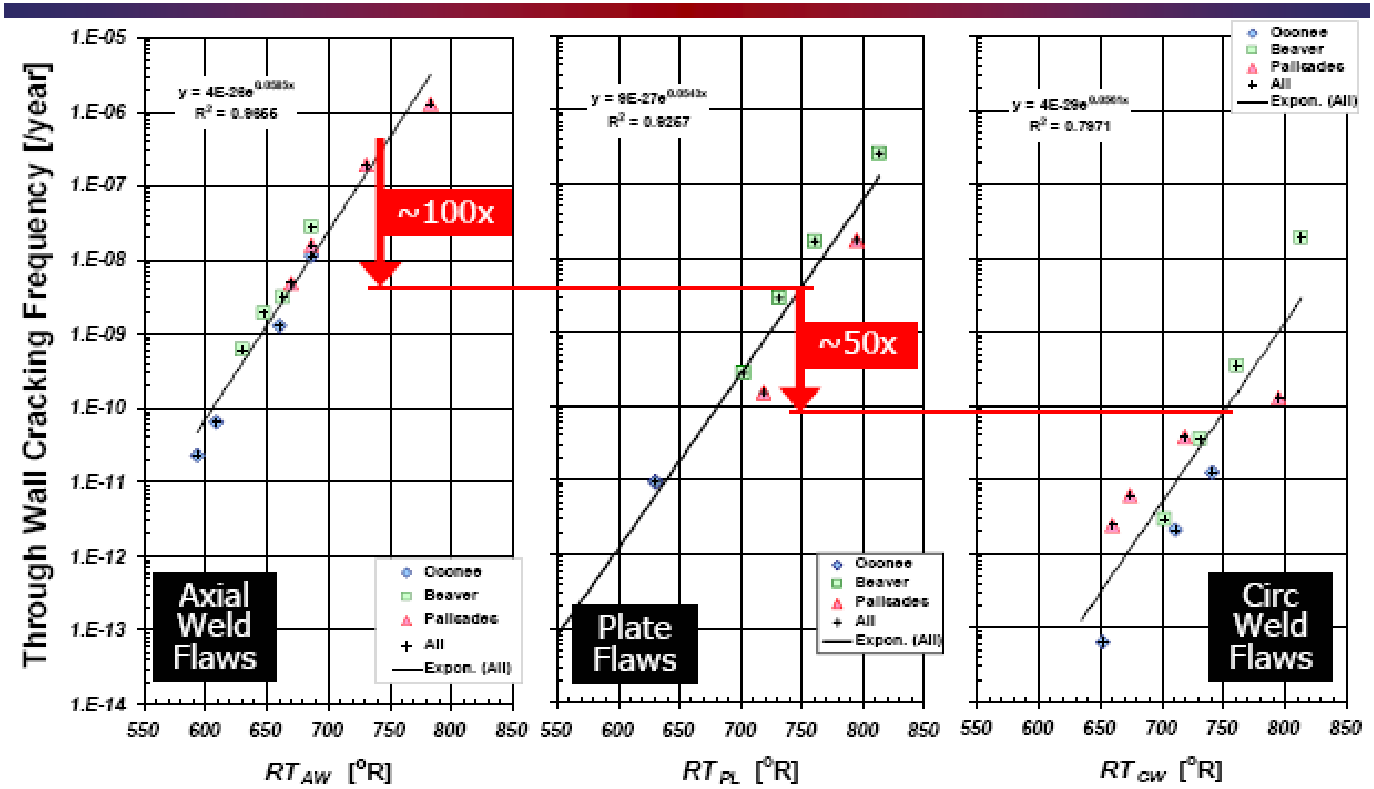
- Based upon detailed evaluation by NRC, PTS is not expected to be a significant risk contributor during first license extension (60 years)
- What is a risk informed and efficient way for NRC and utilities to interact on a new PTS rule
  - Very simple screening criteria for most plants
  - Deterministic calculation of TWCF based upon results of NRC Risk Study, if needed, to satisfy a risk criteria
  - Sensitivity studies using NRC risk models would be used for plant-specific changes to satisfy the risk criteria
  - No plant-specific detailed risk analyses, like those per R.G. 1.154, would ever be required

Alternative Criteria for the PTS Rule

- Define the "PTS Screening Criterion" as: The values of fluence for the vessel beltline components above which the plant cannot continue to operate without evaluation of the PTS Risk Criterion
  - Associated with a TWCF < 1E-7/year
    - This value provides additional margin above the PTS Risk Criterion
- Define the "PTS Risk Criterion" as: The value of TWCF for the vessel beltline material above which the plant cannot continue to operate without justification (sensitivity study on generic work)
  - Associated with a TWCF limit of 1E-6/year
    - This value is calculated based on RT_{MAX} for which all credible uncertainties have been explicitly addressed in the generic risk analysis (plant-specific risk study not required)

Calculation of TWCF

Materials Factors Controlling Vessel Failure



Why Change the PTS Rule Format

- Simplifies the PTS Rule (for the NRC and utilities)
- Enables utilities to actually manage the key parameter, i.e., fluence
  - Plant personnel that must address PTS can more readily relate to this parameter than RT_{NDT}
- Enables the utilities and the NRC to directly evaluate the impact of plant changes on this parameter, e.g.,
  - Uprating, Life Extension, Fuel Design

Alternative Screening Criterion Fluence Limits Used for PTS Rule Comparison

Maximum Fluence (10^{19} n/cm^2, E>1.0 Mev) for 60 EFPY

RV Beltline Component	B&W Plant Design	CE Plant Design	W Plant Design
Axial Weld	1.38	3.21	1.57
Circ. Weld	1.54	4.04	8.13
Plate or Forging	1.57	4.08	8.26

Technical Basis for Screening Criteria

- Differences in maximum RT_{NDT} are small among the lead plants for the 3 designs
- For a given plant design, maximum fluences at a given operating time are similar
- Fluence limits at 60 EFPY should bound most plants at license extension (60 years)
- Maximum TWCF at 60 years is more than an order of magnitude less than the PTS risk limit for LERF

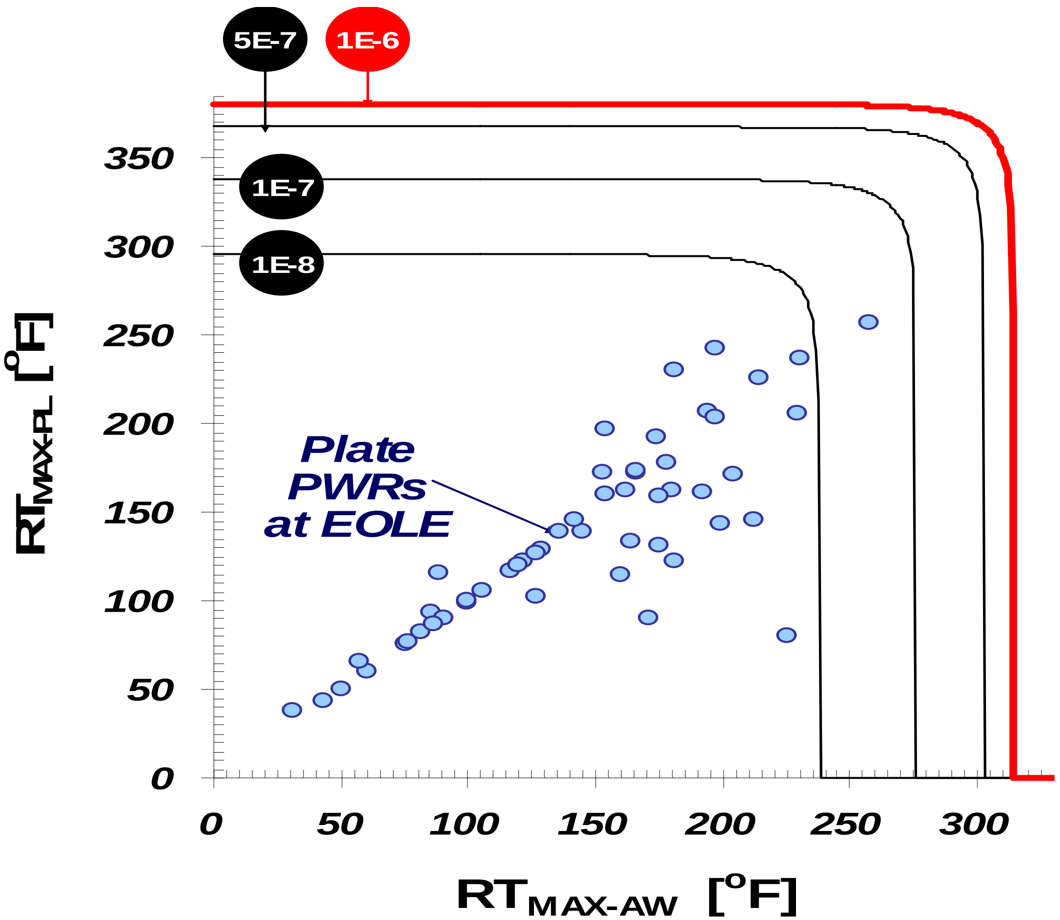
Maximum RT_{NDT} at 40 Years

Table 4.5. Plant list for generalization study.

Tolerance to a PTS Challenge		Plant Name	Most Embrittled Material	$RT_{NDT(u)}$ + Irradiation Shift at 40 years [°F]	Manufacturer
The estimated tolerance to a PTS challenge increases as the number in the next column increases.	1	SALEM 1	PLATE	204	Combustion Engineering
	2	BEAVER VALLEY 1	PLATE	194	Combustion Engineering
	3	TMI-1	A-WELD	186	Babcock & Wilcox
	4	FORT CALHOUN	A-WELD	181	Combustion Engineering
	5	PALISADES	A-WELD	179	Combustion Engineering
	6	CALVERT CLIFFS 1	A-WELD	178	Combustion Engineering
	7	DIABLO CANYON 1	A-WELD	171	Combustion Engineering
	8	DIABLO CANYON 2	PLATE	170	Combustion Engineering
	9	SEQUOYAH 1	FORGING	167	Rotterdam Dockyard
	10	WATTS BAR 1	FORGING	164	Rotterdam Dockyard
	11	ST. LUCIE 1	A-WELD	164	Combustion Engineering
	12	SURRY 1	A-WELD	163	Babcock & Wilcox
	13	INDIAN POINT 2	PLATE	162	Combustion Engineering
	14	GINNA	FORGING	161	Babcock & Wilcox
	15	POINT BEACH 1	A-WELD	159	Babcock & Wilcox
	16	FARLEY 2	PLATE	158	Combustion Engineering
	17	MCGUIRE 1	A-WELD	158	Combustion Engineering
	18	OCONEE 1	A-WELD	157	Babcock & Wilcox
	19	NORTH ANNA 2	FORGING	155	Rotterdam Dockyard
	20	SHEARON HARRIS	PLATE	153	Chicago Bridge & Iron
	21	NORTH ANNA 1	FORGING	153	Rotterdam Dockyard
	22	COOK 2	PLATE	152	Chicago Bridge & Iron
	23	SALEM 2	A-WELD	148	Combustion Engineering
	24	CRYSTAL RIVER 3	A-WELD	141	Babcock & Wilcox
	25	CALVERT CLIFFS 2	PLATE	139	Combustion Engineering
	26	ROBINSON 2	PLATE	138	Combustion Engineering
	27	COOK 1	A-WELD	138	Combustion Engineering
	28	FARLEY 1	PLATE	133	Combustion Engineering
	29	ARKANSAS NUCLEAR 1	A-WELD	129	Babcock & Wilcox

Notes: (a) Plants analyzed in the PTS re-evaluation effort are shown with a shaded background. (b) Circ welds have been omitted because while circ cracks initiate they do not (usually) lead to vessel failure.

TWCF Results at 60 Years



Reference

- Latest NRC PTS Research results from:
  - SMiRT18-D06-4, *Technical Basis For Revision Of The Pressurized Thermal Shock (PTS) Screening Limit In The PTS Rule (10CFR50.61)*, Mark T. EricksonKirk, et al., August 7-12, 2005

Review of ORNL Studies Regarding Risk-Informing ASME Section XI Appendix G

Terry Dickson

Computational Science and Engineering Division

Oak Ridge National Laboratory

NRC-Industry Meeting

March 29, 2006

NRC Headquarters

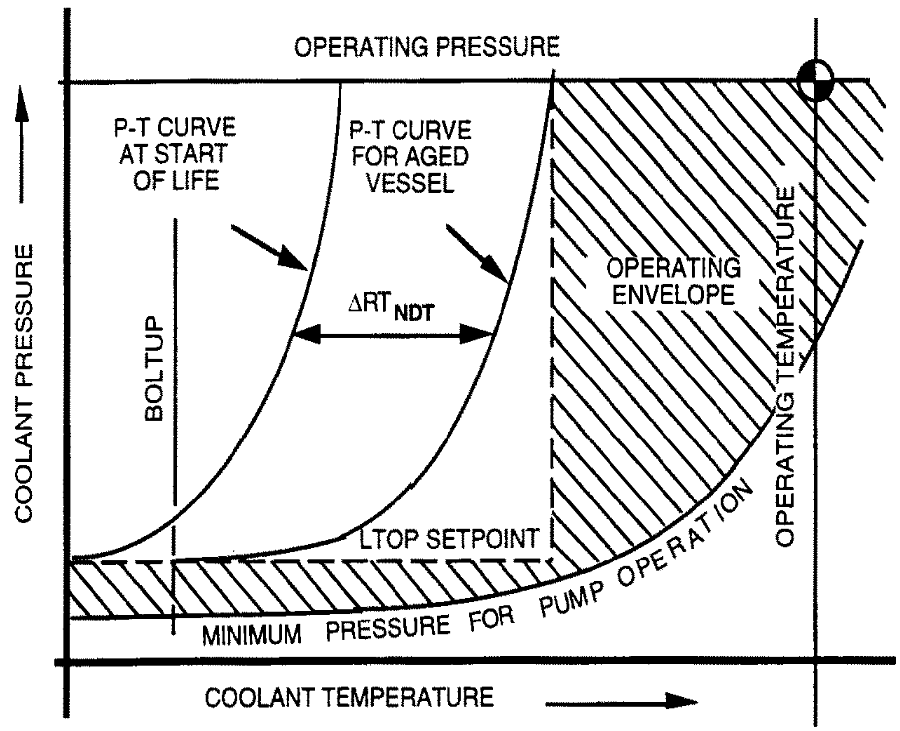
Rockville, Maryland

The P-T operating envelope is progressively restricted to accommodate the effects of irradiation embrittlement of the RPV material

The P-T curve controls the upper-bound to the permissible operating envelope for a RPV during normal start-up and cool-down transients

The P-T curve is currently derived using a prescriptive deterministic fracture methodology in ASME Section XI – Appendix G

An objective of ORNL study is to determine if a technical basis can be established to support a relaxation to the methodology in ASME Section XI – Appendix G

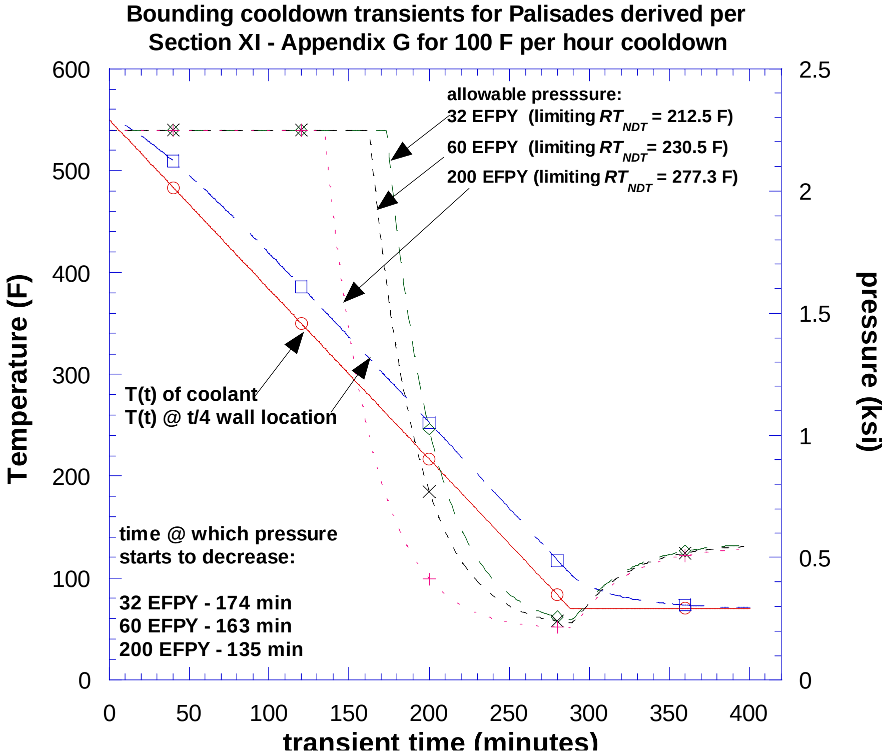


ORNL applied FAVOR in PFM scoping study in 2005 for risk-informing Section XI – Appendix G

