CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Reactor Safeguards Materials & Metallurgy and Plant Operations Joint Subcommittee Meeting

Docket Number: N/A

Location: Rockville, Maryland

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

/s/ Debra Wilensky

Debra Wilensky
Official Reporter
Neal R. Gross & Co., Inc.

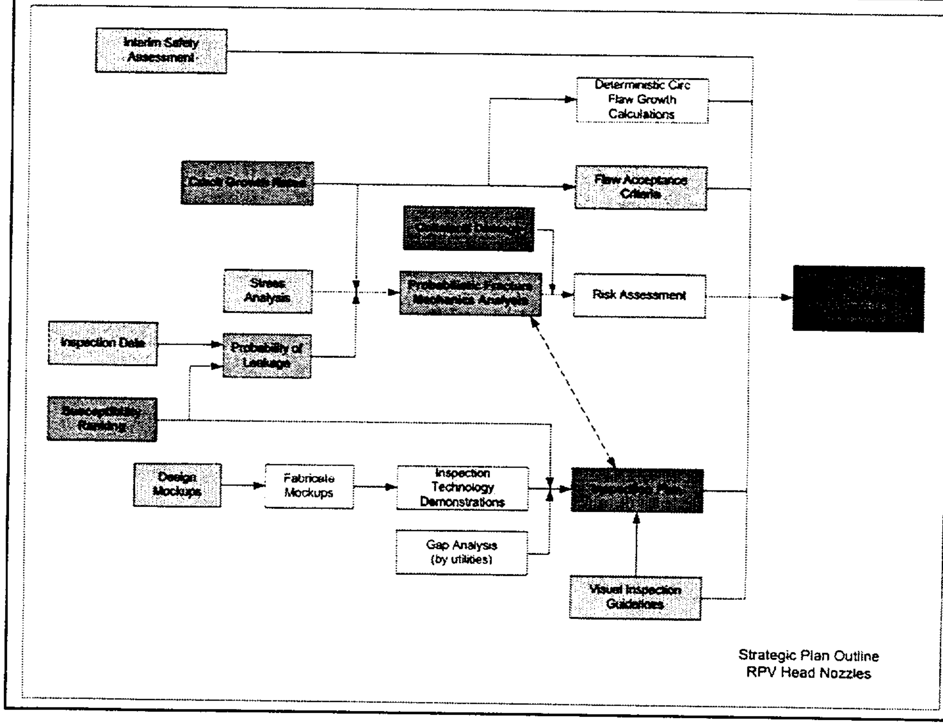
MRP Update to ACRS Materials Subcommittee June 5, 2002

Larry Mathews
Southern Nuclear
Chairman, MRP Alloy 600 Issue Task Group

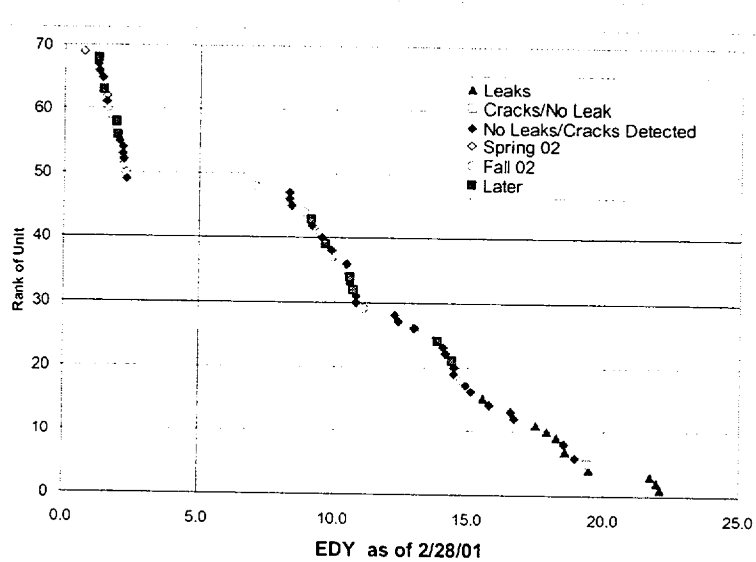
MRP

MRP Presentations

Alloy 600 ITG Status	Mathews	15 min
Alloy 600 Crack Growth Rate	Hickling	45 min
Probabilistic Fracture Mechanics Model	Riccardella	45 min
Collateral Damage	Mathews	10 min
Technical Assessment of DB Degradation Mechanisms	White	30 min
Industry Inspection Plan	Lashley	60 min



Susceptibility Ranking



ACRS 6/5/02.4

Crack growth rate for thick-section Alloy 600 material exposed to PWR primary water

John Hickling, EPRI
for the
MRP Alloy 600 Issue Task Group

MRP

MRP Crack Growth Rate Approach: Overview

- Goal was to establish appropriate CGR guidelines for generic application to thick-section Alloy 600 base material under PWSCC conditions
- MRP panel of international experts on SCC (includes ANL/NRC Research) was established August 2001 and has met several times to date
- Extensive consideration was given to the likely OD environment in the annulus between a leaking CRDM nozzle and the RPV head (prior to Davis Besse incident)
- Relevant arguments remain valid today as long as leak rates are low (typically < 1 liter/h or 0.004 gpm)
- Plant experience has shown this to be the usual case

MRP Crack Growth Rate Approach: Overview (con.)

- Relevant, worldwide CGR results were obtained and re-evaluated so as to screen out inappropriate test data
- Recommended MRP curve for CGR as a function of stress intensity factor (K) was derived taking into account the statistics of heat-to-heat variations and the strong effect of temperature
- Curve was compared with existing field data and recommendations developed for its use in assessing axial/circumferential flaws
- Screened CGR data for base material feeds directly into the probabilistic risk assessment being carried out by SIA

Core members of MRP Expert Panel on PWSCC of Alloy 600

Name	Organization	Name	Organization
Peter Andresen	GE-GRC	Anders Jenssen	Studsvik, Sweden
Steve Attanasio	KAPL	George Licina	SIA
Warren Bamford	Westinghouse	Bill Mills	Bettis
Luisa-Maria Castano	CIEMAT, Spain	Raj Pathania	EPRI
Bill Cullen	NRC Research	Peter Scott	Framatome-ANP, France
Steve Fyfitch	Framatome-ANP	Bill Shack	ANL
John Hickling	EPRI	Glenn White	DEI
Rich Jacko	Westinghouse	Toshio Yonezawa	MHI, Japan
Christer Jansson	SwedPower, Sweden	Ken Yoon	Framatome-ANP

OD Annulus Environment

- Most likely environments
  - Hydrogenated superheated steam, if pressure drop within SCC crack
  - Normal PWR water, if boiling transition well above the J-groove weld
  - Concentrated PWR primary water, if boiling occurs at the exit of SCC crack:
  - situation has been considered in detail for the case usually observed to date, i.e. low leak rates (< 1l/h) and little or no wastage of LAS vessel head
  - full evaluation has not been performed for Davis Besse type situation involving cavity formation and extensive wastage as a consequence of boric acid corrosion

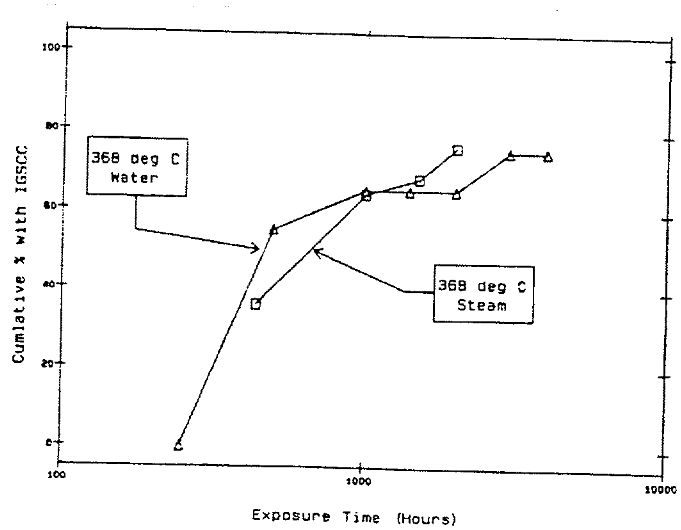
OD Annulus Environment

- **Consideration of oxygen/hydrogen effects common to all three possible environments:**
- Oxygenated crevice environment highly unlikely because:
  - Back diffusion of oxygen is too low compared to counterflow of escaping steam (2 independent assessments based on molecular diffusion models were examined)
  - Oxygen consumption by metal walls would further reduce concentration
  - Presence of hydrogen from leaking water and diffusion through upper head results in a reducing environment
  - Even if concentration of hydrogen was depleted by local boiling, coupling between LAS and Alloy 600 would keep electrochemical potential low
  - Corrosion potential will be close to Ni/NiO equilibrium, resulting in PWSCC susceptibility similar to normal primary water

OD Annulus Environment

- Possible environment #1: hydrogenated steam
- Numerous laboratory tests in (pure) hydrogenated steam (e.g. Economy et al., 1986 – 1995) have shown that PWSCC response is similar to that in normal PWR primary water at the same temperature

OD Annulus Environment



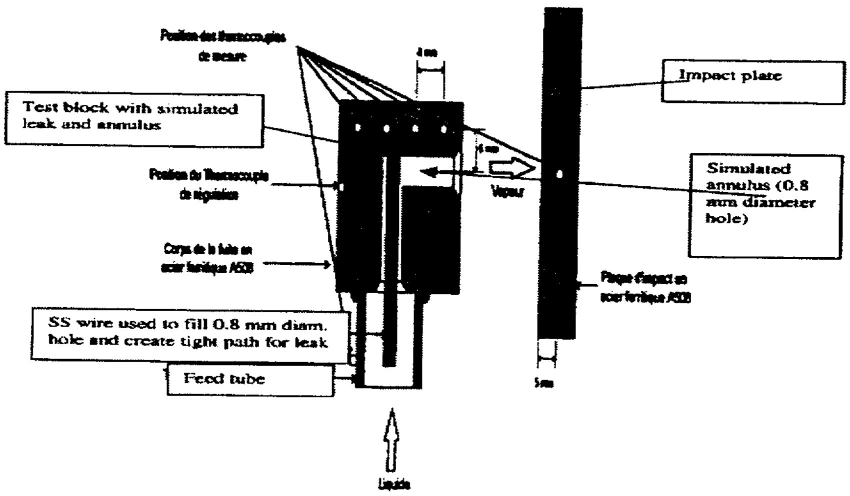
OD Annulus Environment

- Possible environment #2: PWR primary water within normal specifications
- Main focus of subsequent CGR data evaluation by Expert Panel

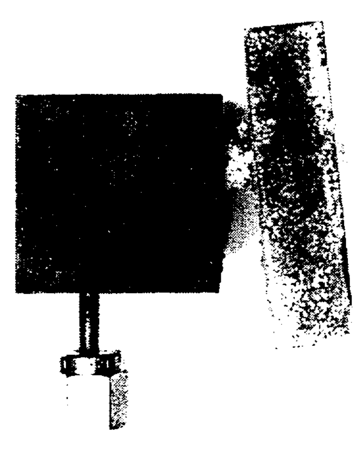
OD Annulus Environment

- Possible environment #3: Concentrated PWR primary water. For low leak rates (< 1 l/h) as mostly observed to date:
  - pH_T between 4 and 9.4 based on MULTEQ calculations
  - Actual pH_T range expected to be narrower due to precipitation of complex lithium-iron borates
  - A French experiment simulating a leak detected such borate compounds and estimated that pH_T of the liquid phase was between 7 and 8
  - A further French test involving slow concentration of a fixed volume of primary water showed no formation of caustic after conc. factor 10^3 (calculated pH_T was ~ 4.5)
  - Cleaning practices followed during head assembly should minimize contamination by sulfates and chlorides and steam flushing will help to remove any residual impurities

OD Annulus Environment: setup for CEA simulation test



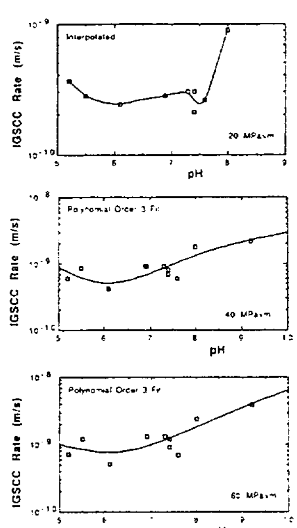
OD Annulus Environment: result of CEA simulation test



OD Annulus Environment

- Possible environment #3: Concentrated PWR primary water (con.)
  - Ohio State study shows no significant effect of pH_T on PWSCC CGR between values of 5 and 8.5 at 330° C
  - For pH_T values between 7.5 and 9, CGR increases slightly, but acceleration factor only around 1.5 even for $pH_T = 9$
  - Expert Panel recommended that a factor of **x2 on CGR** should conservatively cover uncertainties in the exact composition of the annulus chemistry for $4 < pH_T < 9$
  - More acid environments as a result of large leak rates and local cooling of head were NOT considered, but limited data (Berge et al., 1997) suggests that high chloride and oxygen levels are required for IGSCC of Alloy 600 to occur

OD Annulus Environment: results of Ohio State study on effect of pH



MRP CGR database for Alloy 600: screening of available data

- Key technical issues on which screening was based:
- Material within specifications including condition/heat treatment
- Composition within material specifications
- Mechanical strength properties
- ASTM specimen size criteria
- Straightness criteria and crack front mapping
- Standard procedure for welds
- Environment (Li, B, and H_2 concentrations; hydrogen control; temperature; ECP)
- Loop configuration (e.g., once-through, refreshed, static with H_2 control) and flow rate
- Water chemistry confirmation (e.g., Cl, SO_4)
- Crack length confirmed by destructive examination
- Transgranular fraction on fractograph
- Fraction SCC along crack front
- Changing conditions during a test?
- Active constant or cyclic loading versus constant displacement loading (e.g., wedge loading)
- Load during "cool down"
- Crack length versus time data
- SCC crack increment
- Precision on measurement of crack length increase

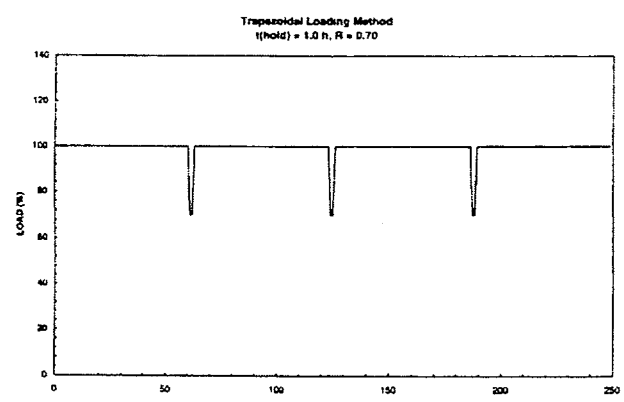
MRP CGR database for Alloy 600: screening of available data

- Screening of data involved 3 iterative steps:
- request to laboratory from which the data originated to re-examine suitability in the light of the key technical issues identified by the Expert Panel (most of the unsuitable data points were identified at this stage)
- further screening by EPRI to remove incompatible data (e.g. where no information on average CGRs was available) and to add conservatism by eliminating displacement-loaded (WOL) specimens with less than 50% initiation of IGSCC across width of fatigue pre-crack
- re-examination of borderline cases by the whole Expert Panel

MRP CGR database for Alloy 600: screening of available data

- No attention was paid to numerous tests where **no** crack growth due to PWSCC was actually observed
- Result of data screening was elimination from further consideration of 203 CGR data points for one or more reasons (main reason individually documented in report)
- Consolidated database contains 158 data points for average CGR during each test (consistent with ASTM practice for measuring fatigue CGRs) plotted at a representative K value (ranged from 14.3 to 54.0 MPa√m)
- All were obtained in controlled primary water using fracture mechanics specimens under either constant load or constant displacement conditions
- Some tests under active load involved periodic unloading (considered to give a potential accelerating effect which is relatively small, at least for susceptible materials)

MRP CGR database for Alloy 600: periodic unloading used in W tests



MRP CGR database for Alloy 600

- Domestic and Overseas material suppliers represented:
  - B&WTP, Huntington, INCO, Standard Steel
  - Creusot-Ondaine, Creusot-Imphy, Tecphy, Arbed, VDM, Schneider-Creusot, Sandvik, Sumitomo Metal
- 26 heats of material with at least 1 screened data point per heat (maximum # = 32 for B&WTP heat 91069)
- Multiple product forms
  - Thick walled tube
  - Forged bar
  - Rolled bar
  - Forged plate
  - Rolled plate
- Information on thermal processing history of material incomplete, so likely effects could not be systematically considered in a deterministic way

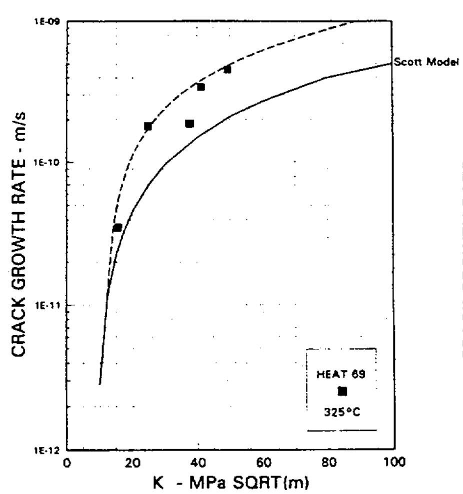
MRP CGR database for Alloy 600

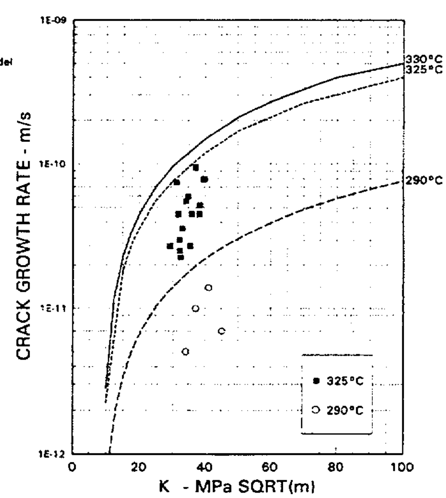
- Multiple Labs
  - Westinghouse, U. S.
  - EdF, France
  - CEA, France
  - CIEMAT, Spain
  - Studsvik, Sweden
- Test temperatures ranged from 290-363 °C (554-686 °F)
- CGR through PWSCC of flaws in Alloy 600 is known to be **highly temperature dependent**, so
- All CGR data points were adjusted to a common reference point (the most typical test temperature) of 325 °C (617 °F) using an activation energy of 130 kJ/mole (31 kcal/mole)
- This represents a consensus value for Alloy 600 crack **growth** data

Derivation of MRP CGR Curve

- Because of the known importance of material processing parameters on CGR, the initial evaluation was based on a heat-by-heat analysis of the screened database
- Insufficient data points were available from any single heat over a wide range of K values to determine the form of CGR dependence on stress intensity factor
- Shape of curve to be fitted was adopted from the Scott equation, originally developed (1991) using inspection data for axial cracks in the roll transitions of SG tubes
- This much larger database of CGR measurements is considered to provide a more reliable indicator for the form of the CGR versus K dependence:
- $da/dt = \alpha(K-9)^{\beta}$ with Scott exponent $\beta = 1.16$

Derivation of MRP CGR Curve: examples of original results (2 labs)





Derivation of MRP CGR Curve

- Adoption of the Scott equation results in an apparent crack tip stress intensity factor threshold, K_{th}, of 9 MPa√m (8.19 ksi√in).
- However, no actual CGR data for CRDM nozzle materials is available at K values < approx. 15 MPa√m
- Not critical for intended use to analyze detected axial flaws, since K values will already be above this
- Use of the Scott exponent β = 1.16 may result in conservative estimations of CGR at high K values, since some test and field data appears to indicate the appearance of a plateau in the curve

Derivation of MRP CGR Curve

- Assuming the form of the Scott equation, a mean power-law constant α was then calculated for each of the 26 heats of material in the database according to

$$f(\alpha,\beta)=\sum_{i=1}^{n}\{\ln(\dot{a}_i)-[\ln(\alpha)+\beta\ln(K_i-K_{th})]\}^2$$

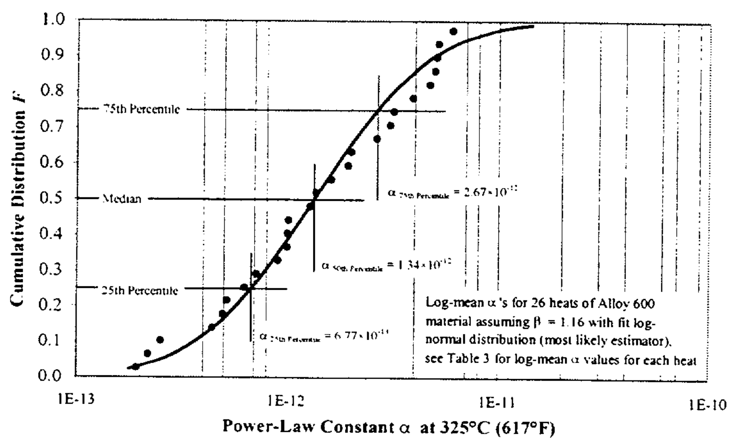
Derivation of MRP CGR Curve

Heat Rank	Material Supplier	Product Form	Number of Data Points	Log Mean Power-Law Constant α at 325°C (617°F) SI Units[1]	Log Mean Power-Law Constant α at 325°C (617°F) English Units[2]
1	Creusot-Imphy	Forged Bar	21	6.01E-12	8.32E-03
2	B&WTP	Thick-wall Tube	4	5.16E-12	7.15E-03
3	French Supplier	CRDM Nozzle	9	5.08E-12	7.03E-03
4	Tecphy	Rolled Bar	7	4.96E-12	6.88E-03
5	B&WTP	Thick-wall Tube	4	4.71E-12	6.52E-03
6	VDM	Rolled Plate	2	3.92E-12	5.43E-03
7	Schneider-Creusot	Forged Bar	1	3.19E-12	4.42E-03
8	B&WTP	Thick-wall Tube	32	3.07E-12	4.25E-03
9	B&WTP	Thick-wall Tube	1	2.65E-12	3.68E-03
10	Arbed	CRDM Nozzle	3	2.01E-12	2.79E-03
11	Creusot-Imphy	Forged Plate	1	1.94E-12	2.69E-03
12	Schneider-Creusot	Forged Bar	1	1.62E-12	2.24E-03
13	Huntington	Thick-wall Tube	1	1.37E-12	1.90E-03
14	Huntington	Rolled Plate	14	1.29E-12	1.78E-03
15	*Not Listed*	Forged Bar	2	1.02E-12	1.41E-03
16	Sumitomo Metal	Thick-wall Tube	1	1.01E-12	1.40E-03
17	Sandvik	Thick-wall Tube	27	1.00E-12	1.39E-03
18	Standard Steel	Forged Bar	1	9.09E-13	1.26E-03
19	Huntington	Thick-wall Tube	12	7.21E-13	9.99E-04
20	*Not Listed*	Forged Bar	3	6.31E-13	8.74E-04
21	Tecphy	Rolled Bar	1	5.18E-13	7.18E-04
22	Huntington	Plate	1	4.97E-13	6.89E-04
23	Creusot-Ondaine	Forged Bar	4	4.44E-13	6.15E-04
24	Inco	Rolled Bar	1	2.51E-13	3.48E-04
25	Sandvik	Thick-wall Tube	2	2.18E-13	3.03E-04
26	Huntington	Thick-wall Tube	2	1.93E-13	2.67E-04
		Log-Mean for All Data Points	158	1.96E-12	2.72E-03
		Log-Mean of Heat Log-Means	26 Heats	1.34E-12	1.86E-03

Derivation of MRP CGR Curve

- Distribution describing CGR variability was then taken as the log-normal fit to the ordered median ranking of the α values for the 26 heats, using most likely estimator methodology

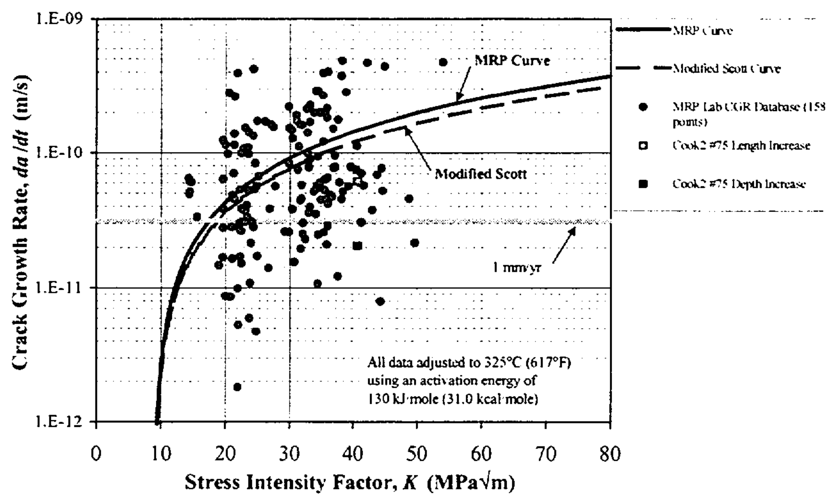
Derivation of MRP CGR Curve



Derivation of MRP CGR Curve

- Recommended CGR curve is based on 75th percentile level of the distribution of CGR variability as a function of material heat and represents the mean of the upper half of the distribution
- MRP curve lies approx. 20% above the Scott equation
- Approach is consistent with ASME code considerations, where the goal is to make a best estimate of crack growth
- Addresses the concern that cracking detected in operating plants would tend to be in components fabricated from more susceptible Alloy 600 heats
- Likely that CRDM nozzles supplied by some material vendors may crack at a significantly lower rate than indicated by the MRP curve

Derivation of MRP CGR Curve



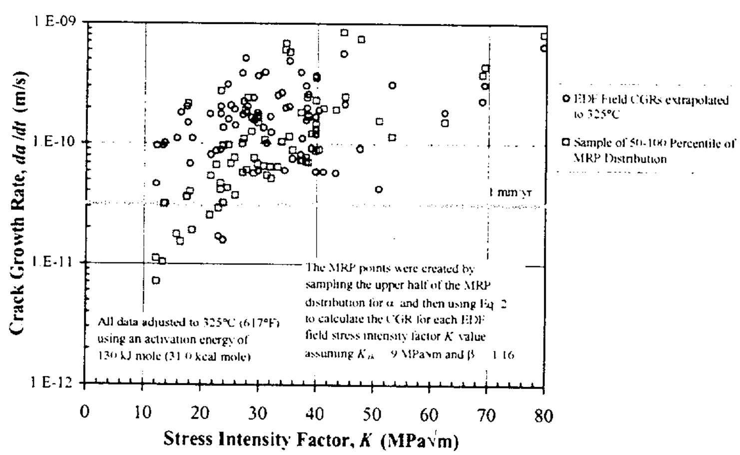
Comparison of MRP database with available plant CGR data

- Large uncertainties exist in reported values of CGRs from operating plants due to:
- uncertainties in ultrasonic measurements of crack size at two or more different times
- uncertainties in the estimates of K, which depend on estimates of residual stress
- uncertainties in the actual operating temperatures of CRDM nozzles in different plants and in different countries
- Limited US data (from D.C. Cook nozzle #75) lie well below the MRP curve

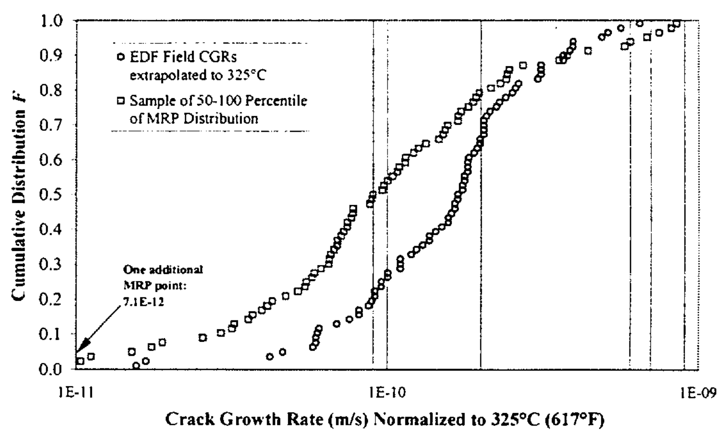
Comparison of MRP database with available plant CGR data

- Most extensive measurements of CGR in operating plants are from France.
- The data have been extrapolated by the MRP from the reported operating temperatures in the plants to a standard value of 325°C for comparison purposes
- Values are compared with the results of predicted CGRs calculated by using:
  - the reported *K* values for the French field data
  - random sampling from the upper half of the MRP distribution for CGRs
  - the *K*-dependence of the Scott equation

Comparison of MRP database with available plant CGR data



Comparison of MRP database with available plant CGR data



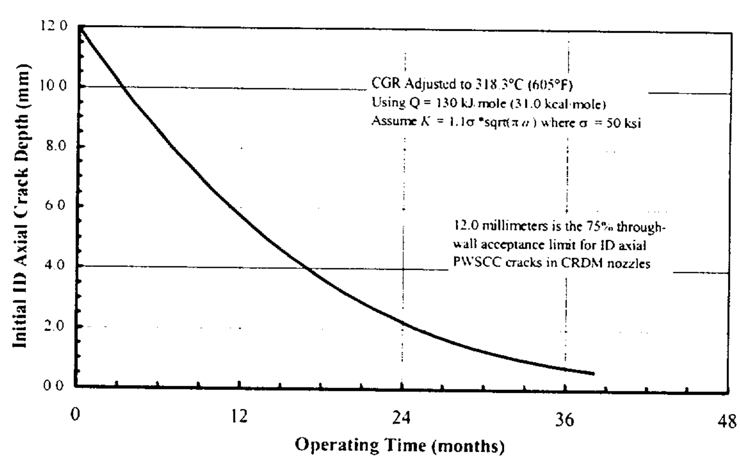
Comparison of MRP database with available plant CGR data

- Agreement with French field data is quite reasonable considering the uncertainties involved
- Supports the choice of the 75th percentile curve from the MRP distribution as representative of the rates expected for axial crack growth in CRDM nozzles
- In no case did the actual measured CGR in the through-wall direction exceed 4 mm/yr (0.16 in/yr) for data from French plants of fundamentally Westinghouse design
- This figure was adopted in France, independent of nominal upper head temperature, to justify continued operation with axial cracks up to 11 mm (0.43 inches) deep for a one-year fuel cycle