Applied bounding cool-down transient (100 F / hr) with allowable pressure derived in accordance with the prescriptive deterministic methodology in Section XI – Appendix G

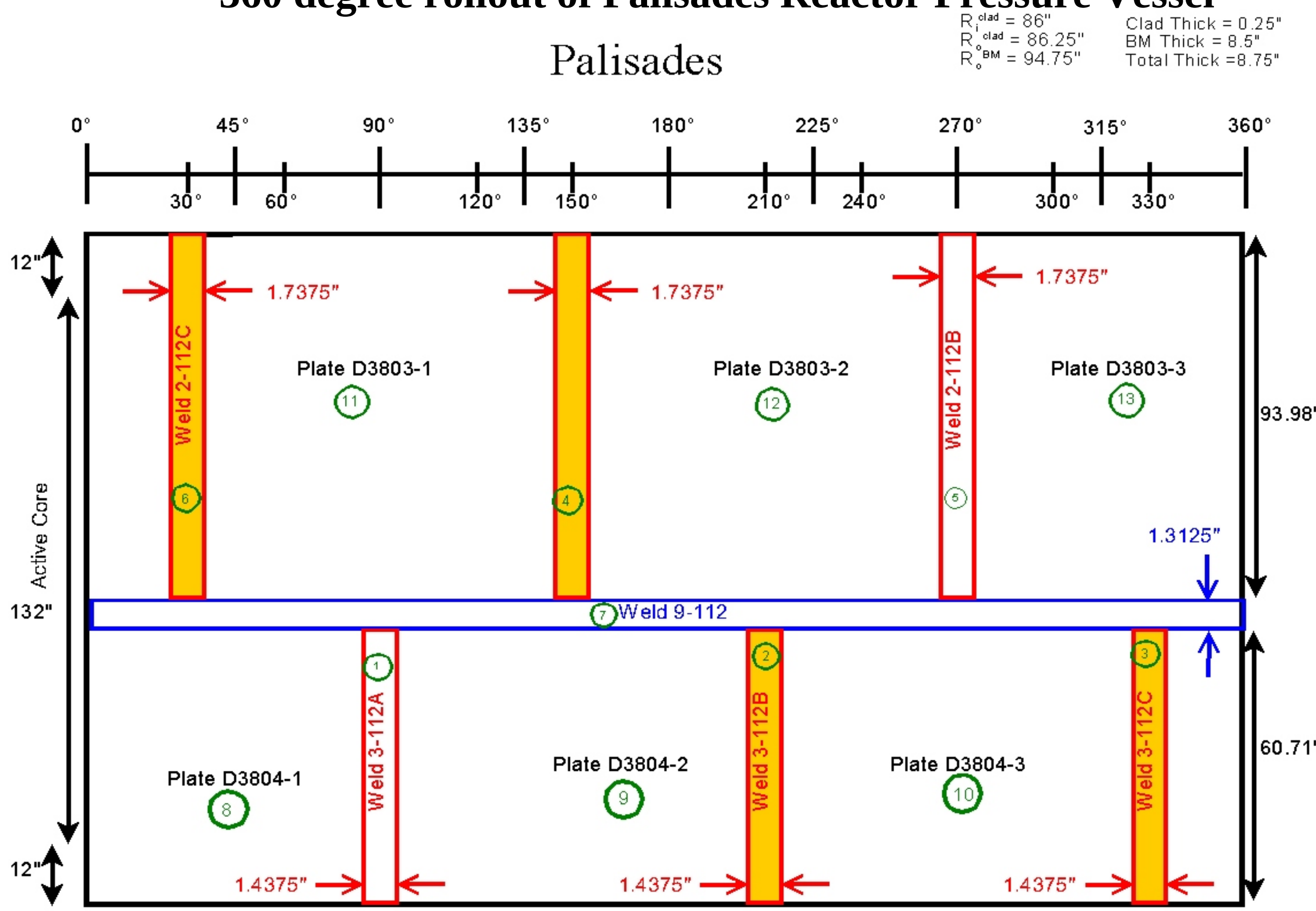
Analyses were performed for Palisades since, from the PTS re-evaluation, it was the most limiting plant

Utilized embrittlement and flaw characterizations from PTS re-evaluation



Scoping PFM analyses for normal operation transients associated with reactor startup and shutdown have been performed for Palisades since it was the most limiting RPV in the PTS re-evaluation (axial welds are the most highly embrittled RPV regions)

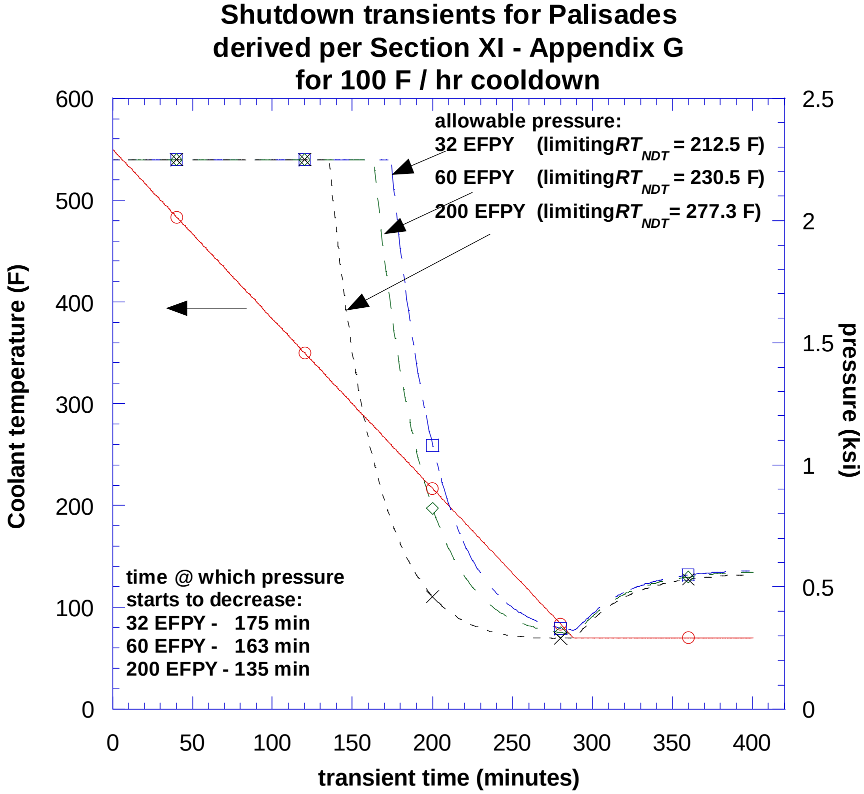
360 degree rollout of Palisades Reactor Pressure Vessel



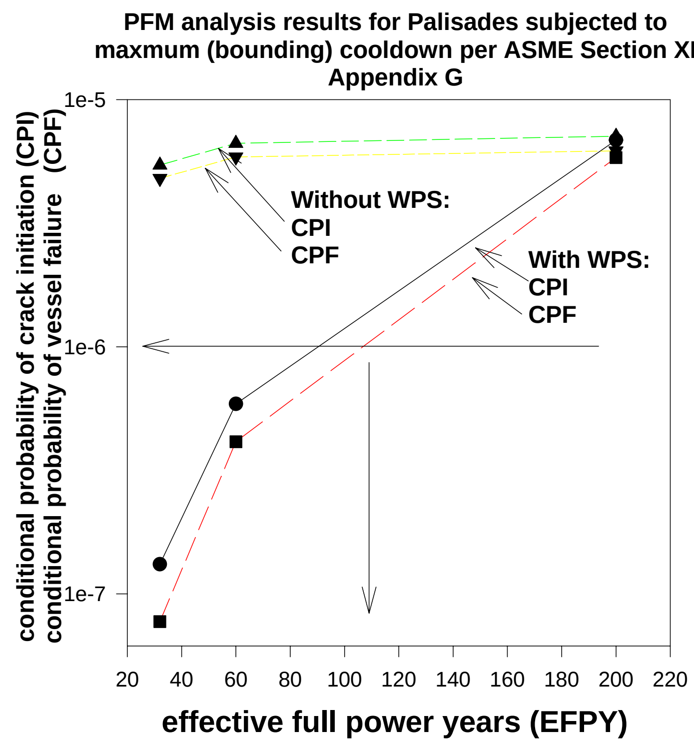
UT-BATTELLE

Scoping PFM analysis results for bounding cool-down transients are in compliance with proposed new acceptance criteria (for PTS) of 1.0e-6 failed RPVs per reactor operating year for over 60 EFPY (when WPS is included in the model)

Bounding cool-down transients for Palisades per Section – XI Appendix G



CPI and CPF computed with and without WPS



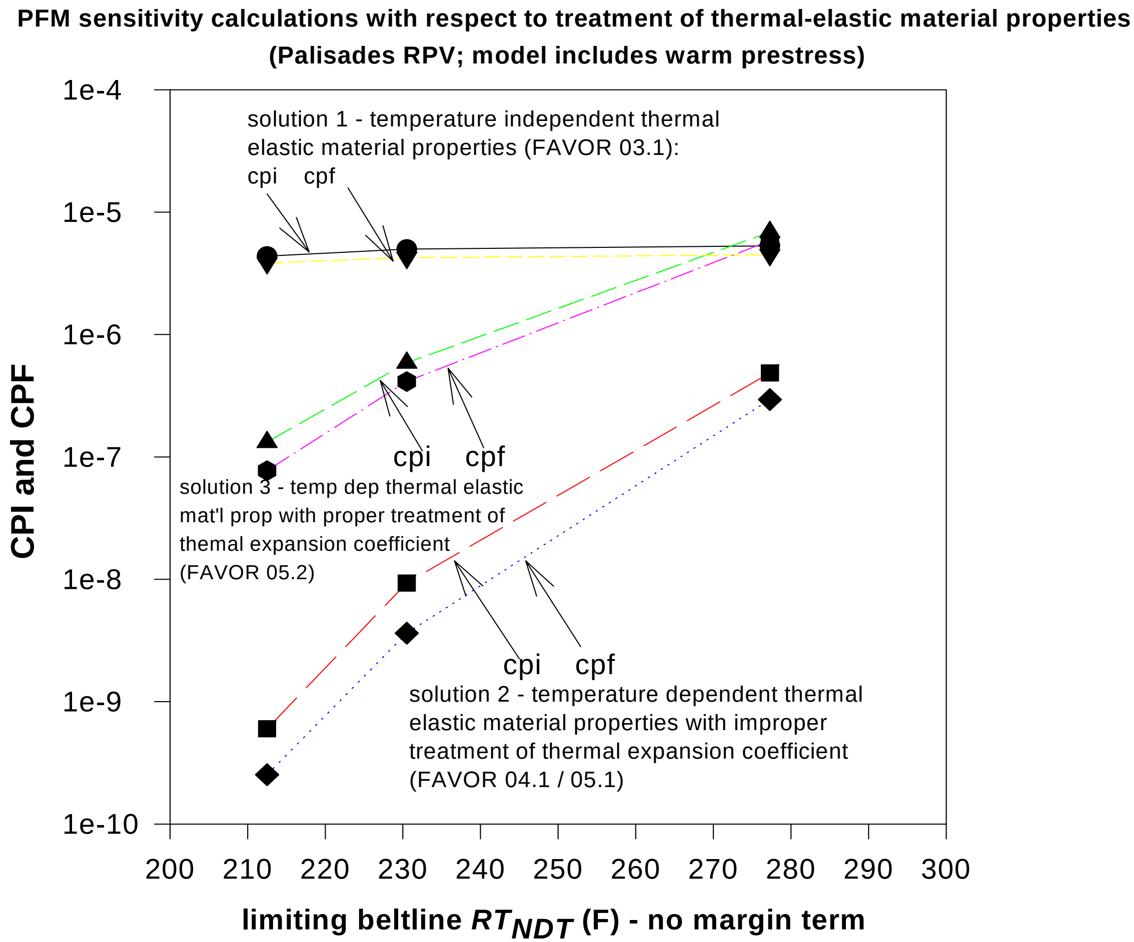
Results generated with FAVOR 05.2 (as of 10-20-05)

Oak Ridge National Laboratory
U.S. Department of Energy

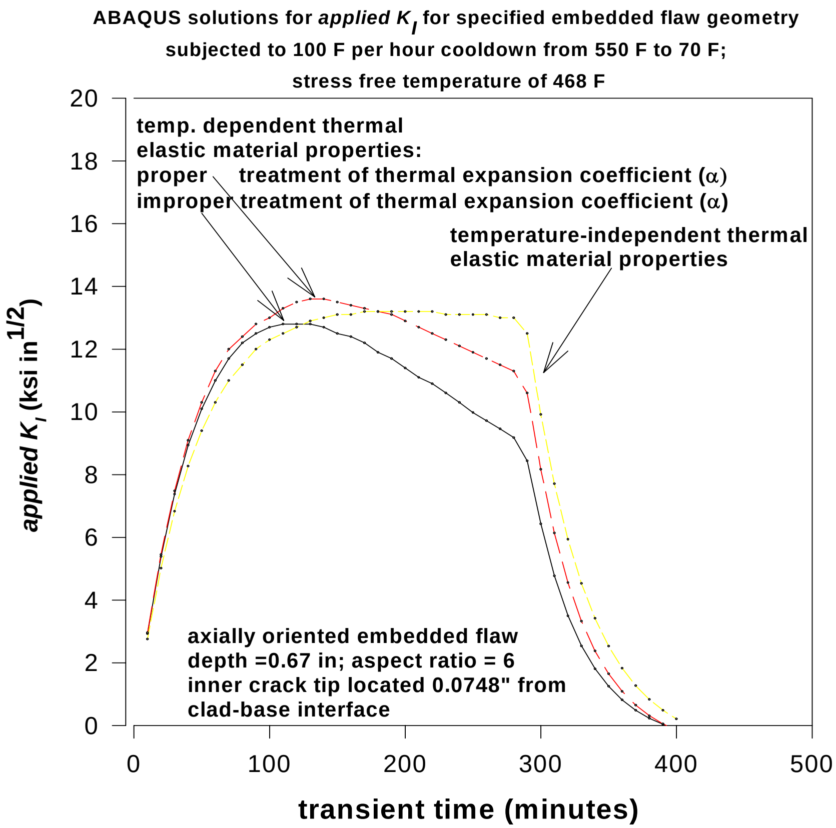
UT-BATTELLE

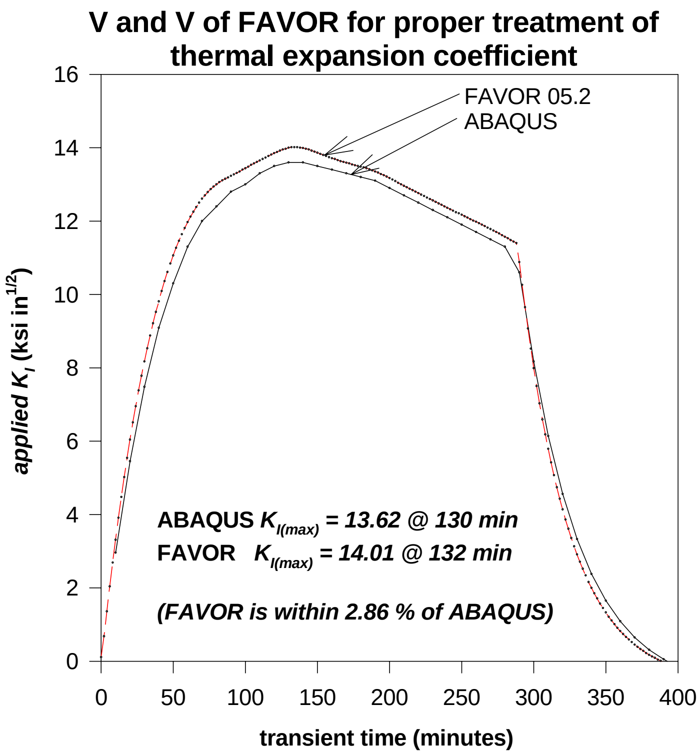
PFM analysis results for bounding cool-down transients are very sensitive to the treatment of thermal-elastic material properties

Application of temperature-dependent thermal elastic material properties, in conjunction with WPS, result in considerably smaller probabilities



Temperature-dependent thermal-elastic material properties has little impact on magnitude of peak loading; however, causes peak to occur at an earlier time, which in conjunction with WPS, can have significant impact on fracture analysis of flaw





Possible approaches to risk informing Section XI – Appendix G

(I) Provide technical basis for providing a relaxation to the current prescriptive deterministic method, such as:

(a) Remove factor of 2 in derivation of acceptable pressure
(b) Modification of reference flaw size