Application of MRP CGR Curve

- The MRP recommended curve is intended for disposition of detected PWSCC flaws in thick-walled Alloy 600 components exposed to normal PWR primary water
- Thus it is directly applicable to **axial ID flaws** detected in CRDM nozzle pressure boundary base material and to flaws below the J-groove weld
- Its use at low crack-tip stress intensity factors (< approximately 15 MPa√m) would involve assumptions not currently substantiated by actual CGR data for CRDM nozzle materials
- In practice, however, K values will already be above this

Application of MRP CGR curve: example calculation (ID axial flaw)



CGR in OD Annulus Environment

- For evaluation of (hypothetical) **OD cracking** above the J-groove weld, the MRP recommends that CGR values from the curve be multiplied by 2x to allow for uncertainty in exact composition of the external chemical environment
- A subgroup of the Expert Panel have revisted the relevant arguments in the light of the Davis Besse experience and found that they remain correct as long as leak rates are low (typically less than 1 liter/h or 0.004 gpm)
- Plant experience has shown this to be the usual case

CGR in OD Annulus Environment

- Analysis would no longer be valid, however, if leak rates were sufficiently high to result in a large, local decrease in temperature and appreciable corrosion of low-alloy steel
- Limited data on SCC in concentrated boric acid solutions indicate that
  - Alloy 600 is very resistant to TGSCC (material design basis)
  - high levels of oxygen <u>and</u> chloride are necessary for intergranular cracking to occur at all
  - effects are then worse at intermediate temperatures, suggesting that mechanism is different from PWSCC

ACRS 6/5/02.44

Probabilistic Fracture Mechanics Analysis of CRDM Nozzles

Presented at:

ACRS Meeting
Rockville, MD

Presented by:

Dr. Peter C. Riccardella
Structural Integrity Associates

June 5, 2002

Outline of Presentation

- Overview of Probabilistic Fracture Mechanics Methodology for RPV Top Head Nozzle Cracking
- PFM Analyses in support of MRP Inspection Plan
  - Susceptibility Categories
  - Inspection Types and Frequencies

Key Elements of RPV Head Nozzle PFM Analysis

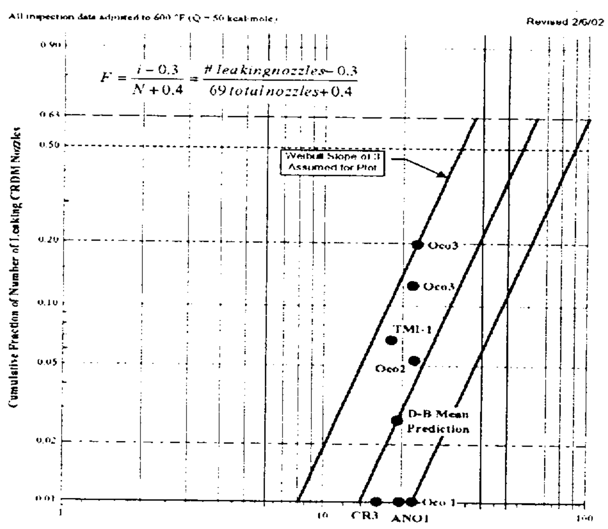
- Probability of Leakage
  - Weibull Model based on Experience to Date
  - Incorporated into Monte Carlo Model
- Fracture mechanics modeling for Stress Intensity Factors
  - Through-Wall Cracks
  - Part Through Wall Cracks
- Stress Corrosion Crack Growth Statistics
- Effect of Inspections
  - Inspection Interval
  - Inspection Reliability

Weibull Models for Leakage

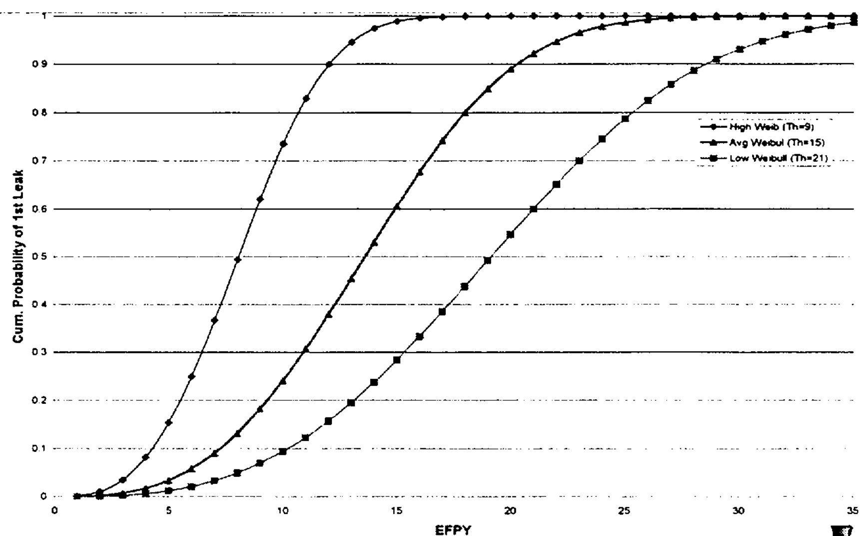
- Analysis by Dominion Engineering – B&W plants w/ Weibull slope of 3
  - Weibull Slope = 3.0
  - Weibull Theta* = 15.36 (avg.) ; 9.094 (worst case)

*Theta = Characteristic time to 63.3% probability of at least one leak in a head.

Dominion Engineering Weibull Analysis (Theta = 3)



Weibull Distributions used in PFM
β=3; θ=15 $\pm$ 6



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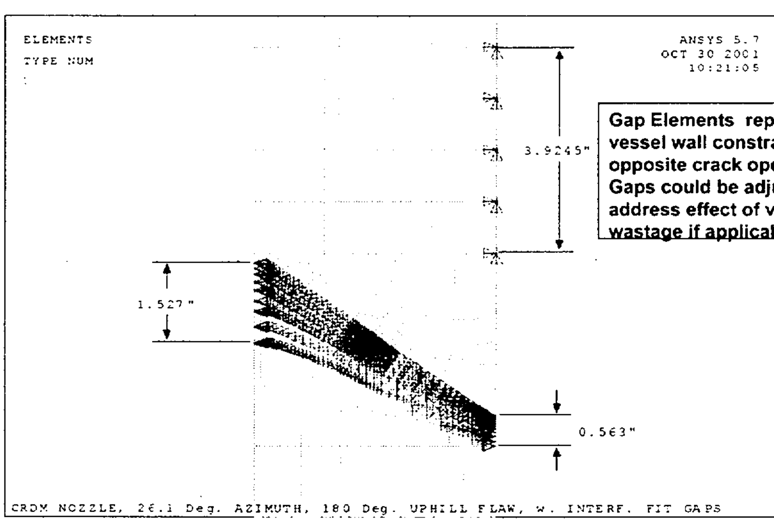
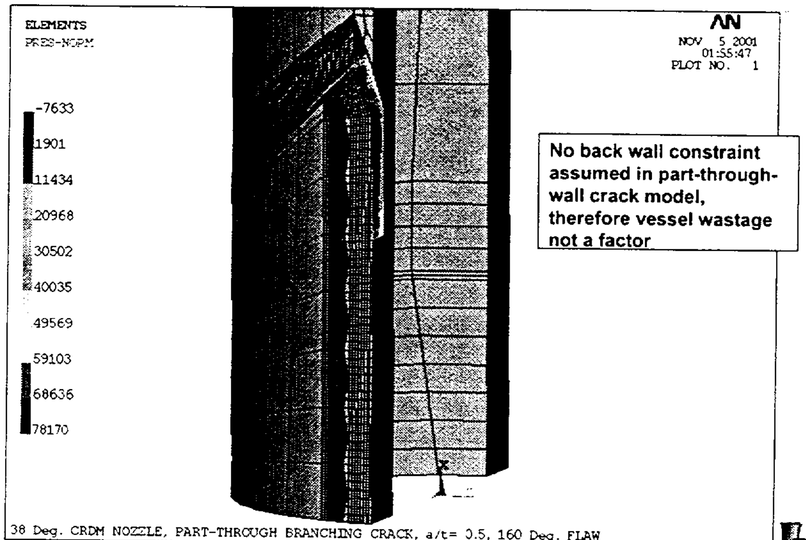
Fracture Mechanics Model
Through-Wall Crack



Gap Elements represent vessel wall constraint opposite crack opening – Gaps could be adjusted to address effect of vessel wastage if applicable.

ACRS 6/5/02.51

Part-Through-Wall Flaw Model

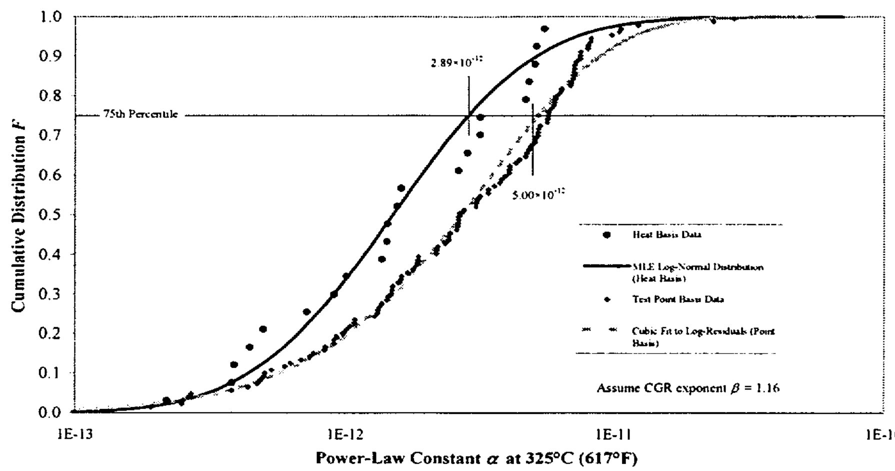


Stress Intensity Factor Results B&W Type Plant

High Yield,
Large Gap Case

Nozzle Angle	Circumferential Crack Length		Stress Intensity 1/2	
	Degrees	Inches	Uphill	Downhill
0°	30	0.9664	20.8	N/A
	70	2.2550	18.8	N/A
	160	5.1540	20.3	N/A
	180	5.3140	0.64	N/A
	220	6.4950	0.63	N/A
	260	7.6760	0.63	N/A
	300	8.8570	0.62	N/A
18°	30	1.0170	27.2	27.2
	70	2.3730	24.0	24.0
	160	5.4240	24.5	24.5
	180	5.5920	23.4	1.0
	220	6.8350	23.8	2.4
	260	8.0770	26.9	6.0
	300	9.3200	26.5	11.5
26°	30	1.0830	29.7	29.7
	70	2.5260	26.1	26.1
	160	5.7750	26.5	26.5
	180	5.9530	28.4	0.4
	220	7.2760	23.2	1.7
	260	8.5990	23.6	7.5
	300	9.9220	24.9	16.6
38°	30	1.2380	34.4	34.4
	70	2.8830	27.1	27.1
	160	6.6020	29.2	29.2
	180	6.8060	37.7	4.5
	220	8.3190	31.2	6.7
	260	9.8310	26.6	12.7
	300	11.3440	29.9	25.9

SCC Crack Growth Data for Nozzle Material in Reactor Environment



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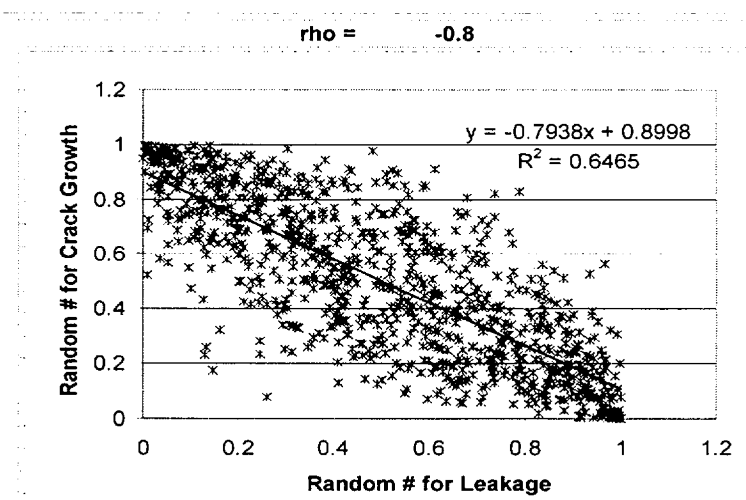
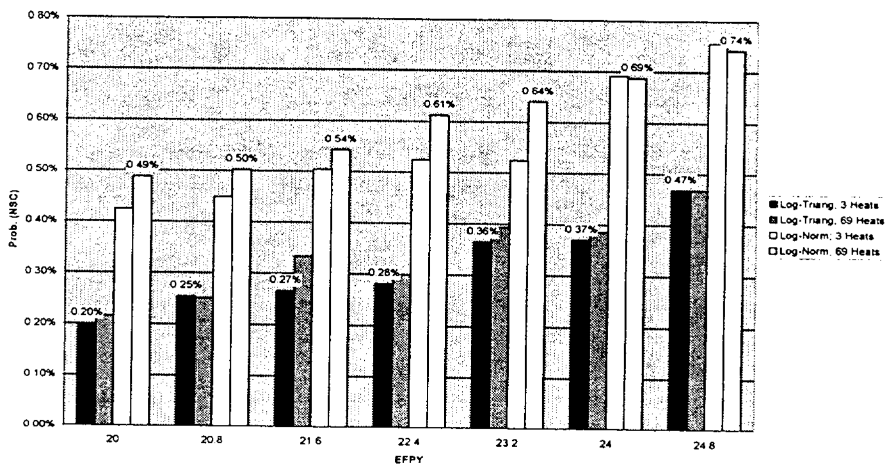
CGR Initiation vs. Growth Correlation

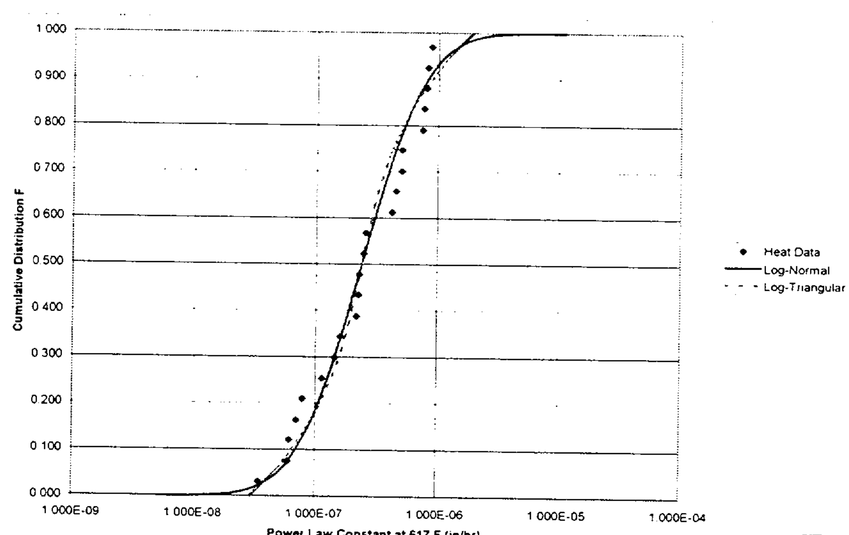


ACRS 6/5/02.55

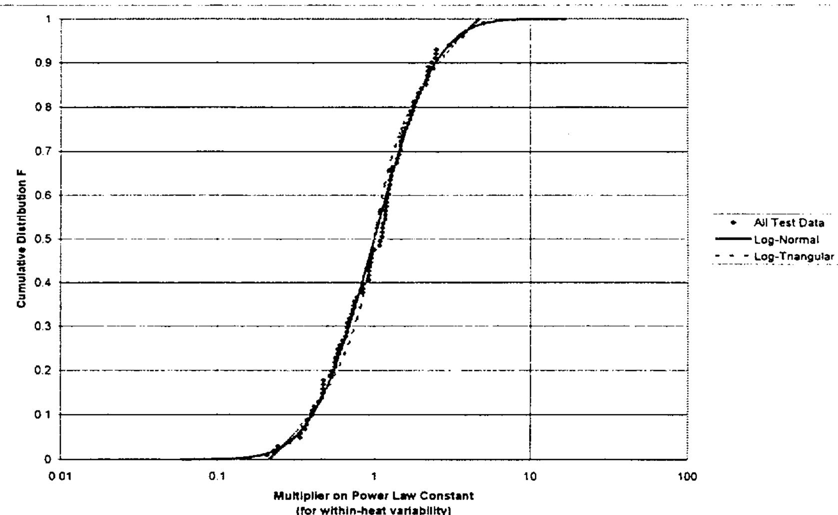
Typical PFM Results (602°F Head Temp.; No Inspection)



CGR Distributions Based on Heat Data

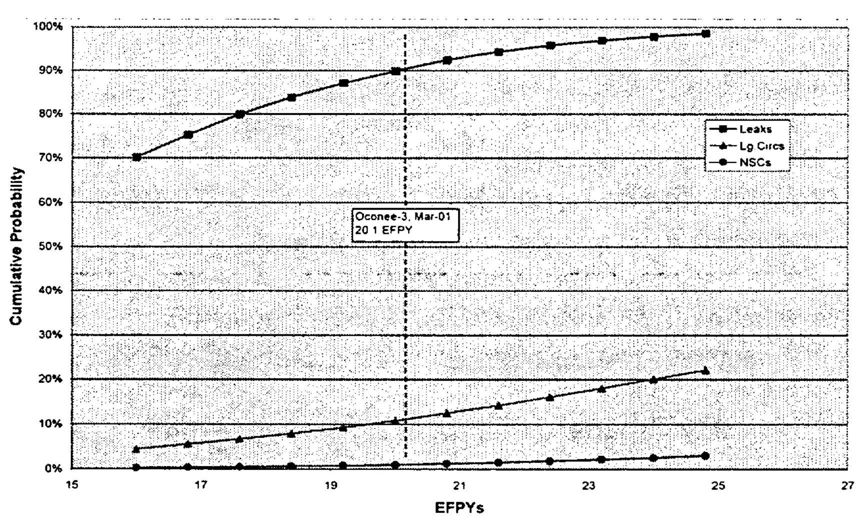


Multiplier on CGR Distribution for Within-Heat Variability



ACRS 6/5/02.58

Benchmarking of PFM Results with respect to B&W Plants



ACRS 6/5/02.59

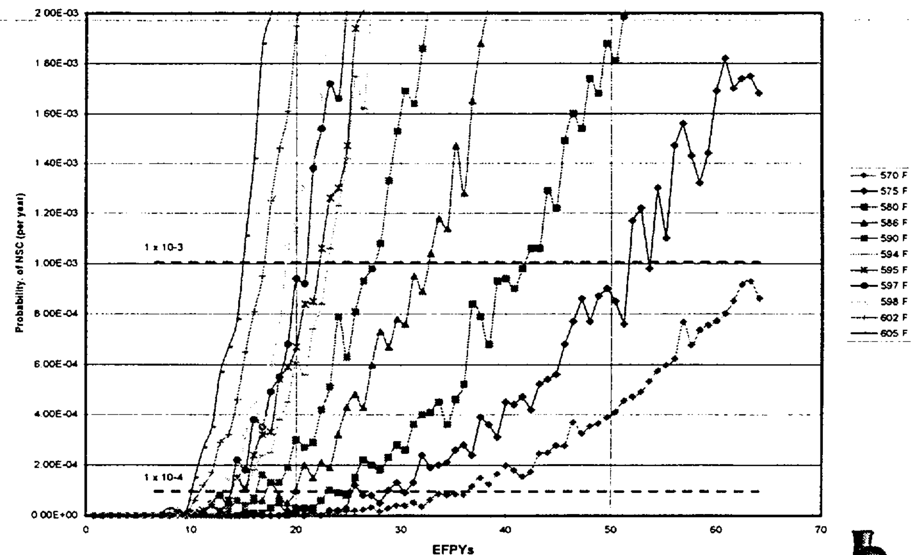
Technical Basis for Inspection Plan
- Basic Concept -

- Start with "benchmarked" analysis parameters from B&W plant analysis
- Analyze plants at various head temperatures
- Set risk categories based on probability of Net Section Collapse (per year) and cumulative leakage probability
- Set inspection intervals based on effect of various inspections on probability of Net Section Collapse (per year)

"Benchmarked" Analysis Parameters

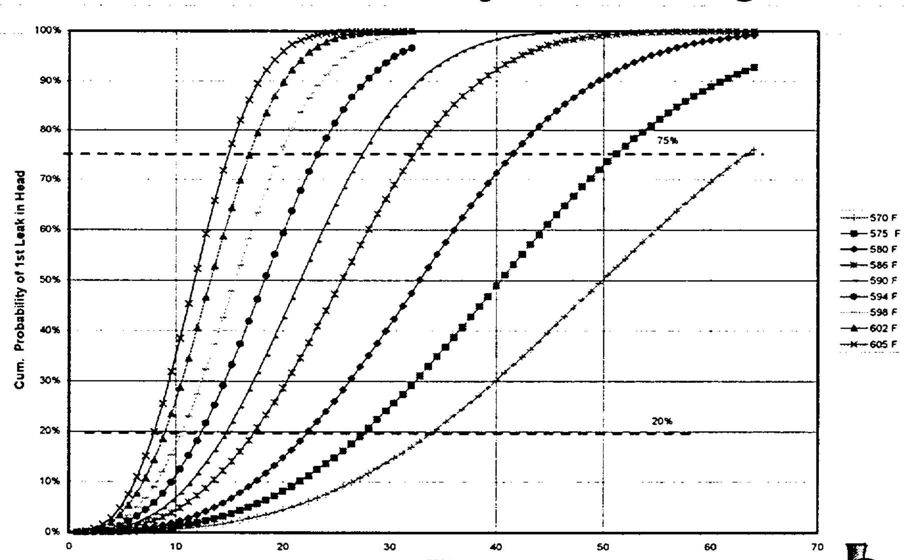
- Head Temperature: Various from 560°F to 605°F
- Weibull Parameters:
  - Slope = 3
  - Beta = 15 ± 6 (Triangular)
- Crack Growth Rate Statistics
  - Heat-to-Heat - Log-Triangular: -15.25 ± 2.212
  - Within Heat – Log-Triangular: 0 ± 1.6
- Crack Growth vs. Leakage Correlation Factors
  - 0.8 – Heat-to-Heat
  - 0.8 – Within-Heat
- Acceptability Criteria: PDF of NSC < 1×10^{-3} per year

Inspection Plan PFM Runs: Probability of NSC (per year)



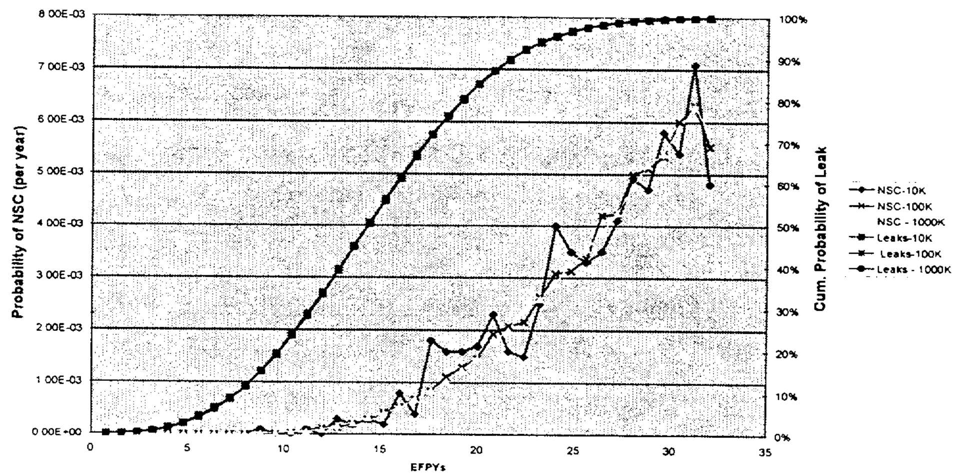
ACRS 6/5/02.62

Inspection Plan PFM Runs: Cum. Probability of Leakage



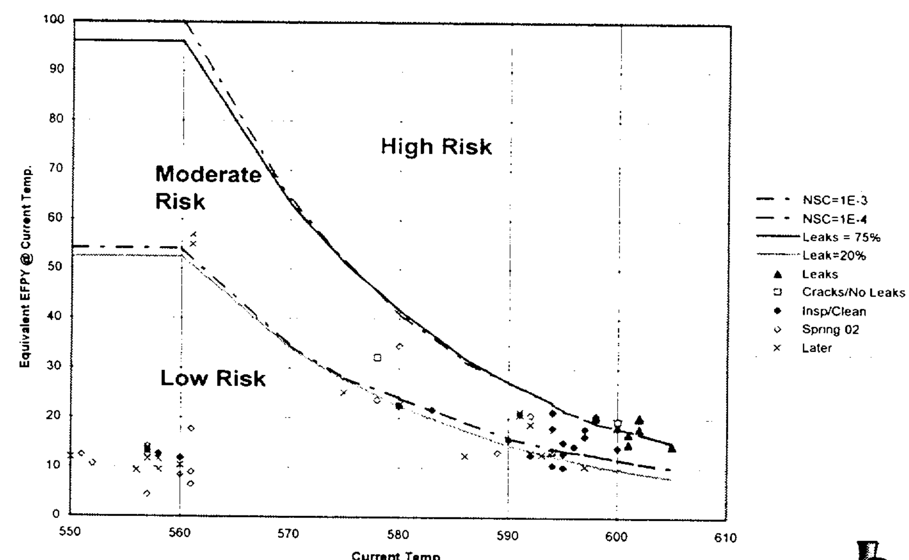
ACRS 6/5/02.63

PFM Convergence Study (@ 600°F)



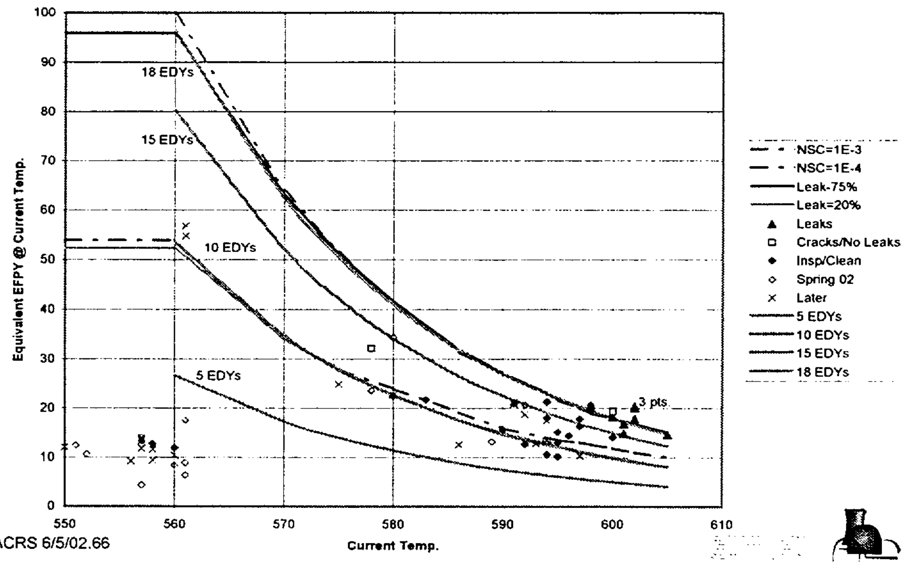
ACRS 6/5/02.64

Definition of Susceptibility Categories Based on PFM Results



ACRS 6/5/02.65

Correspondence of Susceptibility Categories to EDYs

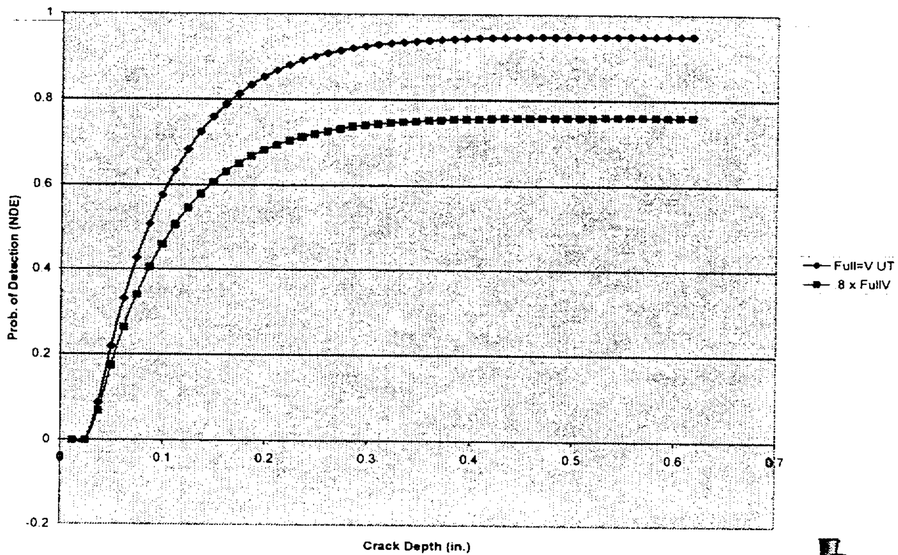


Inspection Frequency Runs: Probabilities of Detection

- Bare Metal Visual Inspections (BMV)
  - Initial POD = 0.6
  - POD for Subsequent Exams = 0.2 x Initial POD (when Leakage missed)
- Non-Destructive Examinations (NDE)
  - POD = f(crack depth) per EPRI-TR-102074[1]
  - 80% Coverage Assumed

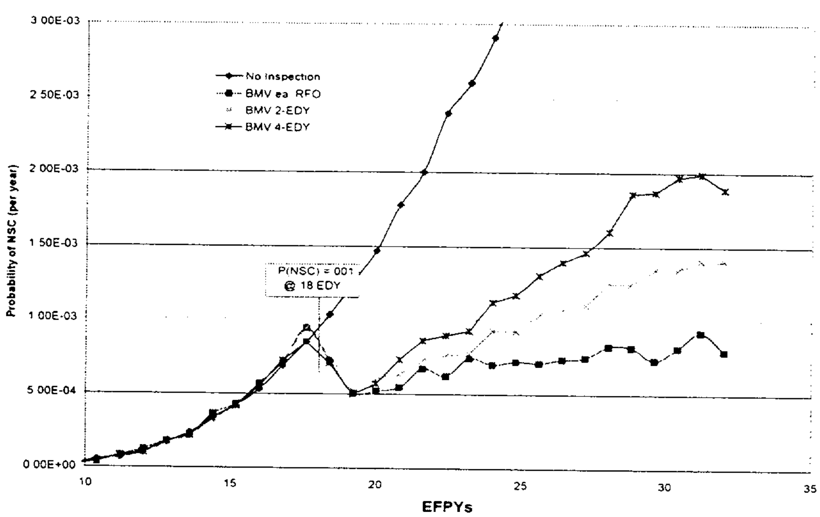
[1]Dimitrijevic, V. and Ammirato, F., "Use of Nondestructive Evaluation Data to Improve Analysis of Reactor Pressure Vessel Integrity, " EPRI Report TR-102074, Yankee Atomic Electric Co. March 1993