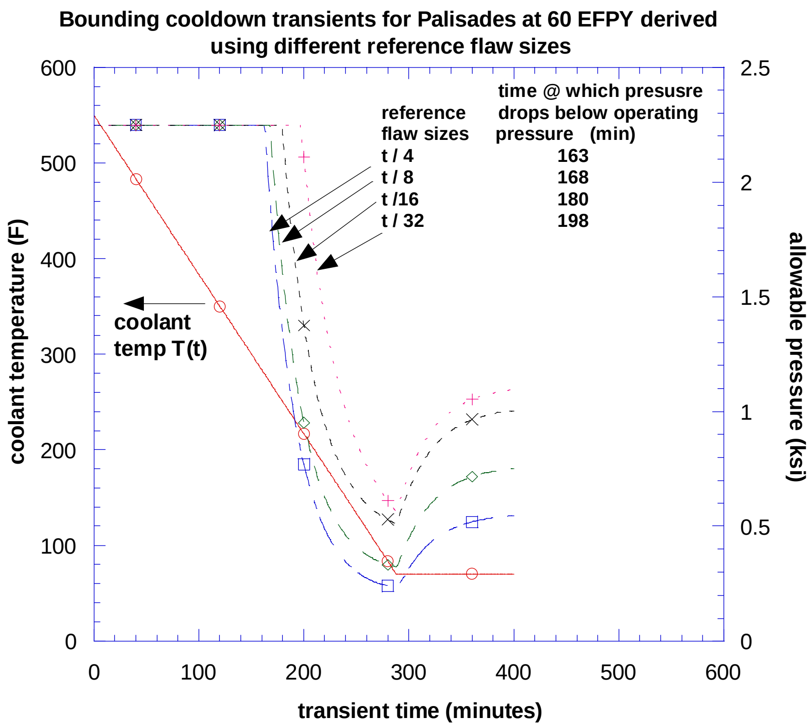
(II) Entirely new rules for deriving limiting P-T curves

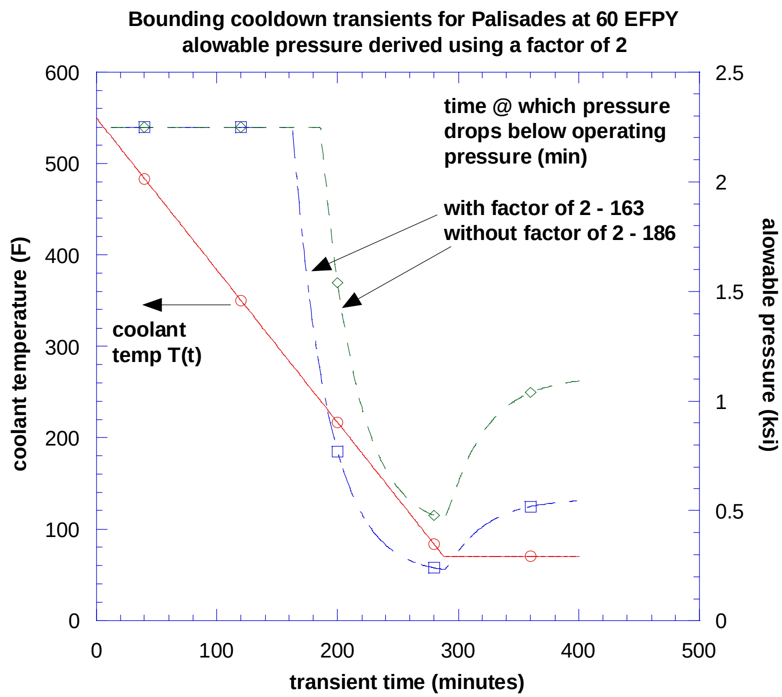
Scoping PFM analysis indicated that relaxations in Section XI – Appendix G deterministic fracture methodology that allow higher pressures

(1) smaller reference flaw size than current t / 4 size
(2) removing the factor of 2 on pressure in derivation

did not increase risk – when WPS included in model
See tables 2 – 7 in Letter Report

All initiations and failures occur at full pressure: Before transients diverge





Oak Ridge National Laboratory
U.S. Department of Energy

UT-BATTELLE

FAVOR^{HT} is being Developed to Calculate Crack Initiation Probabilities for <u>Heat-Up Transients</u>

- During cool-down transients associated with reactor shutdown and PTS, tensile stresses tend to open existing cracks on or near the RPV inner surface

- During heat-up transients associated with reactor startup, tensile stresses tend to open existing cracks on or near the RPV outer surface

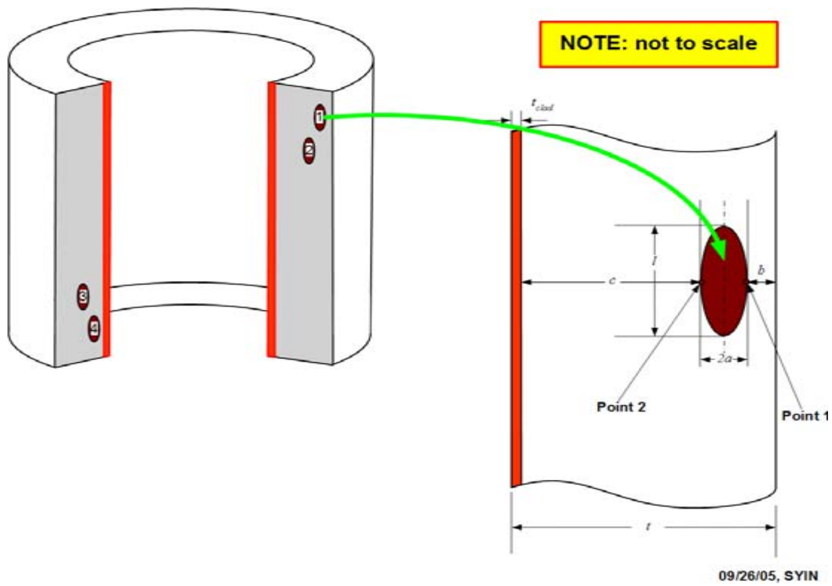
- Previous versions of FAVOR designed for analysis of cool-down transients (fracture mechanics of flaws on or near RPV inner surface)

Initial version of FAVOR^{HT} will calculate conditional probabilities of <u>initiation,</u> for embedded flaw near outer surface<u>,</u> due either to cleavage, but without capability of through-wall crack propagation

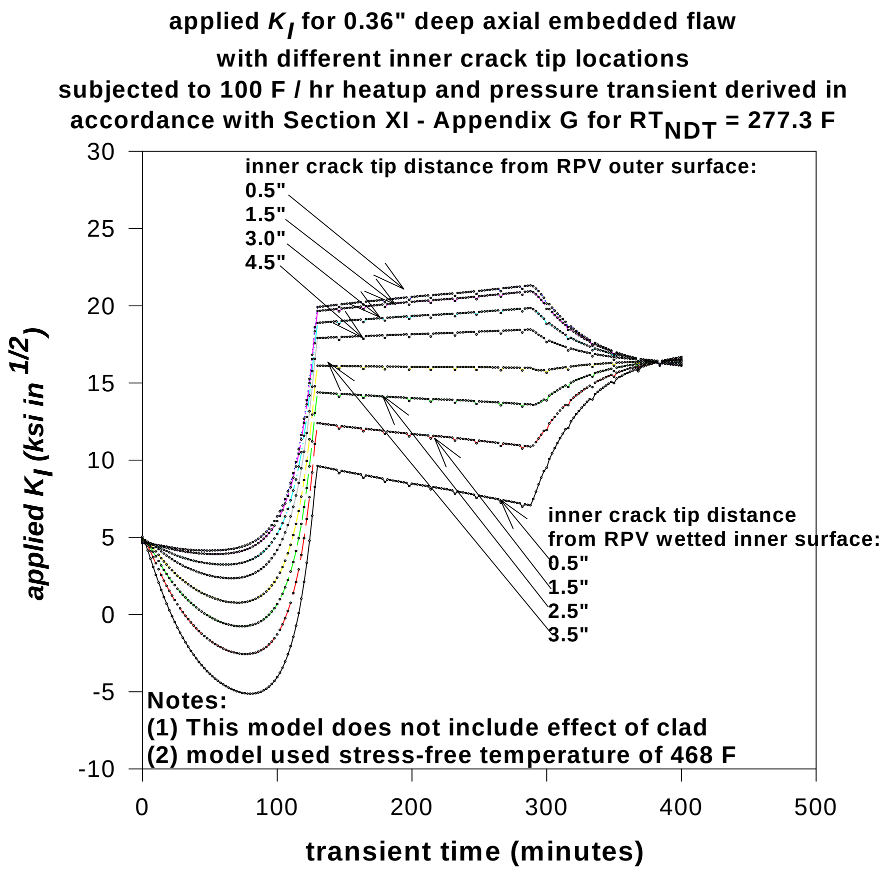
- Therefore, a major requirement for the development of FAVOR^{HT} is to have a validated computational methodology for calculating applied K_I for embedded flaws near the RPV outer surface

The methodology utilized by FAVOR for calculating the applied K_I for embedded flaws near the RPV inner surface has been adapted for calculating the applied K_I for embedded flaws in the outer half of the RPV wall

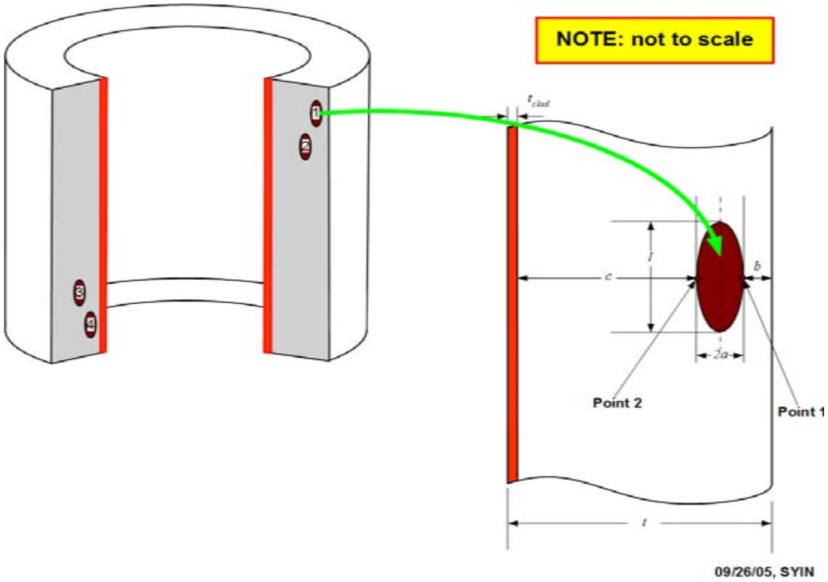
This is accomplished by resolving the nonlinear through-wall stress profile at each time step in a coordinate system that has its origin at the RPV outer surface, as opposed to the RPV inner surface, as is done when calculating the applied K_I solutions for embedded flaws in the inner half of the RPV (with respect to the wetted inner surface)

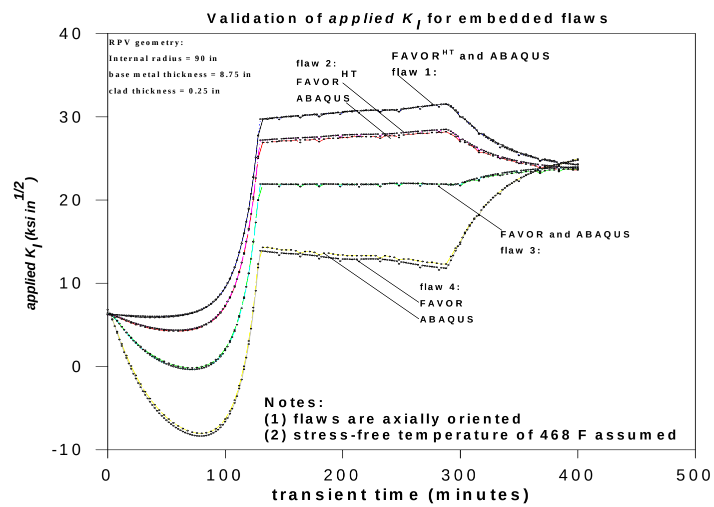


Applied K_I for embedded flaws in first half of RPV wall, relative to the wetted inner surface, are calculated using regular version of FAVOR; whereas flaws in outer half of RPV are calculated using same methodology adapted for embedded flaws in outer quadrant of RPV wall.



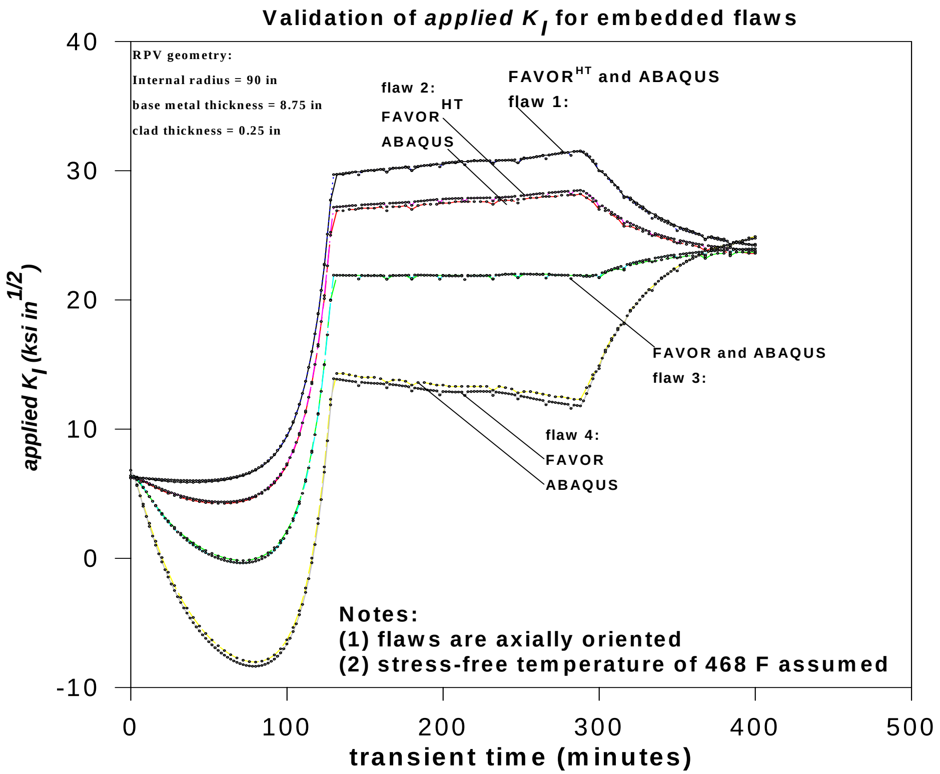
The adaptation of the methodology used by FAVOR (for calculating applied K_I for embedded flaws near the RPV inner surface) has been validated for calculating the applied K_I for embedded flaws close to the RPV outer surface by successfully comparing results with ABAQUS models





flaw model number	c (mm)	b (mm)	depth (2a) (mm)	length (mm)	largest difference in ABAQUS and FAVOR (ksi in$^{1/2}$)
1	189.55	12.7	20	80	0.18
2	126.05	76.2	20	80	0.35
3	69.85	132.4	20	80	0.27
4	6.35	195.9	20	80	0.53

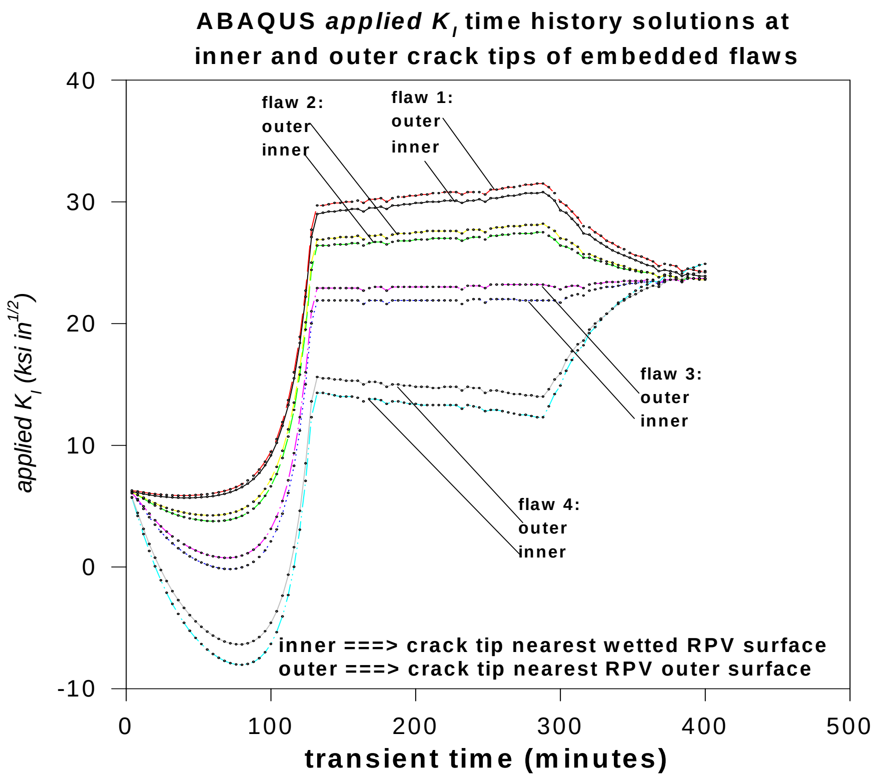
Applied K_I for embedded flaws in first half of RPV wall, relative to the wetted inner surface, are calculated using regular version of FAVOR; whereas flaws in outer half of RPV are calculated using adapted methodology adapted in FAVOR^{HT}