Probability of Detection Curves for NDE



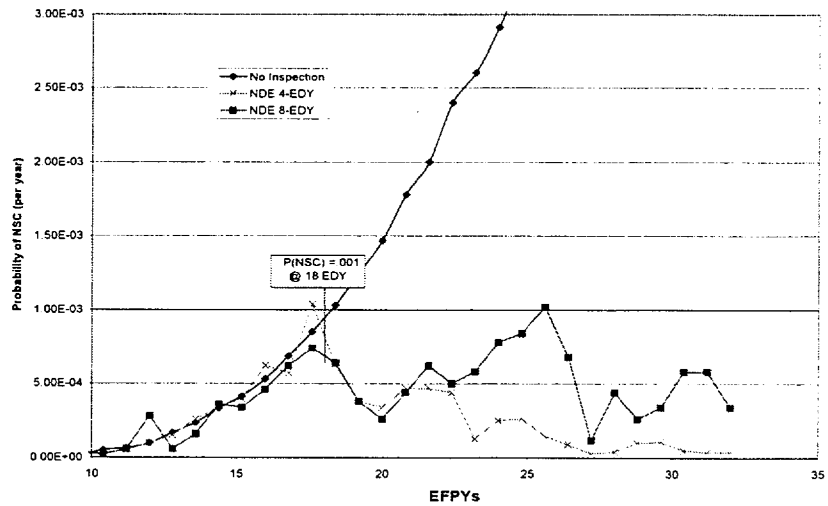
ACRS 6/5/02.68

Inspection Plan Technical Basis: Effect of Visual Inspection Runs



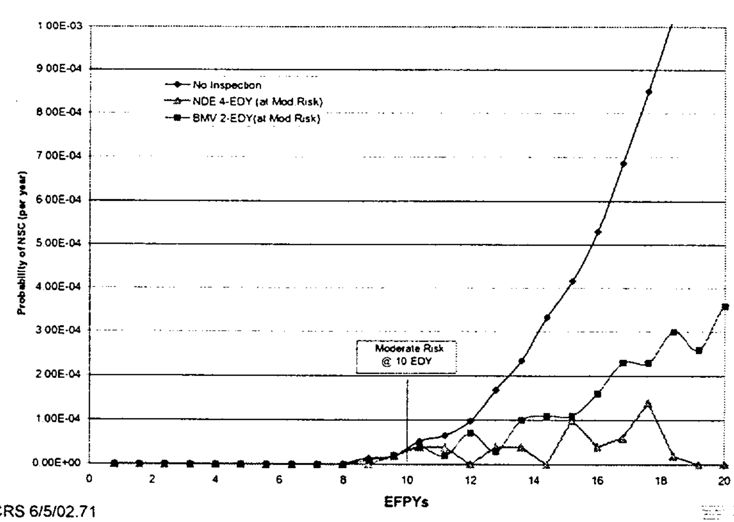
ACRS 6/5/02.69

Inspection Plan Technical Basis: Effect of NDE Inspection



ACRS 6/5/02.70

Effect of Inspections upon Entering Moderate Category



ACRS 6/5/02.71

Deterministic Crack Growth Analyses

- Uses Expert Panel recommended crack growth law
  - 2 x 75th Percentile of all data
  - da/dt = C(K-8.19)1.16

Temperature (°F)	C
580	3.604×10^{-7}
590	4.665×10^{-7}
600	6.008×10^{-7}
602	6.316×10^{-7}
605	6.806×10^{-7}

Deterministic Crack Growth Analyses

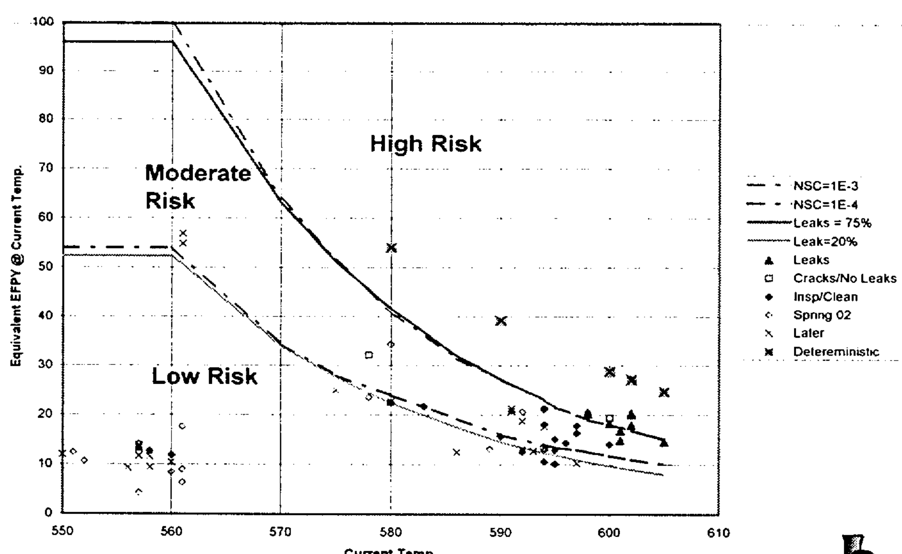
- Uses Stress Intensity Factors from plant specific analysis of Westinghouse plant
  - High Angle Nozzle (43.5° nozzle angle)
  - Higher Ks than B&W plant results

Circ. Crack Length		K
Degrees	Inches	Ksi*in $^{1/2}$
30	1.16	34.4
70	2.70	27.1
160	6.16	29.2
180	6.34	47.2
220	7.75	51.9
260	9.16	58.1
300	10.57	63.7

Deterministic Crack Growth Analysis Results

Temperature (°F)	Time for Initial Flaw Size of 30° Circumference to Grow to 165° and 300° (EFPY)	
	Westinghouse-Type Plant	
	165°	300°
580	23.7	31.7
590	18.3	24.6
600	14.2	19.1
602	13.5	18.2
605	12.5	16.8

Deterministic Crack Growth Results Added to Susceptibility Category Plot



Conclusions

- PFM Incorporates:
  - Weibull model of time to leakage
  - Finite Element Fracture Mechanics model for B&W type head
  - Crack growth rate statistics from Expert Panel
  - Log-Triangular and Log-Normal CGR Distributions
  - Correlation between time to leakage and CGR
  - Effect of various inspection types, intervals and POD
- Inspection Plan Technical Basis Runs:
  - Start with "benchmarked" analysis parameters from B&W plant analysis
  - Analyze plants at various head temperatures
  - Set risk categories based on probability of Net Section Collapse (per year) and cumulative leakage probability
  - Set inspection intervals based on effect of various inspection types and frequency on probability of Net Section Collapse (per year)

Conclusions (cont'd)

- Susceptibility Categories Based on PFM Results
  - Low –Risk:: $0 < \text{EDYs} < 10$
  - Moderate Risk: $10 \leq \text{EDYs} < 18$
  - High Risk: $18 \leq \text{EDYs}$
- Inspection Type and Frequency Results
  - Inspection cases run with conservative POD assumptions
  - BMV each RFO upon entering High Risk Category reduces probability of NSC to acceptable level indefinitely
  - NDE every 4 EDYs upon entering High Risk Category reduces probability of NSC to essentially nil
- Deterministic Crack Growth Results
  - Conservatively bounds times from moderate to high risk susceptibility regions

Collateral Damage

MRP

Collateral Damage

- MRP Performed an Initial Qualitative Assessment of Collateral Damage from CRDM Nozzle Ejection
  - Indicated impact on Conditional Core Damage Probability should be insignificant
  - No impact on ECCS capabilities
  - Effect on shutdown reactivity capabilities minimal
    - Impact and jet loads should not affect significant number of rods
    - Loose parts also have only limited impact

Collateral Damage

- MRP to review Davis-Besse's collateral damage work
  - Expect industry results to be similar
- Discuss with NRC technical staff their assessment of impact of collateral damage on CCDP
- Finalize MRP collateral damage assessment and include in Final Safety Assessment

Technical Assessment of Davis-Besse Degradation

Prepared for Meeting of the
ACRS Materials and Metallurgy
and Plant Operations Subcommittees
June 5, 2002

Prepared by:
G. White
C. Marks
S. Hunt
Dominion Engineering, Inc.

MRP

Contents

- Purpose and Approach
- Material Loss Mechanisms
  - Corrosion mechanisms
  - Erosion mechanisms
  - Flow accelerated corrosion
- Degradation Progression
- Boric Acid Corrosion Tests Simulating Nozzle Leakage

NOTE: Additional information and results are provided in the May 22, 2002, presentation to the NRC staff on this subject, which is available on the NRC website area for reactor head degradation.

Purpose and Approach

Purpose

- The purpose of the technical assessments is to complement plant experience in answering the following questions:
  - If a significant amount of RPV head material loss occurs, will it be detectable visually from above the head (either directly or through the presence of deposits)?
  - Is there a period of time following initiation of a through-wall leak for which there is assurance that no unacceptable reactor vessel head corrosion will occur?
- In addition, the technical assessments also address current questions regarding the progression of material loss mechanisms (i.e., understanding of degradation progression)

Approach

- The basic approach is to examine how the various potential material loss mechanisms vary as the leak rate is increased from 10^{-6} to 1.0 gpm and the initial tight nozzle annulus becomes a large cavity through material loss. Evaluations focus on:
  - Thermal-hydraulic environment
  - Chemical environment
  - Properties of boric acid and boron compounds
  - Relevant experimental results and plant experience
- The leak rate is expected to be the key parameter:
  - Expansion cooling increases with leak rate, potentially permitting a liquid film to reach the top head surface
  - Increasing leak rates result in higher velocities and potentially erosion or flow accelerated corrosion

Approach (continued)

- The leak rate also determines the amount of boric acid deposits that exit the pressure boundary
- The results of corrosion and erosion rate evaluations are used to bound:
  - The timeframe for significant degradation
  - The volume of low alloy steel material loss versus the volume of deposits produced

Material Loss Mechanisms

- Corrosion mechanisms
- Erosion mechanisms
- Flow accelerated corrosion

Material Loss Mechanisms
Overview

- Chemical Mechanisms
  - Low-oxygen, boric acid corrosion (deaerated, concentrated boric acid solutions)
  - Dry boric acid or boric oxide crystal corrosion
  - Classic crevice corrosion (conductive liquid in the crevice forms an ionic path to allow dissolution deep in crevice remote from oxygen at crevice mouth)
  - Galvanic corrosion (driving corrosion potential due to dissimilar metal couple between Alloy 600 nozzle and low-alloy-steel (LAS) head)
  - "Classic" boric acid corrosion (aerated, concentrated boric acid solutions)
  - Molten boric acid corrosion

Material Loss Mechanisms
Overview (continued)

- Flow-Enhanced Chemical Mechanisms
  - Two-phase flow accelerated corrosion (FAC) (low oxygen; boric acid not required)
- Mechanical Mechanisms
  - Droplet or solid particle impingement erosion
  - Flashing-induced erosion
  - Steam cutting erosion
  - Single-phase erosion

Material Loss Mechanisms
Matrix

PRELIMINARY

Possible Material Loss Mechanisms	Extent of Wastage: Initial Tight Annulus	Enlarged Annulus	Small Cavity	Large Cavity
Deaerated Boric Acid Corrosion Conc. Boric Acid Corrosion but $DO_2 \approx$ 0-10 ppb	Low rates			
Dry BA or Boric Oxide Crystal Corrosion Corrosion in Contact with Dry Crystals and Humidity	Low rates			
Single-Phase Erosion Potential Erosion if High Steam Velocities	Possible for high leak rates	Less likely than for tight annulus		Large flow area precludes high velocities
Flow Accelerated Corrosion (FAC) Low-Oxygen Dissolution through Surface Oxides	Possible if liquid velocities high enough and temperature low enough			Unlikely as oxygen stabilizes
Impingement / Flashing-Induced Erosion Droplet and Particle Impact Opposite Crack Outlet	Possible if droplets right size and momentum			
Crevice Corrosion Liquid Ionic Path from Top Head Surface	Believed not to be likely because low alloy steel does not passivate in an aerated, concentrated boric acid			Not possible because no crevice geometry
"Occluded Region" Galvanic Corrosion Driven by Potential Difference Btw Dissimilar Metals	Possible at locations where liquid solution exists			
"Molten" Boric Acid Corrosion Corrosion in Pure or Nearly Pure Melted BA Crystals	Possible but rate expected to be lower than for aerated BAC			
Aerated Boric Acid Corrosion (BAC) Concentrated Boric Acid Solution with Oxygen	Not possible due to low oxygen deep in crevice	Unlikely	Possibly	Up to 1-5 inches per year

Degradation Progression

Degradation Progression
Leak Rate is Main Controlling Parameter

PRELIMINARY

Increasing Leak Rate →

Nozzle/Weld Condition	leak path to annulus	leak path to annulus	leak path to annulus	likely crack in nozzle wall reaches above top of weld on OD and ID	crack in nozzle wall reaches relatively high above top of weld on OD and ID
Annulus Condition	hypothetical clogged annulus	possibly some opening up of annulus	likely some opening up of annulus	may require some opening up of annulus	likely requires some opening up of annulus
Leak Rate	hypothetical zero leak rate	< on the order of 0.001 gpm	roughly between 0.001 and 0.01 gpm	roughly between 0.01 and 0.1 gpm	> on the order of 0.1 gpm
Liquid Velocity Exiting Crack	0 ft/s	less than on the order of 0.01 ft/s	roughly between 0.01 and 0.1 ft/s	roughly between 0.1 and 1 ft/s	> on the order of 1 ft/s
Local Temperature	600°F	Close to 600°F	at least roughly 500°F	roughly between 212 and 500°F	Close to 212°F
Liquid Location	fills annulus up to hypothetical blockage	all liquid vaporizes close to bottom of annulus	liquid film unlikely to exist high in annulus	liquid film may cover much of annulus walls	liquid film covers local top surface of head
Possible Significant Mechanisms	• none	• possibly very minor galvanic	• possibly some galvanic corrosion • erosion until annulus opens slightly	• likely some galvanic corrosion • minor erosion and FAC, • possibly aerated BAC if annulus is opened enough	• aerated BAC on top of head, • possibly molten BAC, galvanic corrosion, erosion, or FAC
Pounds of Boric Acid Deposits Released in 2 years	at least small amount extruded	< on the order of 7 lbs	roughly 7 to 70 lbs	roughly 70 to 700 lbs	> on the order of 700 lbs
		all or most other leaking CRDM nozzles	EPRI & CE Annulus Tests		Davis-Besse Nozzle #3

Degradation Progression

- Condition 1a. If—contrary to plant experience—a leak path crack forms in the absence of leakage to the top surface of the head
  - There will be low oxygen, zero velocity, and no vaporization-driven concentration mechanism, so material loss rates will be small
- Condition 1b. For tight nozzle cracks that allow a leak path
  - The leak rate will be limited and the annulus downstream of the crack will boil dry within a short distance
  - Erosion and FAC will not be active due to very low liquid velocities
  - Small amounts of boric acid or boric oxide crystals will accumulate on the top head surface

Degradation Progression *(continued)*

- Condition 2. As the crack widens and the minimum leak path flow area increases
  - Flashing-induced erosion or FAC may initiate the material loss process
  - Galvanic corrosion may be important if cooling is sufficient to allow liquid to exist over a significant height in the annulus
  - These mechanisms could be expected to produce greater relative material loss deep in the annulus, consistent with Davis-Besse Nozzle #2 and the EPRI BAC leaking annulus tests
- Condition 3. As the leak rate increases and the wastage area grows from a small cavity to a large, open cavity
  - Aerated boric acid corrosion (up to 1-5 inches per year) may occur

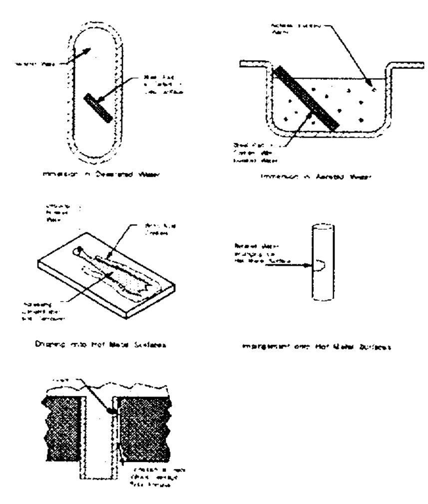
Degradation Progression *(continued)*

- The geometry of the Davis-Besse Nozzle #3 cavity may indicate that aerated BAC removing material from the top surface down toward the cladding replaced corrosion and/or erosion deep down in the annulus as the dominant degradation mode
  - The slope of the walls of the cavity change with distance from the top head surface
  - Heat transfer calculations show considerable local cooling of the head for the range of leak rates believed to apply to this nozzle, indicating an aerated, concentrated liquid boric acid solution film on the top head surface adjacent to this nozzle
  - Laboratory tests and plant experience indicate relatively high corrosion rates for low alloy steel exposed to aerated, concentrated liquid boric acid solution in comparison to other material loss mechanisms
  - Gravity-driven flow of this liquid film would tend to produce the observed oblong shape of the Nozzle #3 cavity

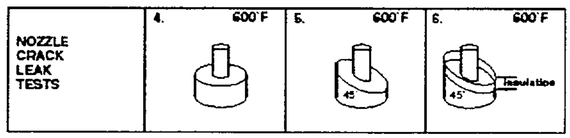
Boric Acid Corrosion Tests Simulating Nozzle Leakage

BAC Tests Simulating Nozzle Leakage *Overview*

- An extensive set of experimental data has been compiled and reported in the EPRI *Boric Acid Corrosion Guidebook, Revision 1*
  - Tests by several organizations prior to 1995
  - Tests of a range of conditions
    - Deaerated water
    - Aerated water
    - Dripping
    - Impingement
    - Leakage into annulus
  - Tests performed by EPRI at Southwest Research Institute in 1996/97
- Results of additional tests performed by CEA in France have been made available to EPRI

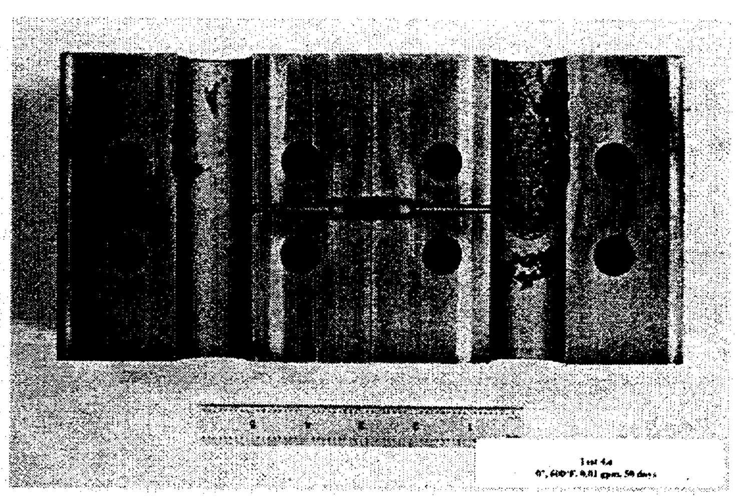


BAC Tests Simulating Nozzle Leakage
EPRI Annulus Test Matrix



Test Number	Temperature (F)	Flow Rate (gpm)
4a	600	0.01
4b	600	0.10
5a	600	0.01
5b	600	0.10
6a	600	0.01
6b	600	0.10

BAC Tests Simulating Nozzle Leakage
Typical Sectioned EPRI Test Specimen



BAC Tests Simulating Nozzle Leakage
Test Conclusions

- The maximum corrosion rates in both the EPRI and CE tests were about 2.0 – 2.5 in/yr
- The maximum corrosion rates occurred at leak rates of about 0.01 gpm with decreasing corrosion rate as leak rate was increased above 0.01 gpm
  - One test by CE at a low leak rate (0.002 gpm) showed a low corrosion rate
- While the tests may not represent the initial conditions of a very tight fit, they are considered to represent anticipated conditions once the annulus opens up to about 0.005"
- While the corrosion depth can be greater below the exposed surface than at the surface, the tests showed relatively large amounts of boric acid deposits for the range of flow rates tested

Inspection Plan

PWR Reactor Pressure Vessel Head Penetrations

Michael Lashley, South Texas Project

Status of Inspection Plan

- Inspection Plan and technical bases were presented to NRC staff on May 22
  - Technical Bases documents will be provided to NRC in June 2002.
- Comments received in following areas
  - Plan should address inspections for both wastage and nozzle ejection issues
  - Timeframe for wastage development
  - Leakage past tight interferences
  - Policy issue of detecting degradation through leakage
  - Address replacement head

Purpose

- Provide guidance and the basis for a long-term management program for RPV Head penetrations.
- Preserve structural integrity thereby ensuring safe operation.
  - GL 88-05 program remains the primary defense against boric acid wastage.
  - Inspection frequencies have been conservatively established relative to the structural integrity of the RPV Head.
- Provide a graduated approach to inspections to allow early detection of leakage or through-wall cracking prior to challenging structural integrity or significant wastage.
  - Structural integrity is defined as maintaining an acceptably low probability of developing cracking that could lead to nozzle ejection.

Scope

- Applies to the pressure boundary of the RPV head penetrations fabricated from Alloy 600 with Alloy 82/182 weld material.
- Does NOT apply to RPV head replacements and nozzle repairs with Alloy 690 and Alloy 52/152
- Assumes that a GL 88-05 walk down of the plant is effectively performed each refueling outage.

Effective Degradation Years - EDY

- Based on years of operation, normalized to 600F (as of 2/28/01)
- Effective Degradation Years (EDY) may be a more appropriate way to rank for wastage potential
  - Leaking crack as important as large circ flaw
  - Independent of ONS 3
- Although similar to old way, rank for some units changes
  - Old rank - combination of head temperature, operating time to date, and time left to ONS3 equivalence.
  - EDY rank - just time and temperature at current (2/28/01) time

Risk Informed Basis

Probabilistic fracture mechanic (PFM) analyses using a Monte-Carlo simulation algorithm

- Included experience-based time to leakage correlations
  - used a Weibull model of plant inspections to date,
  - fracture mechanics analyses of various nozzle configurations containing axial and circumferential cracks, and
  - MRP developed crack growth rate data for Alloy 600.
- Performed to determine the probability of leakage and failure versus time for a set of input parameters:
  - head operating temperature,
  - benchmarked against experience to date
- Sensitivity studies were performed for various:
  - inspection types (visual or NDE) and
  - inspection intervals.

Risk Based Susceptibility

- Moderate susceptibility boundary:
  - The number of EDYs at which a plant reaches
    - probability of one leaking nozzle = 20% (approximately equal to the probability of net section collapse (NSC i.e. nozzle ejection) = 1×10^{-4}
- High susceptibility boundary:
  - The number of EDYs at which a plant reaches:
    - probability of nozzle ejection = 1×10^{-3} (approximately equal to the probability of one leaking nozzle = 75%)
      - consistent with NRC RG 1.174 guidance for change in Core Damage Frequency.

Plant Categories

- Low Susceptibility:
  - less than 10 Effective Degradation Years, EDY (defined as Effective Full Power Years @ 600F), without a leak or identified crack
- Moderate Susceptibility:
  - greater than or equal to 10 EDY and less than 18 EDY without a leak or identified through-wall crack
- High Susceptibility:
  - greater than or equal to 18 EDY or units that have identified leaks or through-wall cracks.

CRDM/CEDM J-Groove Weld Inspection Bases

- Circumferential cracks in the J-groove weld do not pose a significant risk of nozzle ejection.
- Lack-of-fusion: extent to still maintain structural integrity is similar to the acceptable extent of through-wall circumferential cracking (i.e. >75% of the circumference).

CRDM/CEDM Head Penetration Flaw Acceptance Criteria

- Visual evaluation criteria
  - EPRI Technical Report 1006899, Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of the RPV Head: Revision 1,March 2002.

- Non-visual evaluation criteria
  - MRP and ASME Section XI Code are working to develop final criteria, and until those criteria are issued, NRC-proposed criteria may be used.

Inspection Schedule – Low Susceptibility

For low susceptibility plants (< 10 EDY):

- Perform a Bare Metal Visual (BMV) examination of 100% of the CRDM/CEDM penetrations once per 10 years, beginning no later than the third ISI interval.

- Or, perform NDE (i.e., non-visual examination) of 100% of the CRDM/CEDM penetrations and associated J-groove welds once per 10 years, beginning no later than the third ISI interval.

Inspection Schedule – Moderate Susceptibilty

For moderate susceptibility plants (10 EDY≤X< 18 EDY)*:*

- Perform a BMV examination of 100% of the CRDM/CEDM penetrations at the 1st RFO upon entering this category and once every 2 EDY not to exceed 5 EFPYs.

- Or, perform NDE (i.e., non-visual examination) of 100% of the CRDM/CEDM penetrations and associated J-groove welds at the 1st RFO upon entering this category and once every 4 EDY not to exceed 10 EFPYs.

Inspection Schedule – High Susceptibility

For high susceptibility plants (≥18 EDY)*:*

- Perform a BMV examination of 100% of the CRDM/CEDM penetrations at every RFO upon entering this category, **<u>AND</u>**
- Perform NDE (i.e., non-visual examination) of 100% of the CRDM/CEDM penetrations and associated J-groove welds within 4 EDY upon entering this category or issuance of this Plan, whichever is later
  - Exceptions to 100% NDE for undue hardship.

OR

- Perform NDE (i.e., non-visual examination) of 100 % of the CRDM/CEDM penetrations and associated J-groove welds at the 1st RFO upon entering this category and once every 4 EDY not to exceed 6 EFPYs.

Inspection Plan

- **Plants with leak(s) or through wall cracks identified:**
  - *Discovery Inspection*
    - Perform a non-visual examination of the CRDM/CEDM penetrations and associated J-groove welds to characterize the crack or leak identified.
    - Indications are evaluated or repaired in accordance with flaw evaluation guidelines.

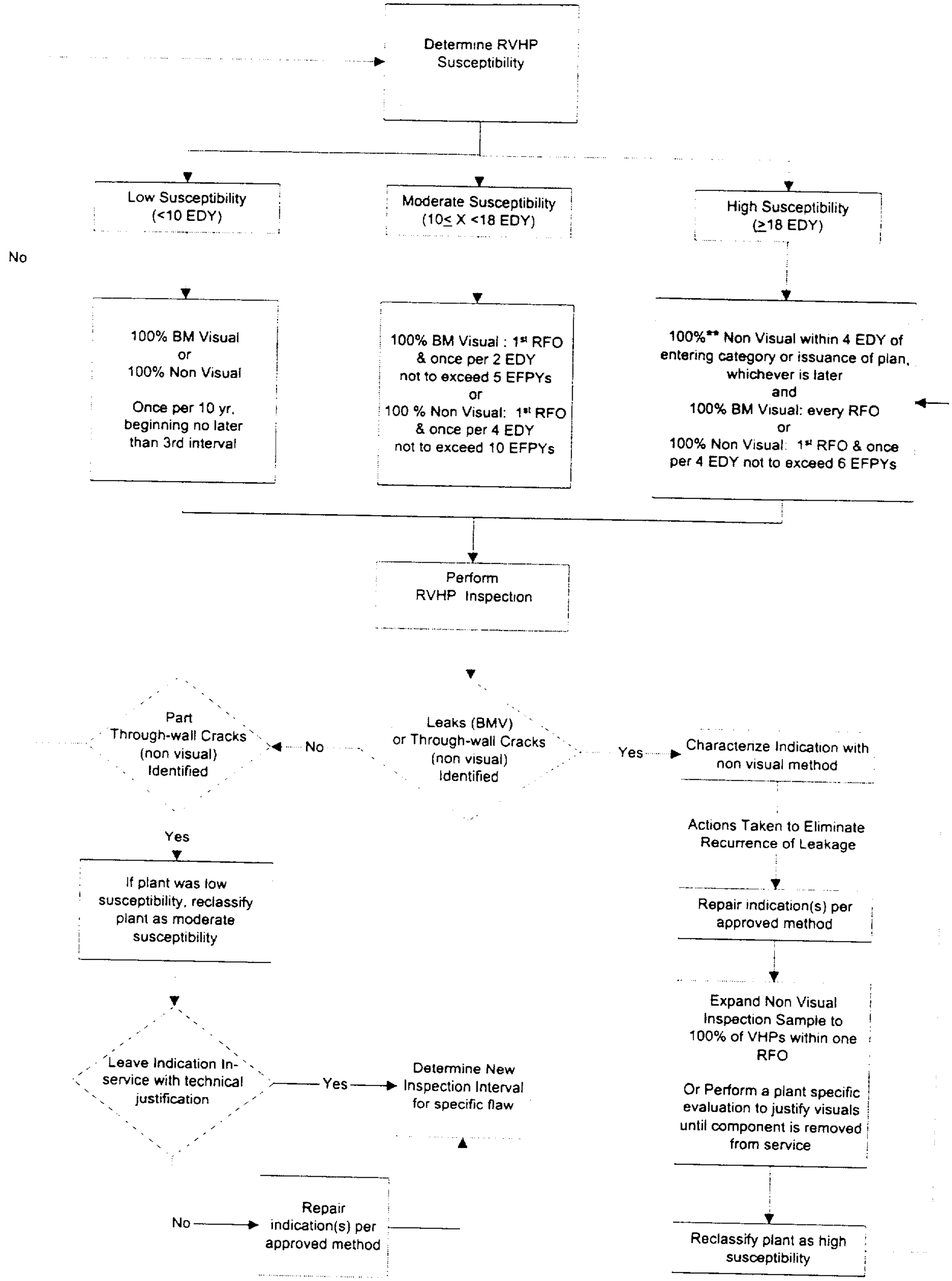
Plants with leak(s) or through wall cracks

Expansion of Inspection (to be implemented no later than next RFO)

- Perform NDE (i.e., non-visual examination) of 100% of the CRDM/CEDM penetrations and associated J-groove welds.
  - Indications are evaluated or repaired in accordance with flaw evaluation guidelines (Reference 4).
- Or, perform an evaluation to justify continued visual examination until the RVH component is removed from service.
- Or, perform NDE at a frequency to be determined such that the 3x safety margin of a hypothetical circumferential crack growing above the weld is not exceeded prior to the next inspection.