FAVOR^{HT} performs fracture analysis at crack tip nearest the RPV outer surface

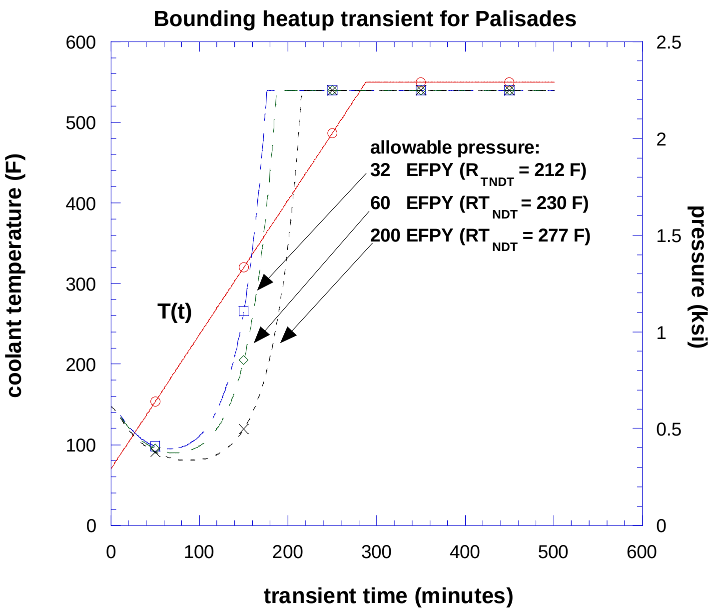
Competing effects: embrittlement decreases from inner surface to outer surface and temperature increases from inner to outer surface (for heat-up)

For cases examined in detail: outer crack tip has higher K_I / K_{Ic} ratio



PFM scoping studies for heatup transients performed with preliminary versions of FAVOR^{HT} indicate very small probability of cleavage fracture

Also, no ductile tearing, as initiating mechanism predicted

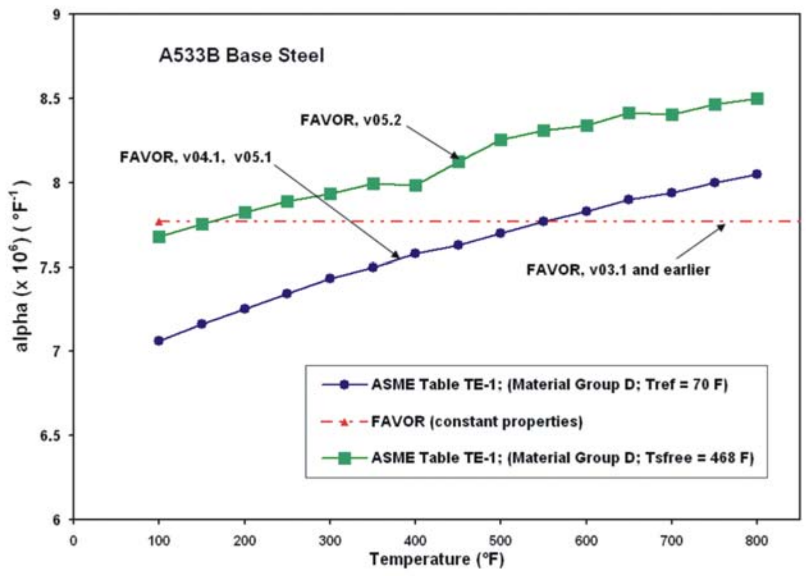


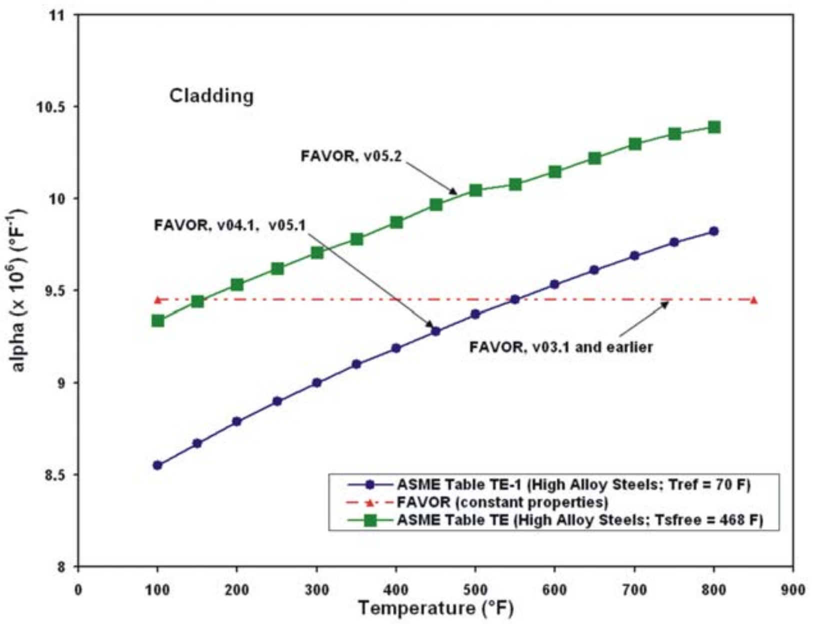
EFPY	*FAVHT CPI due to flaws in outer 3/8 t*	*FAVOR CPI due to flaws in inner 3/8 t*	**Total CPI**
32	0.0e+0	0.0e+0	0.0e+0
60	1.00e-10	0.0e+0	1.00e-10
200	7.94e-10	0.0e+0	7.94e-10

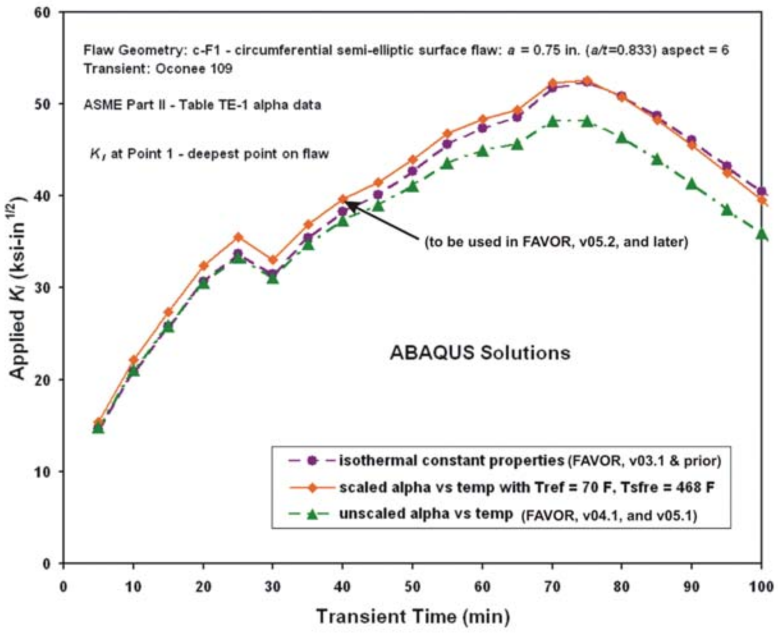
Flaws postulated to reside in inner 3/8 t analyzed with FAVOR code; flaws postulated to reside in outer 3/8 t analyzed with FAVOR^{HT} code.

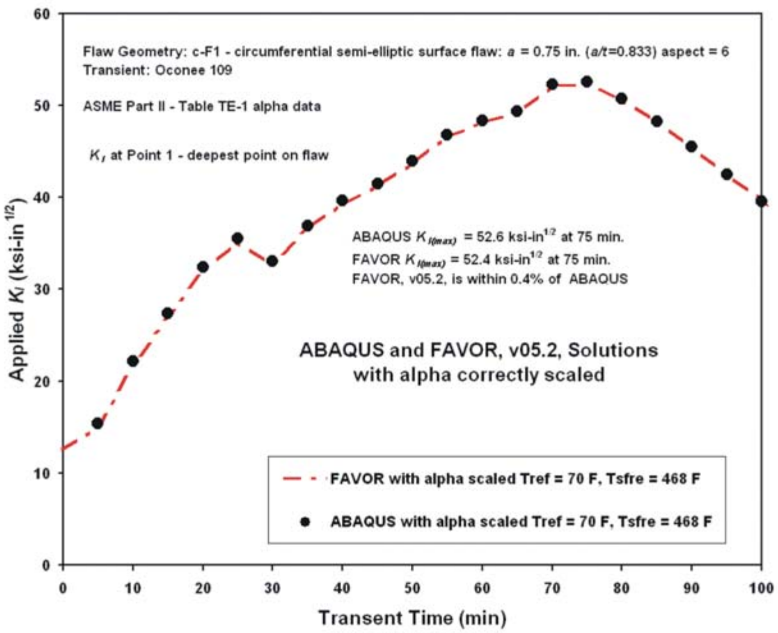
All flaws postulated to have CPI > 0 resided in outer 3/8 t.

BACKUP SLIDES


A533B Base Steel
FAVOR, v05.2
FAVOR, v04.1, v05.1
FAVOR, v03.1 and earlier
ASME Table TE-1; (Material Group D; Tref = 70 F)
FAVOR (constant properties)
ASME Table TE-1; (Material Group D; Tsfree = 468 F)
alpha (x 10^6) (°F^-1)
Temperature (°F)

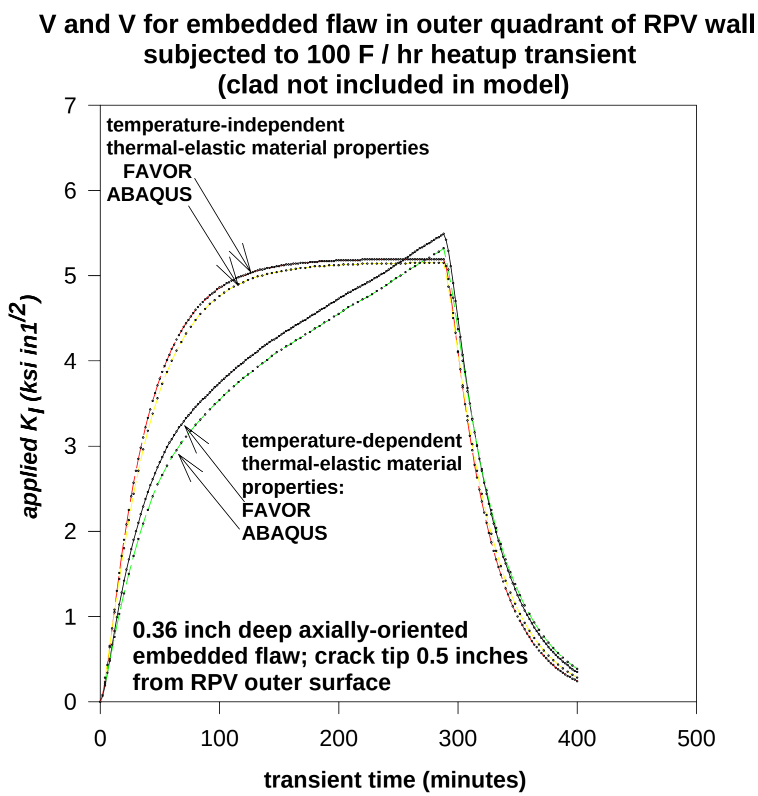

Cladding
FAVOR, v05.2
FAVOR, v04.1, v05.1
FAVOR, v03.1 and earlier
ASME Table TE-1 (High Alloy Steels; Tref = 70 F)
FAVOR (constant properties)
ASME Table TE (High Alloy Steels; Tsfree = 468 F)
alpha (x 10^6) (°F^-1)
Temperature (°F)

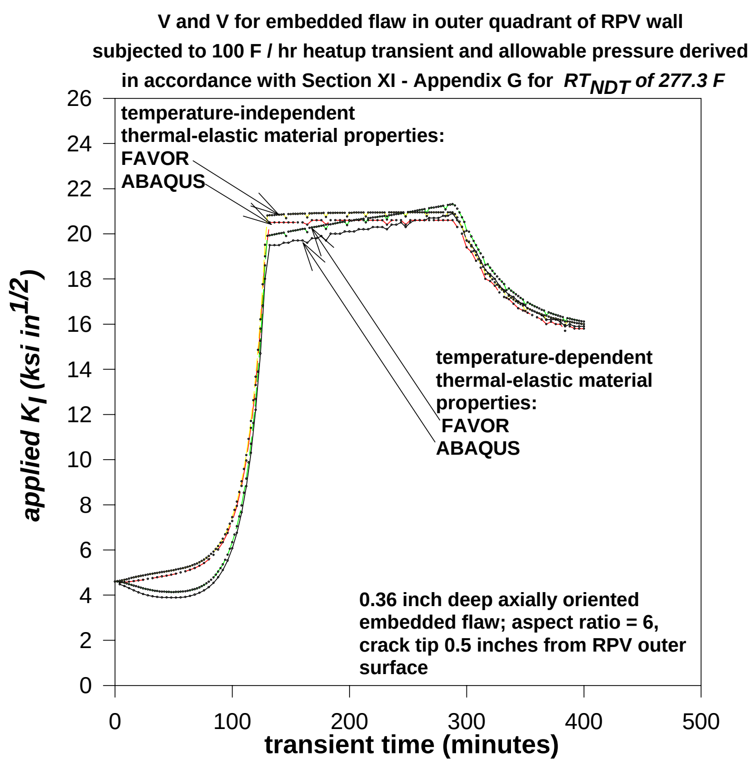

Flaw Geometry: c-F1 - circumferential semi-elliptic surface flaw: a = 0.75 in. (a/t=0.833) aspect = 6
Transient: Oconee 109
ASME Part II - Table TE-1 alpha data
K_I at Point 1 - deepest point on flaw
(to be used in FAVOR, v05.2, and later)
ABAQUS Solutions
isothermal constant properties (FAVOR, v03.1 & prior)
scaled alpha vs temp with Tref = 70 F, Tsfre = 468 F
unscaled alpha vs temp (FAVOR, v04.1, and v05.1)
Applied K_I (ksi-in^1/2)
Transient Time (min)


Flaw Geometry: c-F1 - circumferential semi-elliptic surface flaw: a = 0.75 in. (a/t=0.833) aspect = 6
Transient: Oconee 109
ASME Part II - Table TE-1 alpha data
K_I at Point 1 - deepest point on flaw
ABAQUS K_I(max) = 52.6 ksi-in^1/2 at 75 min.
FAVOR K_I(max) = 52.4 ksi-in^1/2 at 75 min.
FAVOR, v05.2, is within 0.4% of ABAQUS
ABAQUS and FAVOR, v05.2, Solutions
with alpha correctly scaled
FAVOR with alpha scaled Tref = 70 F, Tsfre = 468 F
ABAQUS with alpha scaled Tref = 70 F, Tsfre = 468 F
Applied K_I (ksi-in^1/2)
Transent Time (min)

Work has initiated on the development of FAVOR HT – a code designed to perform fracture analyses of RPVs subjected to heat-up transients associated with the startup of reactors

The adaptation of the methodology used by FAVOR for calculating applied K_I for embedded flaws close to the inner surface of the RPV has been successfully validated for calculating applied K_I for embedded flaws in the outer quadrant of the RPV, close to the RPV outer surface.

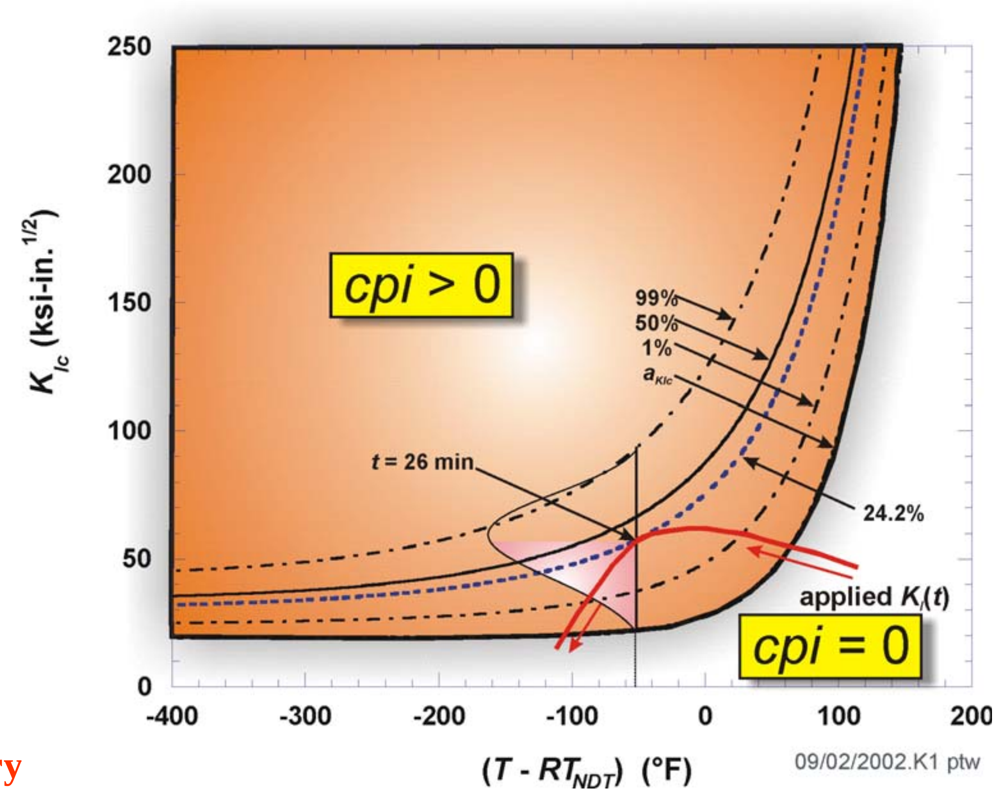




FAVOR Review: cpi is determined from interaction of *applied* K_I and K_{Ic}

Without WPS: for cpi > 0, *applied* K_I must be greater than Weibull "a" parameter which is the lower bound at any transient time

With WPS: for cpi > 0, *max* K_I must be greater than Weibull "a" parameter at transient time before maximum load is reached



Project Plan for Risk-Informing Appendix G

EPRI MRP/NRC Meeting

March 29, 2006

B. Bishop, R. Gamble, R. Lott,
T. Meyer, N. Palm, B. Server

Program Summary and Schedule

- Define Project Goals and Program Plan
  - –Start 3/01/06: Complete 03/17/06
- Define Evaluation Procedure, Methodology & Criteria
  - –Start 03/01/06: Complete 03/17/06
- FAVOR Software Revision (ORNL)
  - –Start 3/01/06: Complete 06/30/06
- FAVOR Software V&V (EPRI/MRP)
  - –Start 7/05/06: Complete 09/01/06

Program Summary and Schedule

- FAVOR Computational Verification
  - Start 3/01/06: Complete 06/30/06
- Collect Analysis Input Data
  - Materials: Start 03/01/06: Complete 03/17/06
  - Flaw Distributions: Start 03/01/06: Complete 09/01/06
  - Components: Start 03/01/06: Complete 09/01/06
  - Events: Start 03/01/06: Complete 09/01/06
- Complete Draft NEPO Funded Report
  - Start 03/01/06: Complete 03/31/06

Program Summary and Schedule

- Process Data to Select Relevant Analysis Input
  - –Start 07/10/06: Complete 09/01/06
- Perform Fracture Mechanics Analyses
  - –Start 09/05/06: Complete 12/30/06
- Evaluate Other Component/System Impact
  - –Start 10/09/06: Complete 02/02/07
- Define Technical Strategy to Risk-Inform Appendix G
  - –Start 09/05/06: Complete 02/02/07

Program Summary and Schedule

- Assess Appendix E Compatibility With Risk-Informed Criteria
  - –Start 09/05/06: Complete 03/02/07
- Complete Reports
  - –EPRI Report: Start 01/02/07: Complete 04/30/07
  - –ASME Basis Document: Start 03/19/07: Complete 06/15/07
- Project Status Meetings
  - –Start 03/01/06: Complete 12/31/07
- ASME Code Implementation
  - –Start 02/02/07: Complete 06/30/08

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Define Project Goals and Program Plan

- Develop a risk-informed methodology for ASME Appendix G for constructing normal heat-up and cool down (ASME Service Level A & B) limits for LWRs
- The risk-informed methodology should:
  - Be applicable to PWR and BWR designs and associated normal operating (ASME Service Level A & B) conditions
  - Define normal operating limits that ensure acceptable margins against core damage are maintained
  - Eliminate unnecessary conservatisms that may limit reactor heat-up and cool down, and operational flexibility
  - Have the same ease-of-use that currently exists in ASME Appendix G

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Define Evaluation Procedure, Methodology & Criteria

- Development of the risk-informed ASME Appendix G methodology will be based on the methodology defined in the Phase I, "Proof of Concept"
- The basic equation for determining the allowable pressure for normal reactor heat-up and cool down is
  - $2\ K_{Im} + K_{It} < K_{Ic}$, where
  - K_{Im} is the stress intensity for membrane tension,
  - K_{It} is the stress intensity for the radiant thermal gradient,
  - K_{Ic} is the fracture toughness, or resistance to extension of an existing flaw,
  - 2 is the margin applied to K_{Im}

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Define Evaluation Procedure, Methodology & Criteria

- The approach is to replace the margin of 2 by a risk-informed margin that is determined from probabilistic fracture mechanics analyses. The risk informed margin will be determined using the relationship
  - CPF x event frequency = failure frequency, where
    - CPF is the conditional probability of vessel failure and is determined from FAVOR
    - The event frequency is the start-up and shut down frequency, and is approximately 0.5 to 1 per operating year
    - The failure frequency $\leq$ a risk limit that is based on the criteria used in the PWR PTS evaluation

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Define Evaluation Procedure, Methodology & Criteria

- The risk-informed margin is determined by finding the value that when used to construct the pressure-temperature operating limits results in an acceptable risk when the reactor operates up to those limits.
- The risk-informed margin will be determined for a range of variables including:
  - Reactor design
  - Beltline materials combinations
  - Adjusted reference temperature, ART_{NDT}
  - Heat-up and cool down rates

Technical Approach to Develop a Risk-Informed, ASME Appendix G

FAVOR Software Revision and V&V

- ORNL will complete and provide FAVOR, V06.1, which will be used to generate the data needed to develop the risk-informed ASME Appendix G
  - Incorporate the new embrittlement correlation
  - Incorporate other changes since version 04.1
  - Software for evaluating startup?
- EPRI/MRP to perform verification and validation (V&V) of the changes made to FAVOR going from V04.1 to V06.1

Technical Approach to Develop a Risk-Informed, ASME Appendix G

FAVOR Computational Verification

- PFM computations will be performed with the FAVOR software to verify that computations previously generated by ORNL can be independently duplicated for a range of variables of interest including:
  - Warm prestress
  - Temperature dependant mechanical and physical properties
  - Simplified, more efficient beltline materials and associated fluence distribution models

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Collecting Analysis Input Data

- Materials and fluence related input data
  - –Reevaluate the adjusted reference temperatures using the new embrittlement correlation
    - – All beltline materials in BWRs and PWRs
    - –Consider impact of up-rating and removal of flux reduction
- Flaw related input data
  - –BWR vessel cladding method (impact on surface breaking flaw assumption)
  - –Verify basis to assume no OD surface breaking flaws, i.e., no cladding, multi-pass cladding
  - –Determine applicability of FAVOR flaw distributions to BWRs

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Collect Analysis Input Data

- Component input data
- Identify components that are subject to ASME Appendix G criteria
  - RV nozzles in PWRs and BWRs
    - Fluence above E17 (embrittlement concern?)
    - Fluence below E17 (initial evaluation only?)
  - Components other than the beltline and nozzle regions

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Collecting Analysis Input Data

- Operating event data
  - –Develop a complete set of possible and actual heat-up & cool down events to be evaluated for BWRs and PWRs
  - –Finalize utility desired flexibility and operational constraints
  - –Determine frequency of heat-up & cool down events
  - –Consider low frequency events during normal heat-up & cool down & refueling

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Process Data to Select Relevant Analysis Input

- Vessels and beltline materials that will be evaluated
- Beltline fluence maps
- Operating events
- Components other than the vessel beltline
- Relevant flaw distributions

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Perform Fracture Mechanics Analyses (Beltline Region)

- Deterministic analyses
  - –Sensitivity and scoping analyses to determine the relative importance of variables
  - –Computations to define pressure and temperature time history input for the FAVOR software.
- Probabilistic fracture mechanics (PFM) analyses
  - –Use FAVOR software to compute failure frequency and evaluate the conditions relative to the risk limit
  - –Results will be used to develop the risk-informed ASME Appendix G

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Evaluate Other Component/System Impact

- Perform sensitivity and scoping analyses to determine the relative importance of component and systems variables not included explicitly in the PFM analysis

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Define Technical Strategy to Risk-Inform Appendix G

- Use PFM results to define a risk-informed procedure for constructing reactor normal operating heat-up and cool down pressure temperature limits
- Procedure should have the same ease-of-use that currently exists in ASME Appendix G
- Procedure should use an irradiation index similar to that used for the PWR PTS evaluation
- Procedure should be applicable to PWR and BWR designs and associated normal operating (ASME Service Level A & B) conditions

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Assess Appendix E Compatibility With Risk-Informed Criteria

- Evaluate the acceptance criteria in Appendix E to determine if they are compatible with risk-informed criteria in Appendix G and the PWR PTS evaluation
- Where applicable, modify Appendix E to obtain compatibility between Appendix E and Appendix G and the PWR PTS criteria

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Reports

- Provide a NEPO report that describes the strategy and program plan for risk-informing ASME Appendix G
- Provide an EPRI report that documents in detail the work performed for risk-informing Appendix G, including any changes recommended for Appendix E
- Provide a report that can be used as an ASME technical basis document for risk-informing ASME Appendix G, including any changes recommended for Appendix E

Technical Approach to Develop a Risk-Informed, ASME Appendix G

Project Status Meetings

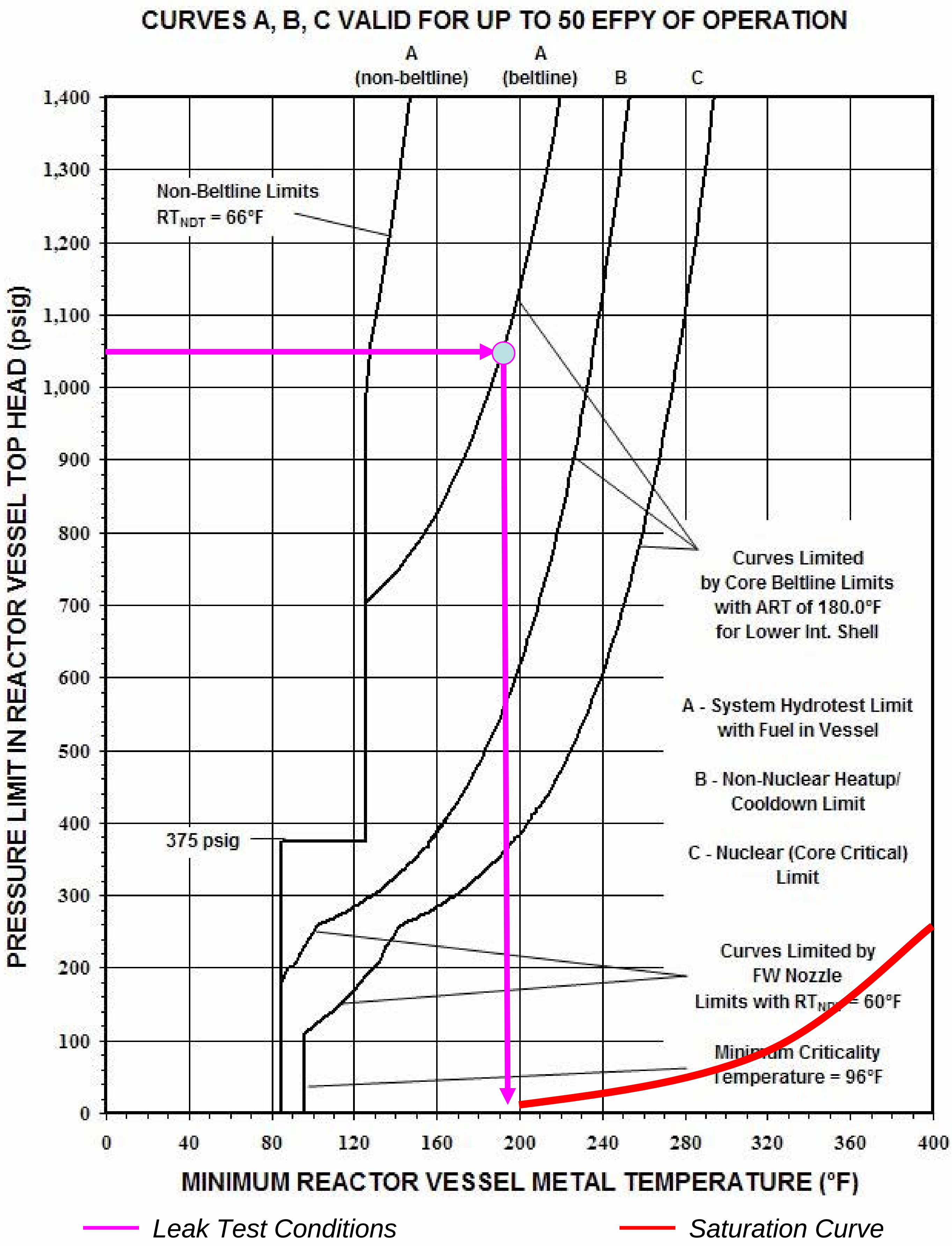
- Project team members will participate in periodic progress meetings to provide status updates and information flow between the project team, and industry and regulatory groups.

Technical Approach to Develop a Risk-Informed, ASME Appendix G

ASME Code Implementation

- Participate in ASME Code activities to implement the risk-informed methodology into ASME Appendix G
- Participate in ASME Code activities to implement changes in Appendix E that may be necessary to provide compatibility with risk-informed criteria in Appendix G and the PWR PTS evaluation

BWR P-T Curves



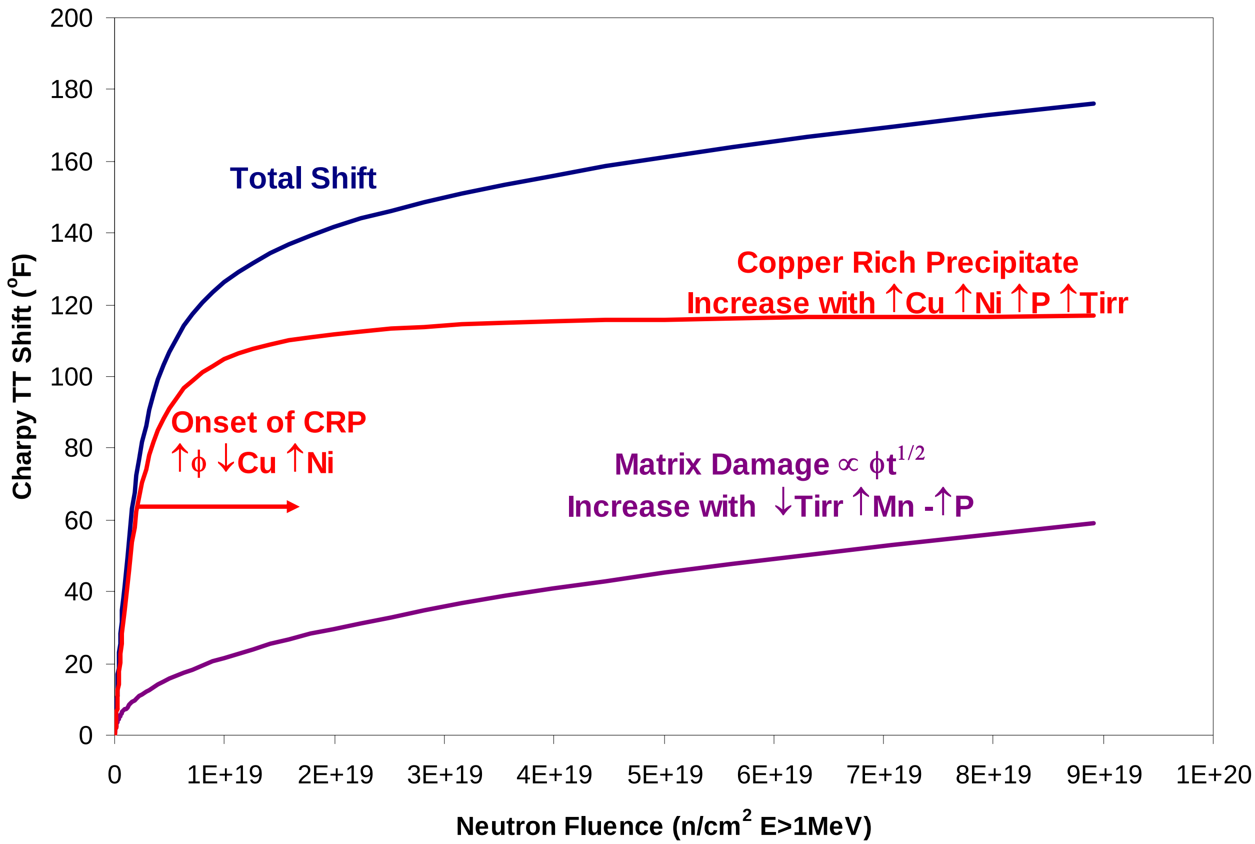
New NRC Embrittlement Trend Curve and Plant Evaluations

Randy Lott & Bill Server
on behalf of
MRP RPV Integrity Focus Group
NRC Headquarters, Rockville MD
March 29,2006

New NRC Embrittlement Trend Curve

- First presented to industry at ASTM E10 Meeting (Feb. 6, 2006).
- Objective of this presentation to explore the practical implications of using the proposed NRC embrittlement trend curve for Appendix G applications.