Figure 1

PWR RPV Head Penetrations Inspection Flowchart



** 100% of the CRDM/CEDM penetrations and associated J-groove welds or portions thereof that can be examined without incurring undue hardship

NRC's Vessel Head Penetration Cracking and Vessel Head Degradation Presentations

Presenters

Allen Hiser, 301-415-1034
Andrea Lee, 301-415-2735
Mark Kirk,301-415-6015
Jack Grobe, 630-829-9700
Ed Hackett, 301-415-5660

Advisory Committee on Reactor Safeguards
Materials & Metallurgy and Plant Operations Subcommittees
Rockville, MD
June 5, 2002

STATUS OF NRC BULLETIN 2001-01 REVIEWS

"CIRCUMFERENTIAL CRACKING OF VHP NOZZLES"

Allen Hiser

Presentation to ACRS Subcommittee

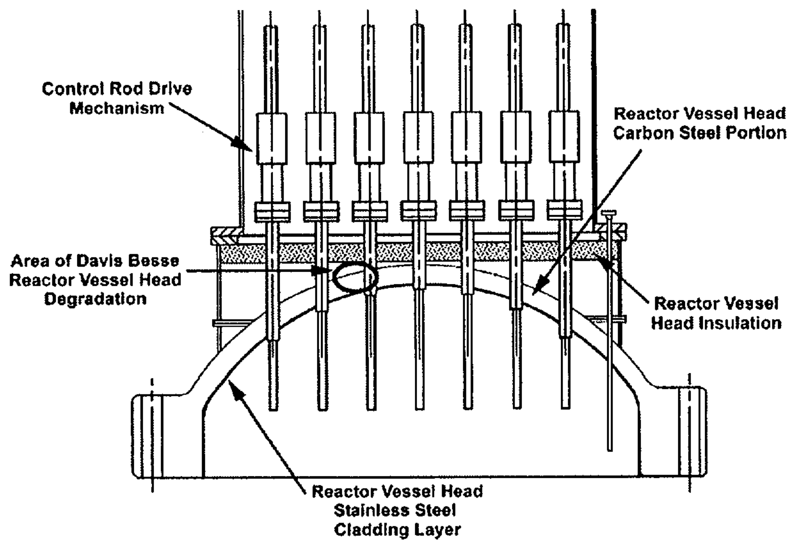
June 5, 2002

STATUS

- No additional inspection findings since ACRS meeting in April 2002
- MRP presentation of proposed inspection plan in late May 2002
- NRC staff is considering generic communication to address interim guidance for vessel head penetration nozzle and vessel head inspections
- Interactions with the industry continue to provide technical basis for NRC staff development of long-term inspection requirements, including activities within the appropriate ASME Code groups

STATUS OF BULLETIN 2002-01, "REACTOR PRESSURE VESSEL HEAD DEGRADATION AND REACTOR COOLANT PRESSURE BOUNDARY INTEGRITY"

Reactor Vessel Head Degradation Location



Briefing of the Advisory Committee on Reactor Safeguards

June 5, 2002

Andrea D. Lee
301-415-2735

BULLETIN 2002-01 - REQUESTED INFORMATION

Issued March 18, 2002 to assess all PWR plants

Within 15-days

> **Summary of the RPV head inspection and maintenance programs**
>
> **Evaluation of the ability of inspection and maintenance programs to identify degradation**
>
> **Description of any conditions that could have lead to degradation and the corrective actions taken to address such conditions**
>
> **Plans, basis, and schedule for future inspections of the RPV head and VHP nozzles**
>
> **Basis for continued operation until the inspections can be performed**

Within 30-days after inspection: Results

Within 60-days: Boric acid corrosion prevention program for ensuring integrity of the rest of the reactor coolant pressure boundary

BULLETIN 2002-0 REVIEW STATUS

15-day responses from all 69 PWRs except for Davis-Besse

Staff has not identified any plants with conditions similar to those that lead to the degradation at Davis-Besse

Priority categorization for contacting plants

High priority (7 plants) : Beaver Valley 1, Callaway, Fort Calhoun, Indian Point 2, Indian Point 3, Salem 2, and St. Lucie 1

Medium priority (4 plants): Calvert Cliffs 1, San Onofre 2, Sequoyah 1, and Sequoyah 2

Low priority (8 plants): Catawba 1, Catawba 2, Farley 2, Harris, Millstone 3, Point Beach 1, Point Beach 2, and San Onofre 3

No concern (49 plants)

For most licensees, telephone conferences and supplemental responses have resolved questions on material condition of RPV head

60-day responses from 68 of 69 PWRs

60 day responses were due May 18

Staff review started (approximately 20%) but not complete

NRC Assessment of Margin Available at Davis Besse

Mark Kirk, Wally Norris, Nilesh Chokshi

RES/DET/MEB

Paul Williams, Richard Bass

Oak Ridge National Laboratory

Gery Wilkowski , Dave Rudland

Engineering Mechanics Corporation of Columbus

ACRS Briefing: Materials and Metallurgy & Plant Operations Subcommittees
USNRC Headquarters • Rockville, MD • 5th May 2002

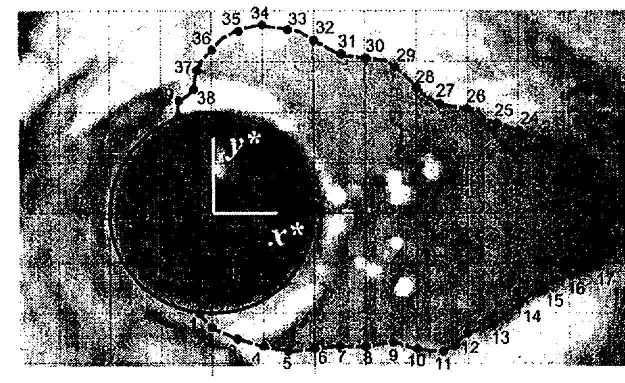
VG 1

Overview of Presentation

- **Deterministic assessment of margins**
  - **Scope of investigation**
  - **Analytical tools**
  - **Findings to date**
- **Next steps**
  - **Further deterministic analysis**
  - **Probabilistic analysis**

RES Assessment of Davis-Besse "Margins"

[redacted]



FOOTPRINT OF WASTAGE AREA

- **Margin left in condition that existed at March '02 shutdown**
- **How much**
  - **More pressure, or**
  - **More wastage**

  could have been tolerated without failure?

Done

Underway

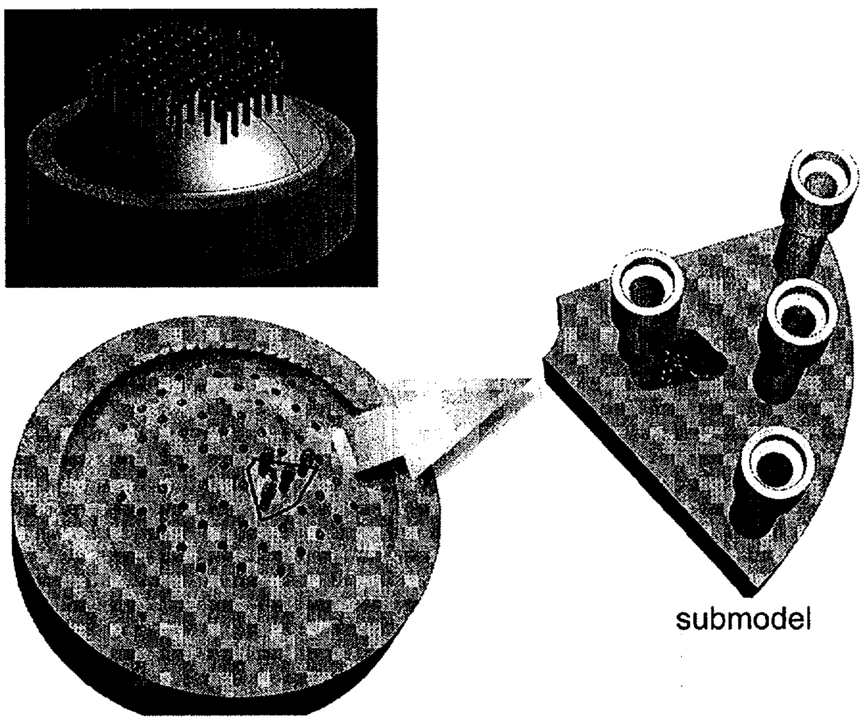
- **Assessment of repair options**

Likely not necessary in view of current licensee plans to procure and modify Midland head

VG 3

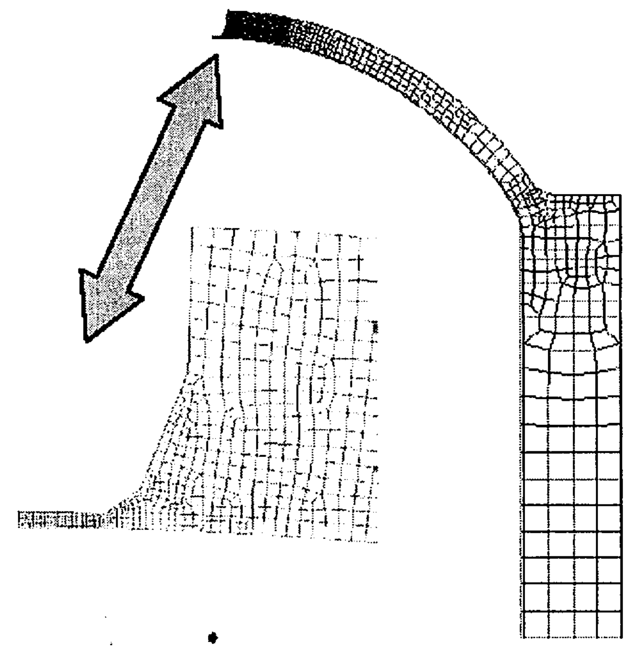
Analytical Tools

[redacted]



- **Most realistic representation of the geometry of both the wastage area and the overall head design**

[redacted]



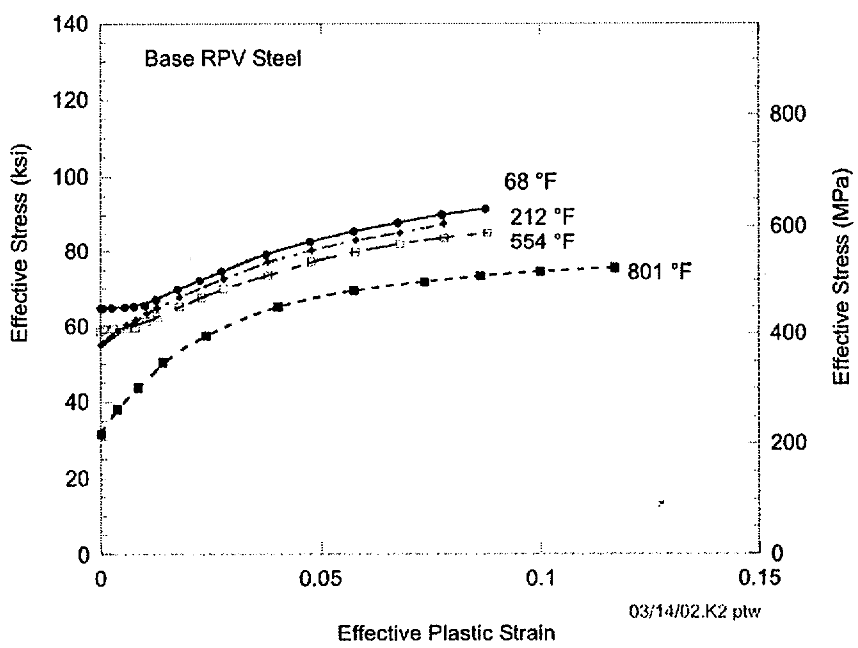
- **Wastage modeled as pit at top of head**
- **More refined cladding model (than possible in 3D)**
- **Allowed easier investigation of additional wastage area needed to produce failure**

Details of Analyses

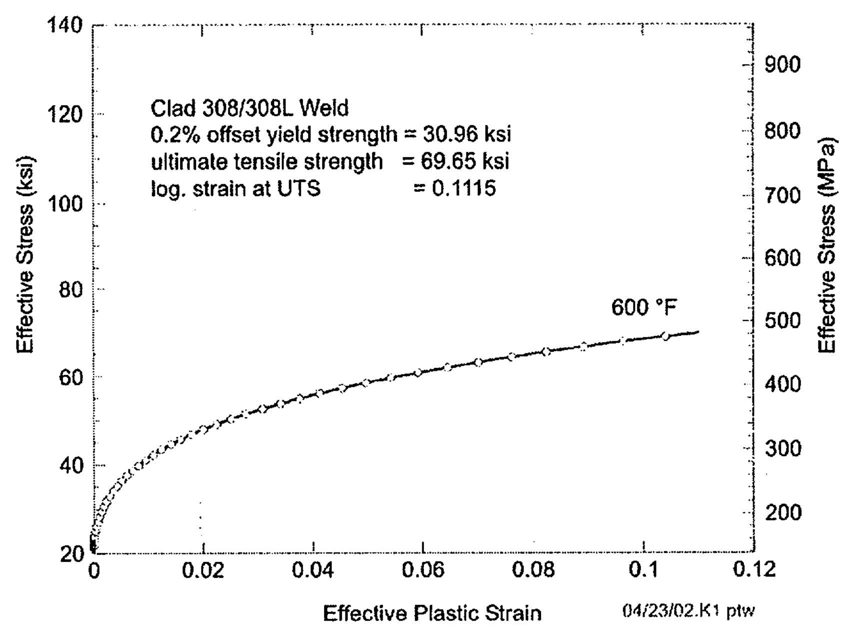
	3D FE Model (*ORNL*)	Axi-Symmetric FE Model (*EMC²*)
Loading	P = Design (2165 psi) or higher T = Operating (600°F), no gradients	
Material Properties	On next page.	
Geometry	■ All penetrations modeled ■ Straight walled 3D cavity ■ Geometry digitized from early photo.	Axial pit at apex of head
Failure Criteria	■ Failure occurs when the average through-thickness equivalent plastic strain in the cladding exceeds 5.5% ■ 5.5% corresponds to the strain at the beginning of plastic instability. Derived from • 11.15% strain in a uni-axial tension test • Assumption that "failure" occurs at same stress level under uni-axial and bi-axial loading.	