Basic Characteristics of the New NRC Embrittlement Trend Curve



Three Basic Parameters of Fit

Matrix Damage Coefficient

MD = $A\phi t^{1/2}$

Onset of CRP Term

$\tanh(\phi t_e)$

CRP Saturation Level

High ϕt

Note: In current version there is no long time bias term.

Details have changed but format is well established.

Appendix G Concerns
- and Evaluations

- Predicted RT_{NDT} values critical to both probabilistic and deterministic analysis.
  - Extensive comparisons between current Reg. Guide and proposed NRC embrittlement trend curve.
  - Sensitivity study to evaluate significance of prediction variables.
- Margin term applied to deterministic analysis based on standard deviation of prediction errors.
  - Compare standard deviations of data correlations.
- Shift attenuation through vessel wall determines RT_{NDT} for both ¼ T and ¾ T locations.
  - Compare predictions to preliminary results of attenuation study.

Analysis of New NRC Embrittlement Trend Curve

Two types of Analysis

- Evaluate the quality of the correlation
  - Compare predictions to surveillance data
  - Subject of previous presentations
  - Reduced to a philosophical discussion
- Evaluate the impact of the proposed curve
  - Examine predictions for existing RPVs
  - Identify limiting materials
  - Understand sensitivity to input variables

Impact of New NRC Embrittlement Trend Curve

<u>Plant Data</u>

- Base Source: NRC-RVID
  - Comprehensive list of belt-line materials
    - Material Type, Heat, Weld Flux
    - Cu, Ni, P
    - 40 year-EOL Fluence (Flux implied)
- Supplementary Source: E900
  - Surveillance Capsule Conditions
    - Temperature (Tc)
- Supplementary Source: EPRI RPVData
  - Underlying Database
    - Mn

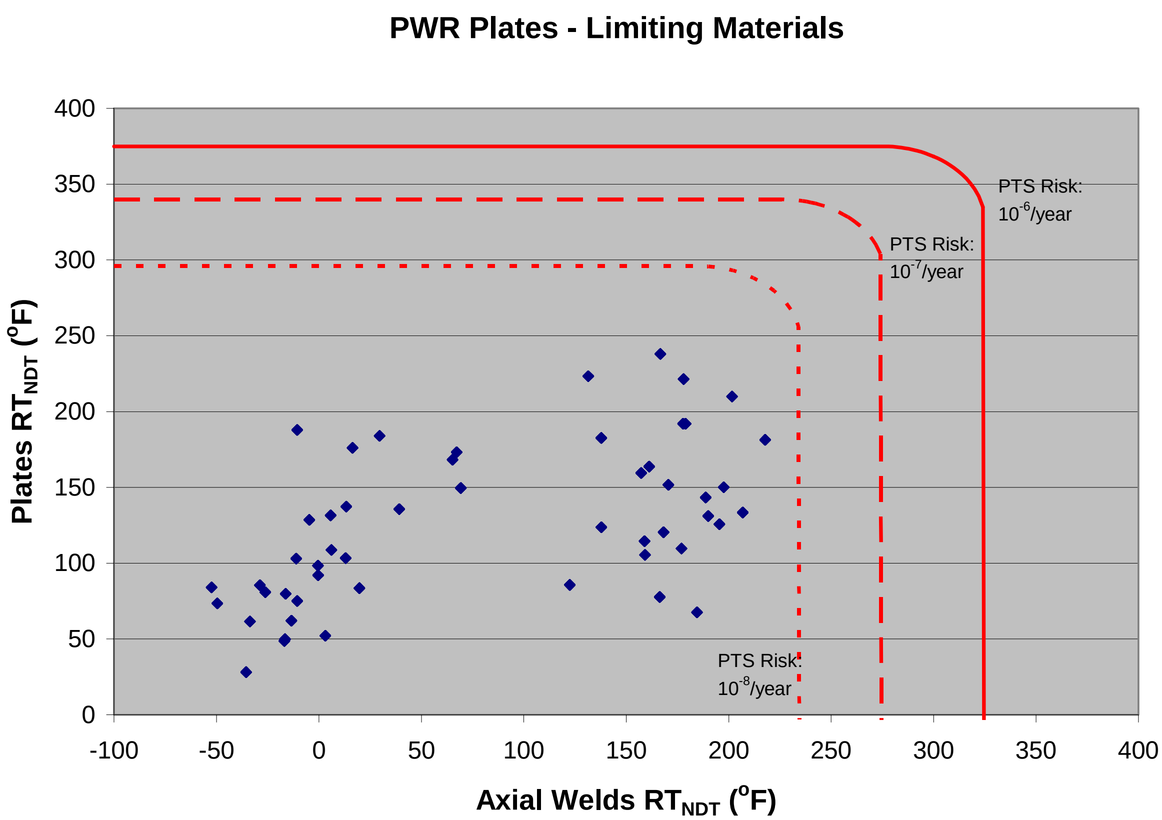
Impact of New NRC Embrittlement Trend Curve

Strategy for Missing Data

- Relevant Mn data identified
  - Forgings: 54/69
  - Plates: 417/496
  - Welds: 156/196
- Relevant P data identified for even higher fraction of plant
- Missing Mn and P data estimated by averaging similar materials
- Default BWR temperature was 529^{o}F

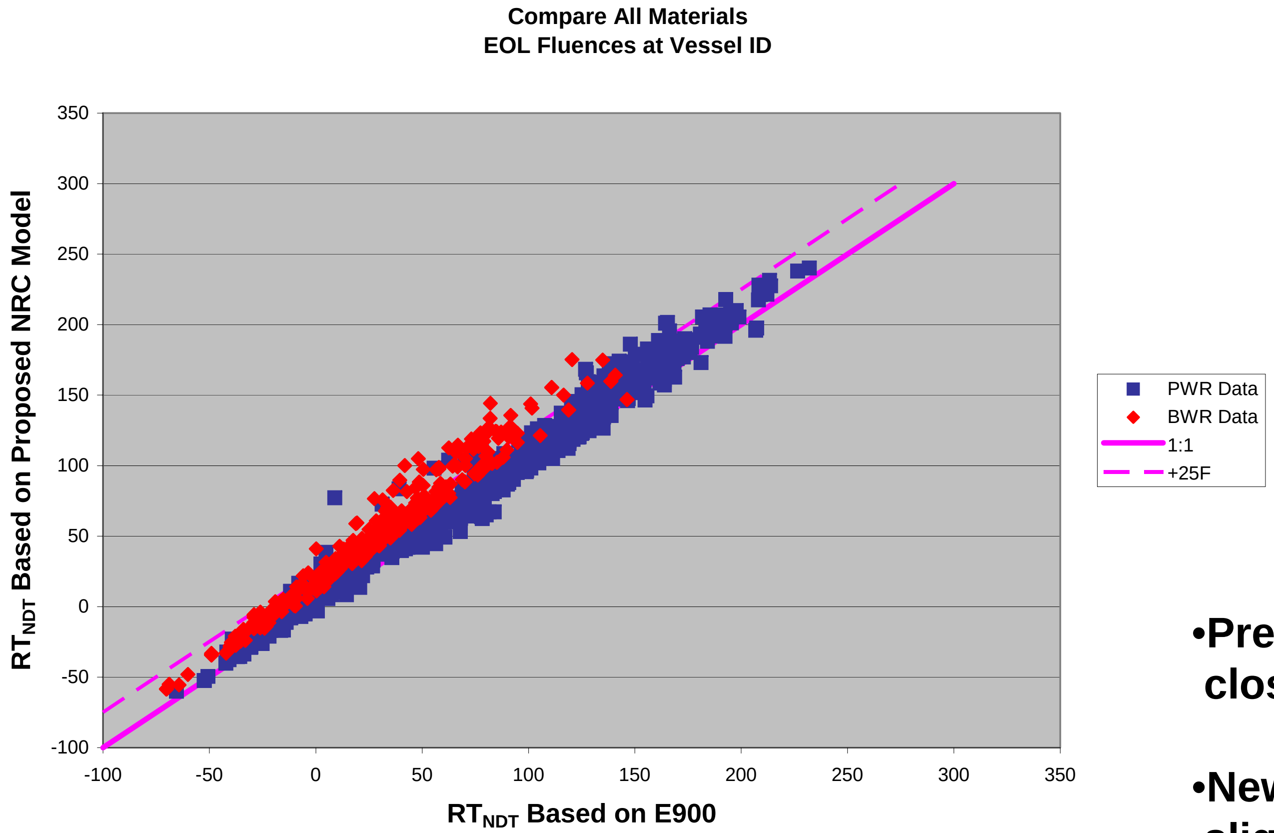
PWR Fleet Wide Evaluation

EOL Fluences - ID Surface
Comparison to Potential PTS Limits

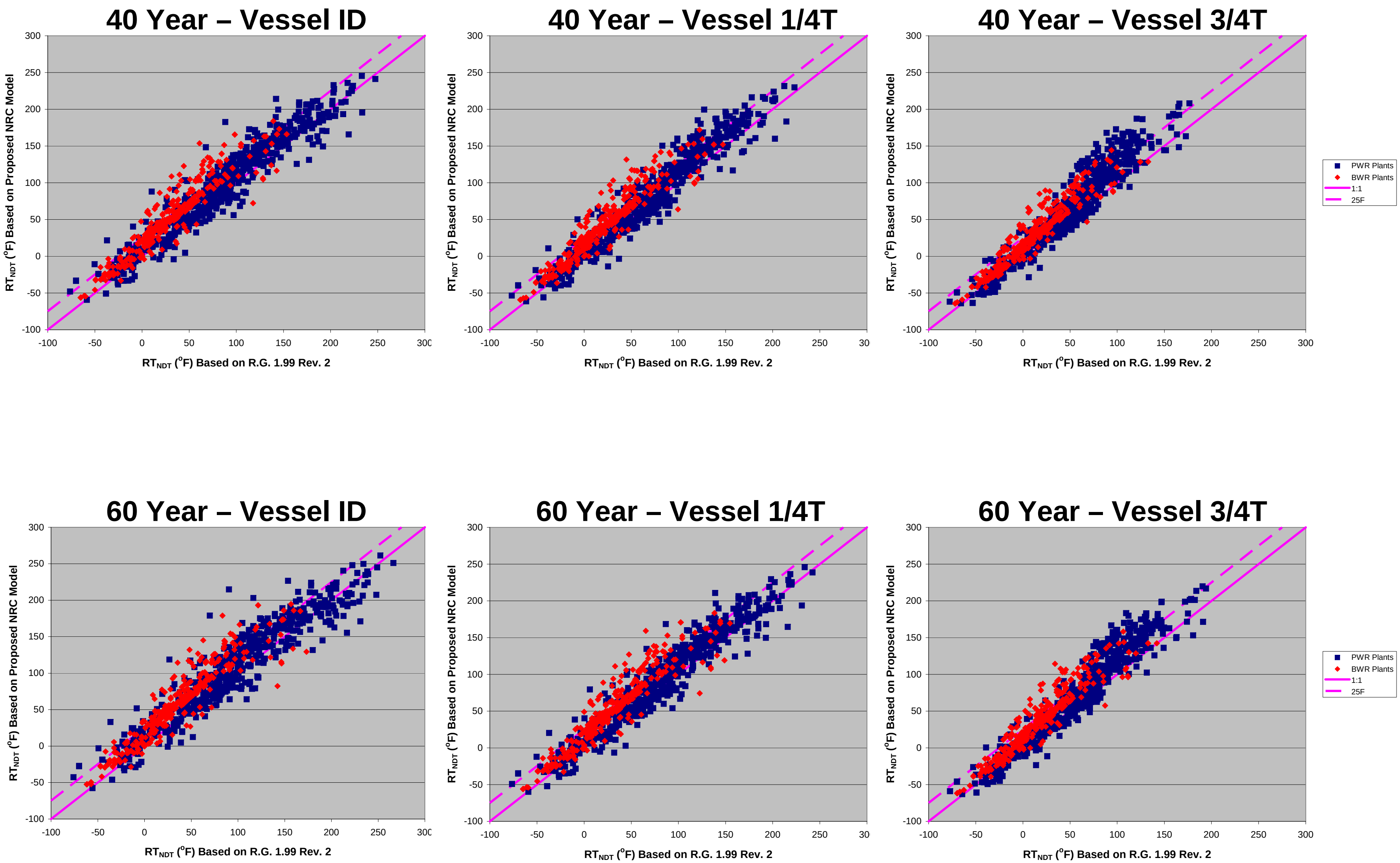


- Maximum RT_{NDT} values (no margin included).
- Some plants have circ welds with RT_{NDT} > axial weld RT_{NDT}.
- Need equivalent “limiting material” definition for Appendix G analysis.

Comparison to E900 Predictions

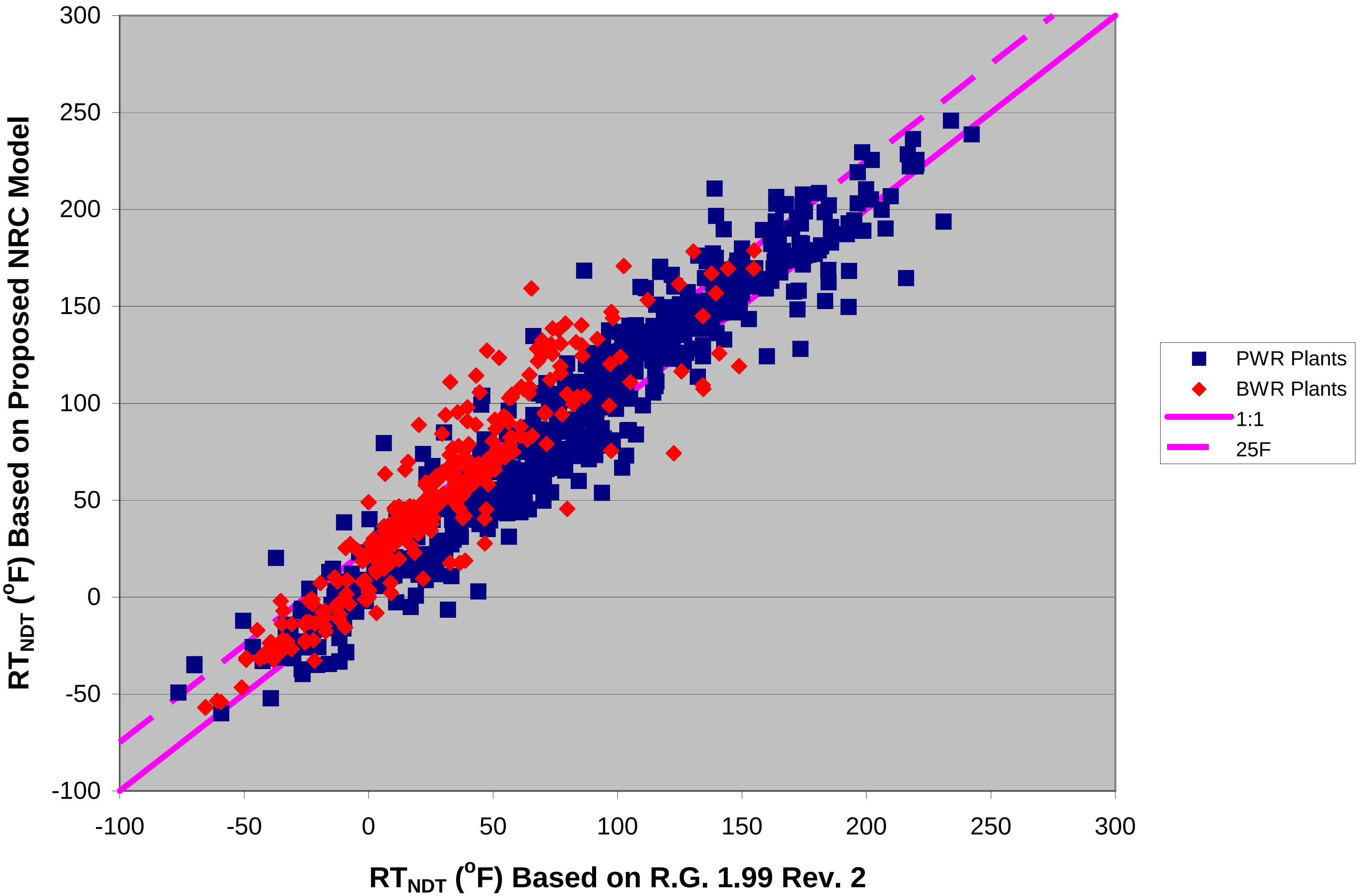


- **Predictions are closely correlated.**
- **New Curve gives slightly higher RT_{NDT} values.**

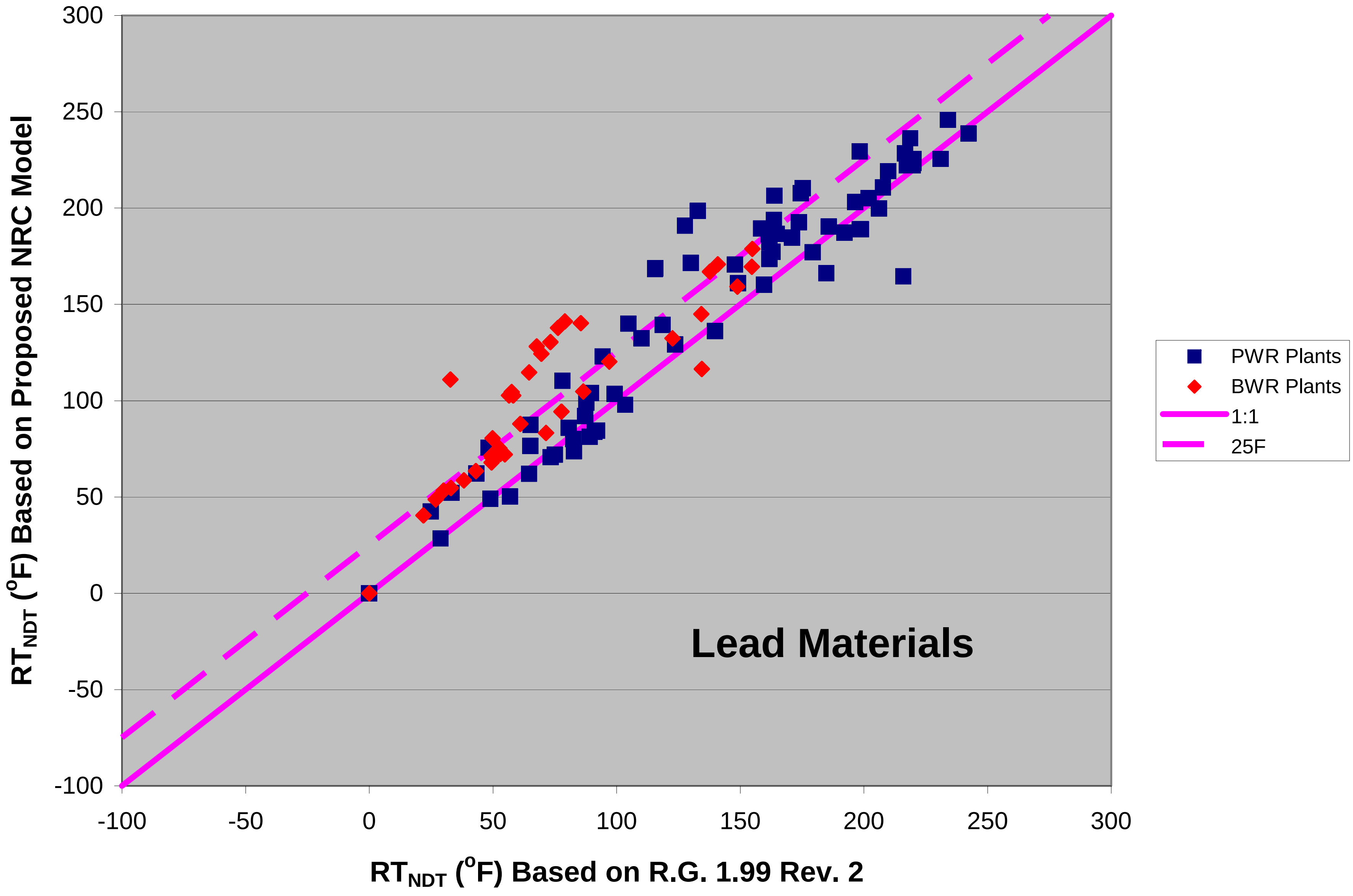


Compare New Curve to R.G. 1.99 Rev. 2

60 Year – Vessel 1/4T



60 Year – Vessel 1/4T

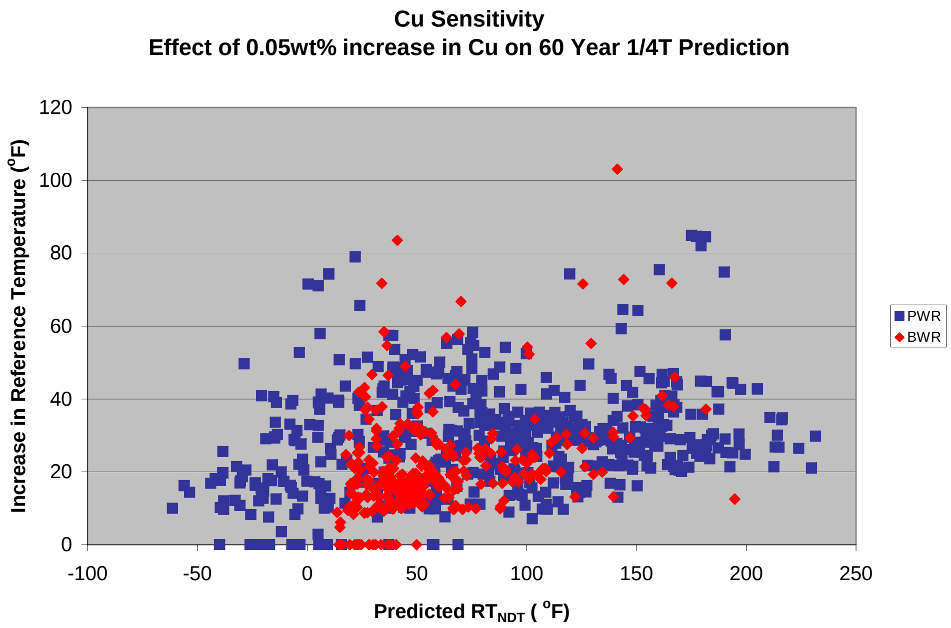


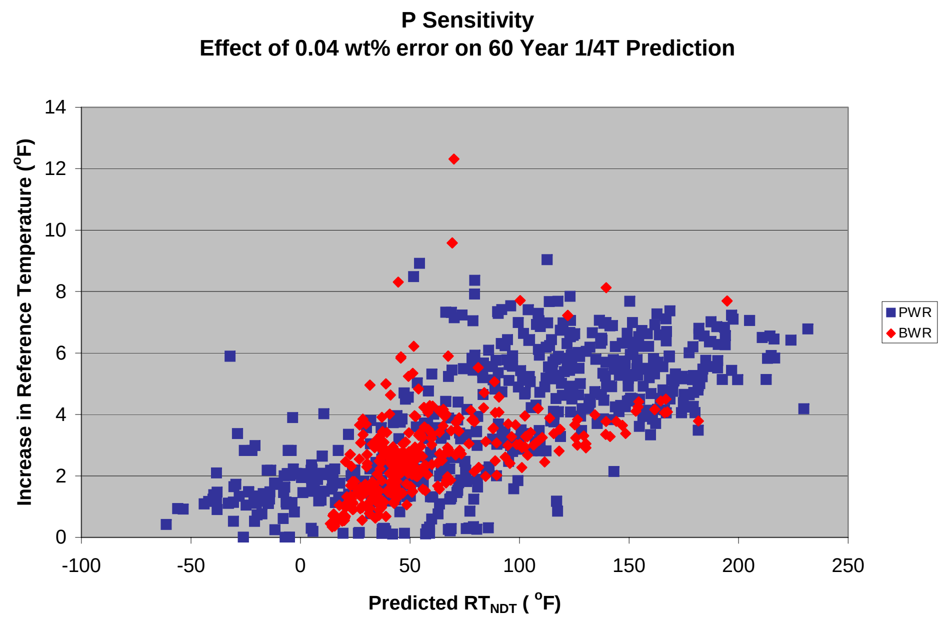
Impact of Moving from Reg. Guide 1.99, Rev. 2 to New Curve

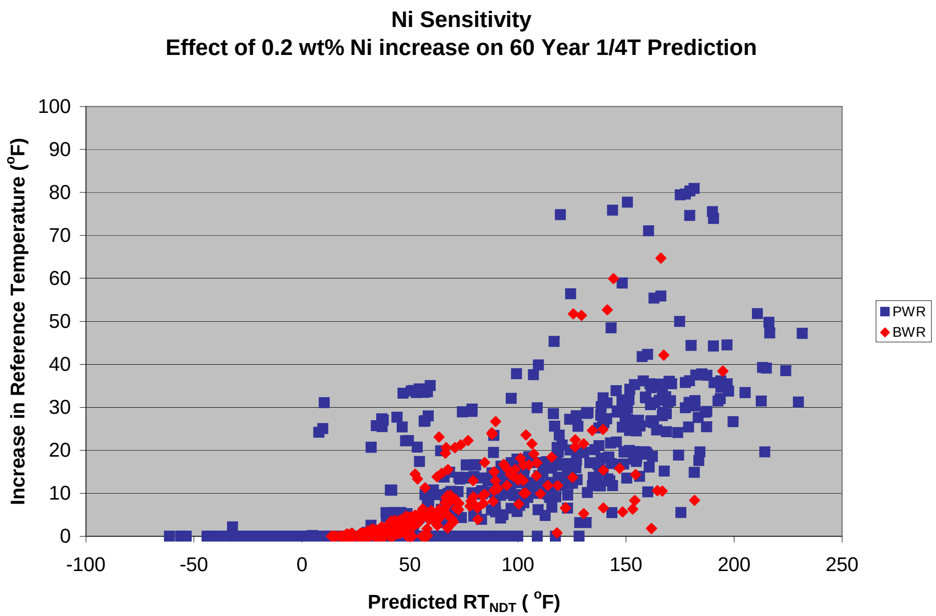
- Low Fluences (BWR): Predicted RT_{NDT} increases by ~25^{o}F
- High Fluences (>5x10^{19}): MD Term increases
- Most plants can expect to see a change in limiting RT_{NDT} between -10^{o}F and +40^{o}F

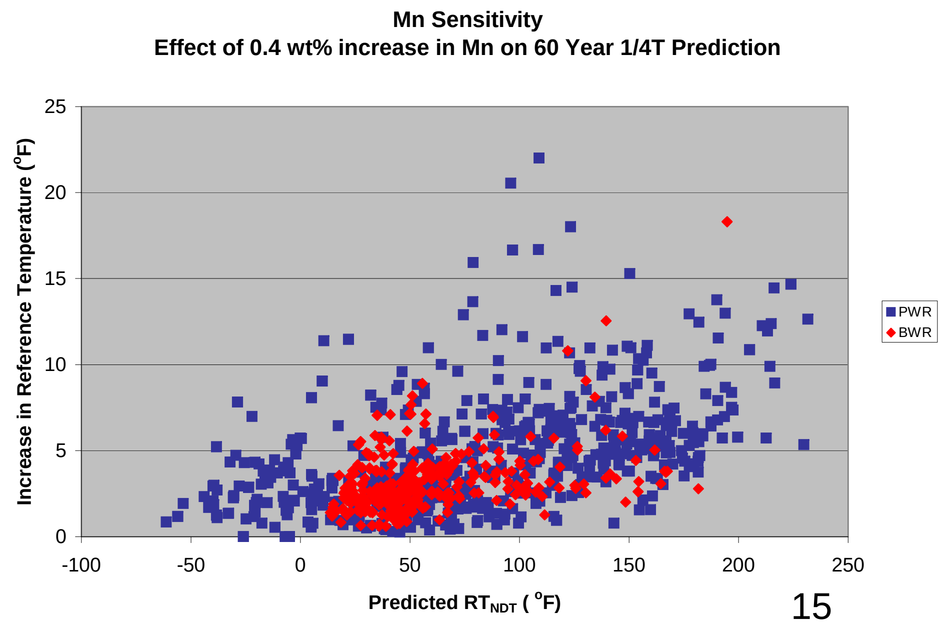
Sensitivity Studies

- Perturb input values one variable at a time to evaluate sensitivity to the uncertainty.
- Sensitivity studies conducted for fluence equivalent to 60 years at 1/4T.

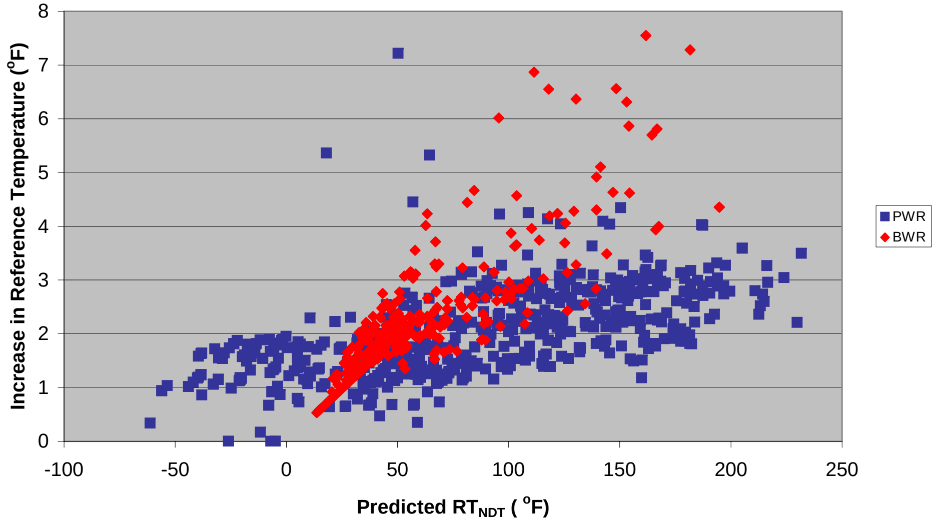

Cu Sensitivity
Effect of 0.05wt% increase in Cu on 60 Year 1/4T Prediction
Increase in Reference Temperature (°F)
Predicted RT_NDT (°F)
PWR
BWR


P Sensitivity
Effect of 0.04 wt% error on 60 Year 1/4T Prediction
Increase in Reference Temperature (°F)
Predicted RT_NDT (°F)
PWR
BWR

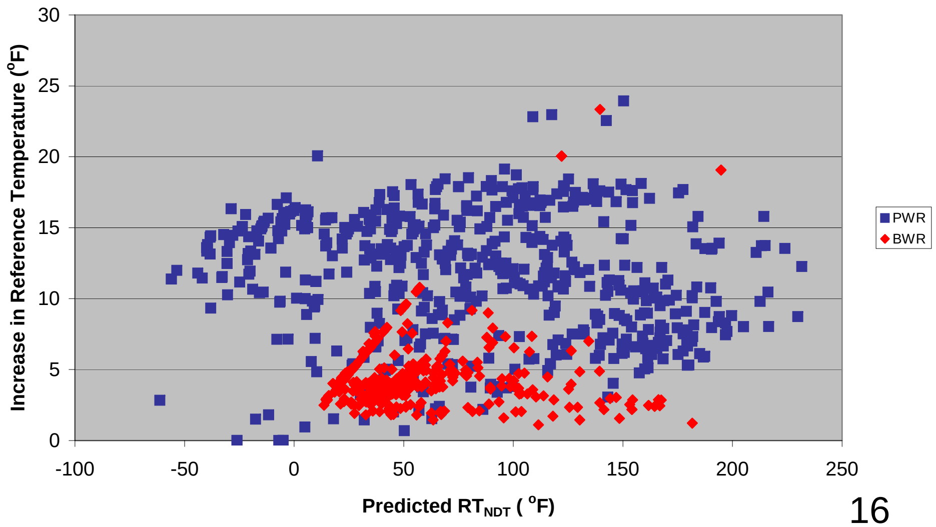

Ni Sensitivity
Effect of 0.2 wt% Ni increase on 60 Year 1/4T Prediction
Increase in Reference Temperature (°F)
Predicted RT_NDT (°F)
PWR
BWR


Mn Sensitivity
Effect of 0.4 wt% increase in Mn on 60 Year 1/4T Prediction
Increase in Reference Temperature (°F)
Predicted RT_NDT (°F)
PWR
BWR

Fluence Sensitivity
Effect of 20% Increase in Fluence on 60 Year 1/4T Prediction



Temperature Sensitivity
Effect of 10°F *decrease* in Tc on 60 Year 1/4T Prediction



Other Application Issues

- Margins
  - If we are moving to a risk based approach we should not need them!
  - At a minimum need to understand purpose of margins.
- Attenuation
  - What attenuation factors are required?
  - What are the rules applied in the Favor Code?

Statistics on Correlations

NRC Presentation to ASTM

Standard deviation of residuals (Sd) about the embrittlement shift model in various subsets, all PWR and BWR calibration and validation data except SRM

Product Form	Sd for Cu ≤ 0.072 wt%	Sd for Cu > 0.072 wt%
Forging	17.5	19.8
Plate	15.0	20.9
Weld	18.6	26.3

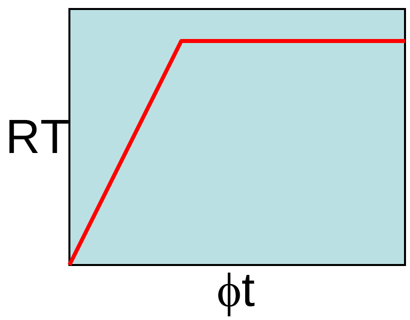
These Sd values by product form should be considered when setting margins.

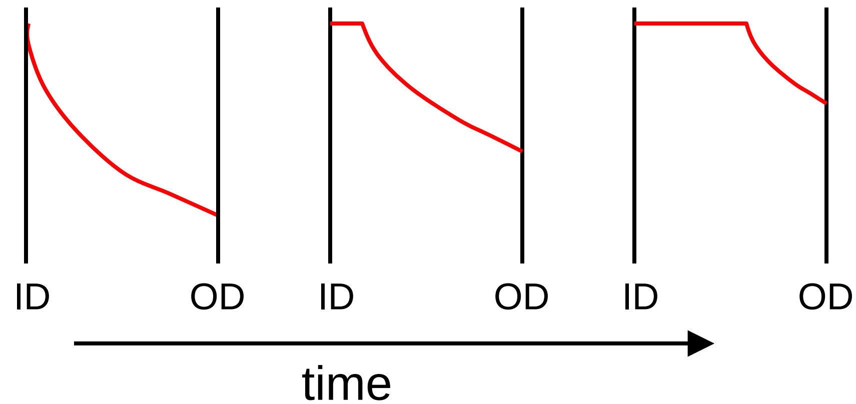
Current NRC Reg. Guide 1.99 Rev. 2

Product Form	Sd
Forging	17
Plate	17
Weld	28

Attenuation Effects

- Embrittlement Curve Saturation





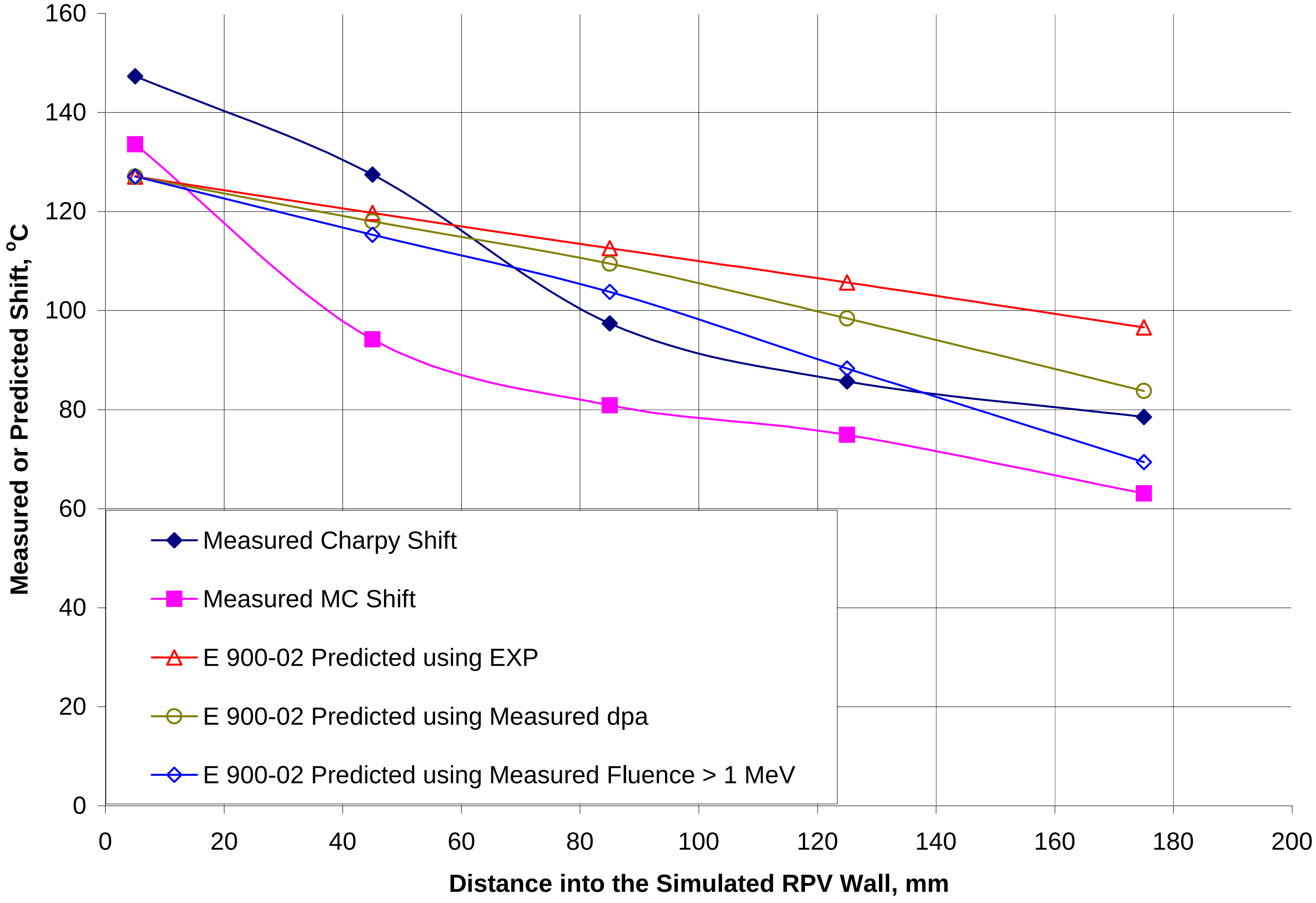
- dpa/Fluence Ratio
  - More displaced atoms per unit fluence in outer portion of RPV
- Flux Effect in New NRC Embrittlement Trend Curve
  - More Embrittlement at Lower Damage Rates

IAEA/MRP Attenuation Study

- Master Curve T_o and Charpy T_{41J} results through the thickness of a 180 mm (7.1-in.) RPV wall for at least two materials representing high and low copper contents
  - Provide a direct check on the attenuation formula now used in US Regulatory Guide 1.99, Rev. 2, ASTM E900-02, and the PTS re-evaluation effort
  - Linde 80 weld metal chemistry is 0.30 wt% Cu, 0.58 wt% Ni, and 0.017 wt% P
- Should see Curve Saturation and dpa/Fluence Effects. Test reactor irradiations exceed flux effect regime.

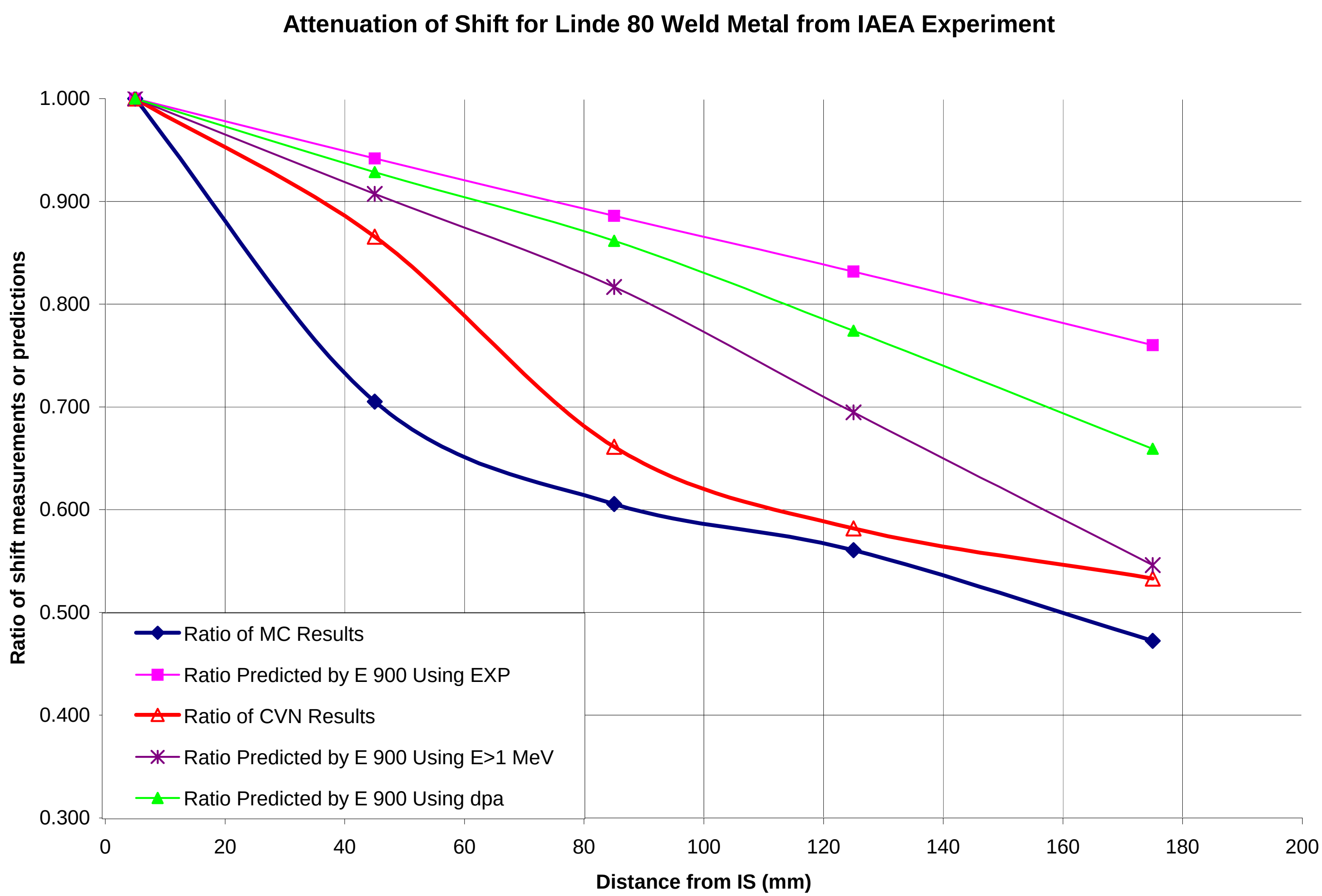
Attenuation Study

Measured and Predicted Shifts in T_0 and T_{41J}



Attenuation Study

Shifts Normalized to Inner Surface.



Preliminary Attenuation Study Conclusions

- Estimated dpa, measured dpa, or E > 1 MeV do not accurately predict through-wall attenuation of the Linde 80 weld metal.
- Attenuation observed near ID surface larger than expected.

- Testing of a low Cu plate will be completed in a few months.
- Testing of the JRQ plate and the WWER-1000 forging would provide additional confirmation and the possibility to investigate different layers between the ID and ¼-T
  - Interested parties are invited to participate in this study

Observations Relevant to Appendix G Applications

- Data compiled for a comprehensive review of the effect of the new NRC embrittlement trend curve on operating plants.
  - Higher RT_{NDT} for most BWRs
  - Flux effects at high and low fluence can slightly increase predicted RT_{NDT} values for PWRs as well.
- Conclusions are relatively insensitive to uncertainties in Mn, P and fluence values.
- Standard deviations of fits for new NRC embrittlement trend curve are comparable to margins in R.G 1.99 R2.
- Additional experimental work is required to determine the role of dpa and neutron flux on damage attenuation.

MRP-ITG Meeting with NRC

Limits and Frequencies for Vessel Heat Up and Cool Down in Risk Informing Appendix G

Bruce Bishop

Ted Meyer

Nathan Palm

Limits and Frequencies for Vessel HU/CD

- Westinghouse Feasibility Study
  - Description of Study
  - Results of Study
- Limiting Cool-down Transients
  - Studies to Determine
  - Transients Considered
- Remaining Evaluations for Next Phases
- Frequency of Cool-Down Events

Westinghouse Study on Feasibility of Accelerated Heat-up and Cool-down

- 45 Years of operational data from 9 Westinghouse PWR plants was analyzed
- Study revealed departures from design basis cool-down assumptions
  - Pressure hold at beginning of cool-down
  - Step changes in temperature rather than constant rate
  - Temperature hold for alignment to Residual Heat Removal (RHR) System
- Feasibility study investigated whether heat-up and cool-down limits could be relaxed both structurally and operationally (e.g. chemistry control)

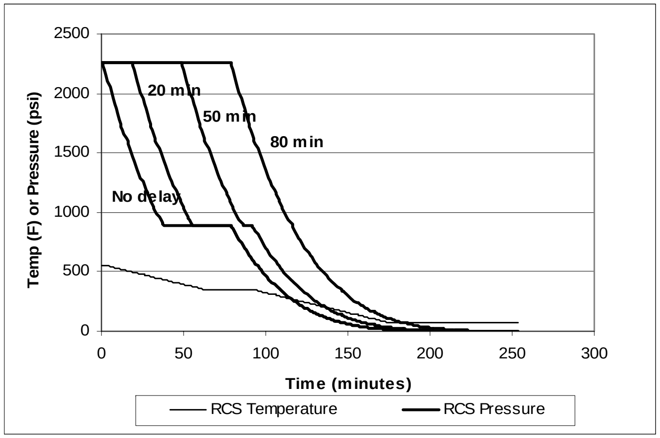
Results from Feasibility Study on Accelerated Heat-up/Cool-down

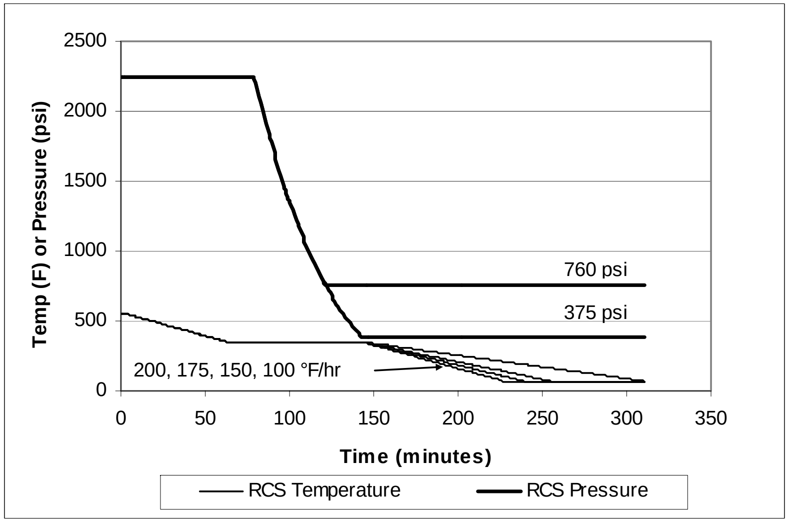
- Structural limits were evaluated
  - Reactor vessel beltline under cooldown conditions determined to be limiting
  - Code developed to perform deterministic Appendix G fracture mechanics evaluations
  - Multiple theoretical transients analyzed at various irradiation levels
- NSSS control systems capabilities were investigated – Up to 200°F/hr cooldown achieved using Westinghouse simulator
- Effects on plant chemistry were investigated
- Overall conclusion was that accelerated cooldown was feasible

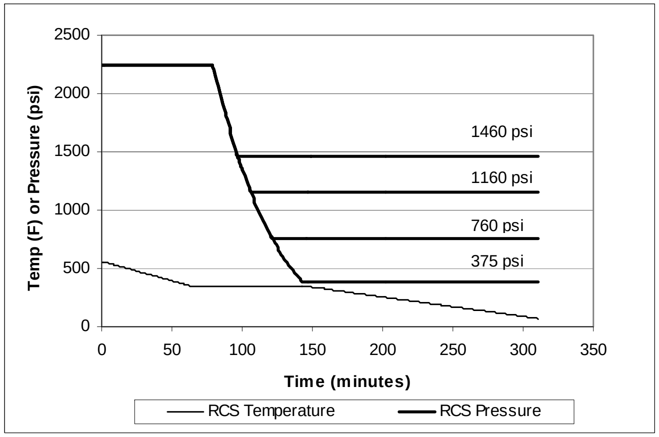
Studies to Determine Limiting Transient

- Goal is to find a transient for which K_I approaches K_{IC} so that when input into FAVOR the resulting frequency of initiation is close to risk goal
- Several considerations in determining limiting transients
  - 200°F/hr cooldown rate for accelerated HU/CD Study
  - Pressure hold at beginning of cooldown
  - Temperature hold at 350°F for RHR alignment
  - 375 psi minimum pressure for RCP operation
- Beaver Valley Unit 1 at 60 EFPY chosen for studies; Maximum RT_{NDT} = 332°F

Limiting Transients Considered







Remaining Evaluations for Next Phase

- *Significance of much lower frequency transients, such as overpressure events*
- Feasibility for less embrittled reactor vessels but with higher RT_{NDT} margin terms
- Applicability to other PWR and BWR NSSS systems and RPV designs
- *Limitations from other plant components or systems (e.g. LTOP)*
- *Flexibility tradeoff between needs of WOG plant operators and requirements of ASME Code*

Frequencies for Cool-down Events

- For conceptual evaluation, one event every 1 or 1.5 years was used for mean frequency
  - Westinghouse used distribution from 1.25 to 1.75 years
  - ATI/Sartrex suggested range of 1 to 2 years
- Phase 2 work to look at two types of events
  - More frequent normal events, like partial cool downs
  - Much less frequent off-normal events, like over pressure
- Time histories of temperature and pressure and frequency of events will be estimated using
  - Previous plant operating experience
  - Fault-tree (PRA) evaluation of what could go wrong in the future, like LTOP system operating failures