Material Stress-Strain Properties

RPV Steel



Cladding

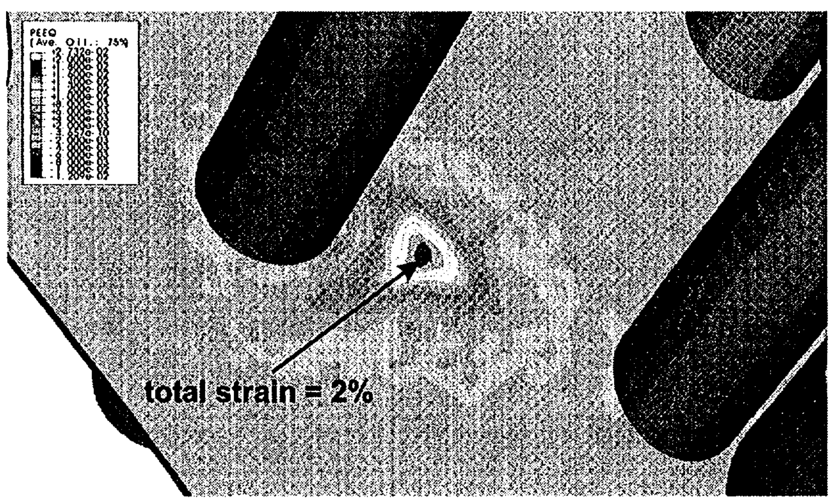


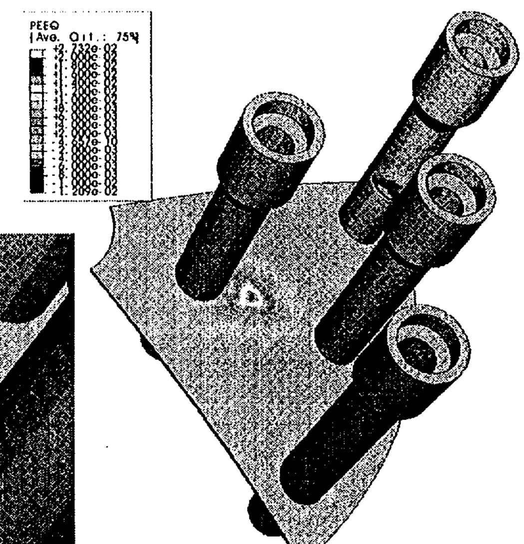
Summary of Findings

→ As-Found Condition ←

- **At operating pressure (2165 psi) the 3D FE model predicts 2% plastic strain in the cladding**
  - **No failure predicted relative to assumed failure criteria**

Pressure = 2.165 ksi





Summary of Findings

→ Margin on Overpressure ←

- **Depending upon**
  - **The particular failure strain (5.5% *vs.* 11%)**
  - **The strain value (average, minimum, etc.)**
  - **Cladding thickness (design, average measured, minimum measured)**

used in the analysis, different margins on overpressure result:

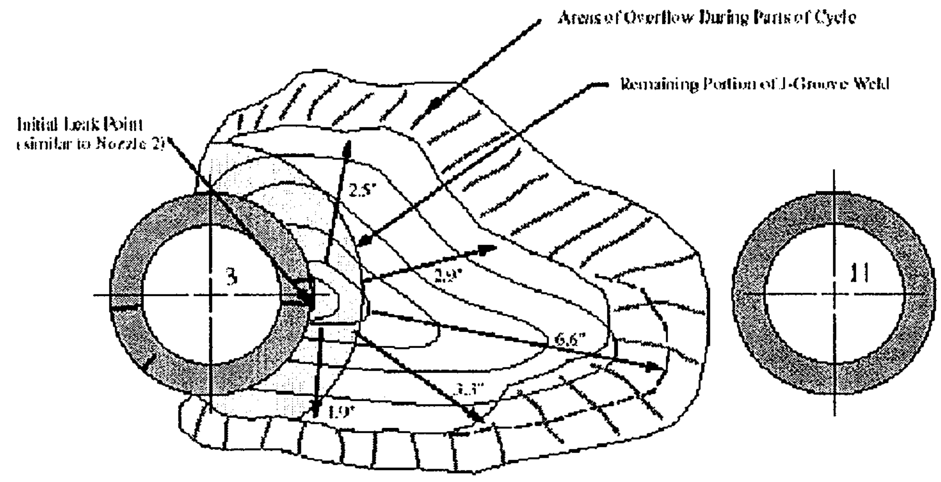
- **SIA (Industry) 3D Analysis: $P_{fail} / P_{oper} = 2.1 - 2.6$**
- **ORNL (NRC) 3D Analysis: $P_{fail} / P_{oper} = 1.4 - 2.0$**
- **EMC² (NRC) 2D Analysis: $P_{fail} / P_{oper} = 1.1 - 1.4$**

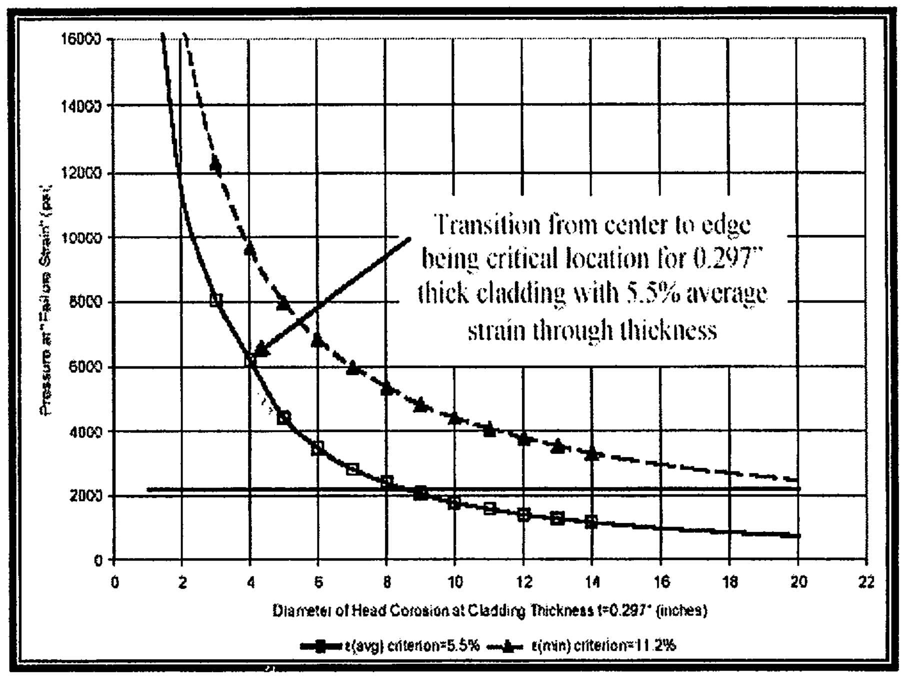
<u>Note</u>: Only the most pessimistic overpressure margins do not exceed the SRV set-point of 110% $\cdot$ P_{oper}

Summary of Findings

→ *Additional Cavity Growth Needed to Fail* ←

- **About 1.9-in. more wastage needed (along maximum growth axis) to cause failure at the operating pressure, assuming**
  - **5.5% failure strain (average through thickness**
  - **Average thickness cladding**
  - **Appropriateness of axi-symmetric model**





VG 9

Next Steps

- **Better definition of failure criteria**
  - **Calibration relative to appropriate data, if data is available**
  - **Determination of significance of different failure criteria (for probabilistic analysis up to 2500 psi)**
- **Cavity growth rate**
  - **Growth rate data**
  - **Growth models**
- **Probabilistic analysis**

Next Steps (details)

- **Re-analyses using ORNL "best-estimate" 3-D FE model of existing cavity up to 2500 psi to quantify failure probabilities**
- **Further evaluation of clad failure criteria by analyzing measured data obtained from (6-in. dia. x 0.25 in. thick.) SS burst disks**
- **3-D FE analyses of cavity growth scenarios to refine estimates of critical wastage area at P_{oper}**

Ongoing NRC Regulatory Activities at Davis-Besse

Davis-Besse Nuclear Power Station

Ongoing NRC Regulatory Activities at Davis-Besse

Background

- Significant Reactor Pressure Vessel Head degradation discovered March 6, 2002

- NRC chartered an Augmented Inspection Team (AIT) March 12, 2002

- NRC issued a Confirmatory Action Letter (CAL) on March 13, 2002

- NRC established the IMC 0350 Oversight Panel April 29, 2002

Ongoing NRC Regulatory Activities at Davis-Besse

Implementation of IMC 0350 at Davis-Besse

- **Reactor Vessel Head Degradation represents a significant and complex technical and regulatory issue**
- **Plant is in an extended shutdown with a regulatory hold in effect (CAL)**
- **IMC 0350 enhances the agency's focus on clearly defining and addressing plant specific issues prior to restart**
- **IMC 0350 provides focused and coordinated regulatory oversight of Davis-Besse**

Ongoing NRC Regulatory Activities at Davis-Besse

IMC 0350 Panel Goals

- Provide oversight and assessment of licensee performance during the shutdown and through restart

- Assure that restart issues are identified and resolved

- Integrate and prioritize agency resources to maximize agency effectiveness and minimize regulatory burden

- Provide a single focus to ensure consistent and effective communication with external stake holders

Ongoing NRC Regulatory Activities at Davis-Besse

IMC 0350 Panel Goals

- **Continue oversight after plant restart until plant is returned to the routine Reactor Oversight Process**

- **Create a comprehensive public record of agency decisions and actions**

Ongoing NRC Regulatory Activities at Davis-Besse

Panel Members

- Jack Grobe, Director, Division of Reactor Safety, RIII
- Bill Dean, Deputy Director, Division of Inspection Program Management, NRR
- Christine Lipa, Branch Chief, Division of Reactor Projects, RIII
- John Jacobson, Branch Chief, Division of Reactor Safety,RIII
- Tony Mendiola, Section Chief, NRR
- Doug Pickett, Project Manager, NRR
- Scott Thomas, Senior Resident Inspector, RIII
- Sonia Burgess, Senior Reactor Analyst, RIII

Ongoing NRC Regulatory Activities at Davis-Besse

Overview of the IMC 0350 Process

- Routine Reactor Oversight Process suspended
- IMC 0350 Panel provides Oversight and Assessment of Licensee Performance
- IMC 0350 Panel Process Plan details major tasks related to Oversight and Restart
- IMC 0350 Restart Checklist identifies all necessary restart items
- Periodic internal and external meetings to discuss licensee progress towards restart

Ongoing NRC Regulatory Activities at Davis-Besse

License submitted Return to Service Plan - May 21, 2002

- Reactor Head Resolution Plan
- Containment Extent of Condition Plan
- System Health Assurance Plan
- Program Technical Compliance Plan
- Management and Human Performance Excellence Plan
- Restart and Post-Restart Test Plan

Ongoing NRC Regulatory Activities at Davis-Besse

Current Inspections

- **AIT follow-up (May - June)**
- **Vessel Head Replacement (May - September)**
- **Extent of Condition - Boric Acid (May - August)**

DAVIS-BESSE REACTOR VESSEL HEAD DEGRADATION LESSONS LEARNED TASK FORCE

Briefing for
Materials and Metallurgy & Operations Subcommittees
Advisory Committee on Reactor Safeguards

Ed Hackett
Assistant Team Leader
Davis-Besse Lessons Learned Task Force

June 5, 2002

Davis-Besse Reactor Vessel Head Degradation Lessons Learned Task Force

Objective/Scope

The Task Force will conduct an independent evaluation of the NRC staff's regulatory processes related to assuring reactor vessel head integrity in order to identify and recommend areas of improvement applicable to the NRC and/or the industry. The scope of the task force effort will include the following five areas: reactor oversight process issues, regulatory process issues, research activities, applicable practices used in the international community, and the NRC's generic issue process.

Davis-Besse Reactor Vessel Head Degradation Lessons Learned Task Force

Charter

- Reactor Oversight Process Issues - evaluate the underlying causes of the Davis-Besse reactor vessel head degradation, and assess whether enhancements to the NRC's reactor oversight process are warranted.

- Regulatory Process Issues - evaluate regulations, the licensing review process, and other NRC regulatory processes such as generic communications to determine whether enhancements are warranted.

- Research Activities - determine whether there are any issues associated with the NRC process of using reactor operating experience and the results of various research programs, including research performed by NRC, requiring improvement.

- International Practices - identify and evaluate foreign regulatory practices related to reactor vessel head degradation to possibly enhance NRC programs and practices.

- Generic Issue Process - evaluate the NRC process for identifying and responding to emerging technical issues, including the implementation of short-term and long-term follow-on efforts by the licensee and NRC.

Davis-Besse Reactor Vessel Head Degradation Lessons Learned Task Force

Team Composition

○ Art Howell (Region IV)	Team Leader
○ Ed Hackett (RES)	Assistant Team Leader
○ Elaine Raphael (NRR)	Administrative Assistant
○ Russell Bywater (Region IV)	
○ Patrick Castleman (NMSS)	
○ Joseph Donoghue (NRR)	
○ Robert Haag (Region II)	
○ Thomas Koshy (NRR)	

- Ron Lloyd (RES)
- Observer - State of Ohio Representative

Davis-Besse Reactor Vessel Head Degradation Lessons Learned Task Force

Schedule

(1) Preparation Phase (6/3 - 6/23)

(2) Review Phase (6/24 - 9/3)

The Task Force expects to complete activities in September, 2002 with the issuance of a comprehensive report documenting team review activities and presenting analyses, conclusions and recommendations.

Davis-Besse Reactor Vessel Head Degradation
Lessons Learned Task Force

Current Status

- All team members reported officially on June 3 and are physically located at NRC HQ
- Team Orientation and Initial Briefings have been completed (June 3-5)
- Preliminary Region III Office Visit scheduled for June 6
- Site Visit and Public Entrance scheduled for June 12 in Oak Harbour, OH
- Interviews with key NRC managers are in progress
- Detailed review plans are being drafted

MANAGEMENT BY LEAKAGE DETECTION

Allen Hiser

Presentation to ACRS Subcommittee

June 5, 2002

ISSUES

- What are appropriate inspection methods and frequencies as related to cracking of nozzles and/or J-groove weld?

- Technical specifications ("no pressure boundary leakage") and ASME Code (determine and correct identified leakage sources)

  - Do not appear to permit operation with "known" reactor coolant pressure boundary leakage
  - Current equipment not capable of detecting low leakage amounts from vessel head nozzle cracking

- What is the role of leakage detection for vessel head nozzles?

  - Limited to defense in depth?

INDUSTRY PROPOSED INSPECTION PLAN

- Industry proposal discussed on May 22 - to be presented later
  - Does not consider explicitly vessel head degradation experience
  - Technical basis is in progress - report is not available
  - Can rely on bare metal visual examinations for moderate susceptibility plants
  - Limited to Alloy 600 heads
  - Assumes "robust" Generic Letter 88-05 program, effectively implemented

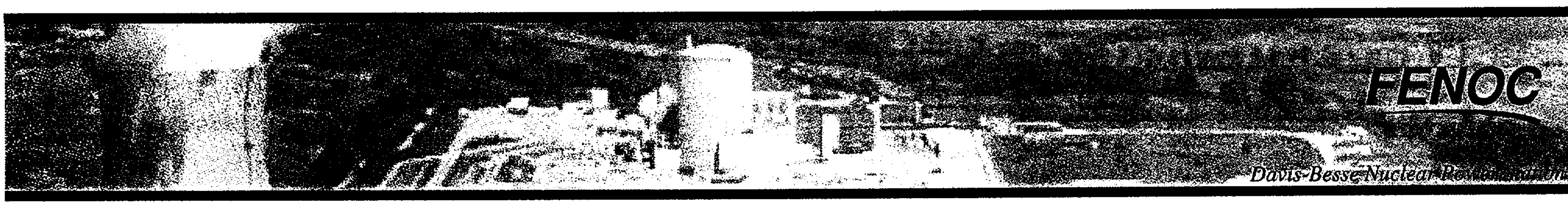
- Summary of NRC staff comments on industry proposal
  - "Relevant" visual conditions requires definition
  - Inspection methods and frequencies requires technical basis
  - Capability and recent experience with NDE should be considered and included
  - Is the plan benchmarked to the onset of unacceptable conditions (leakage?) or discovery of conditions at Oconee?
  - Appropriate application of RG 1.174?
  - Delay of scope expansion to next RFO requires technical basis

STAFF ACTIVITIES AND CONCERNS

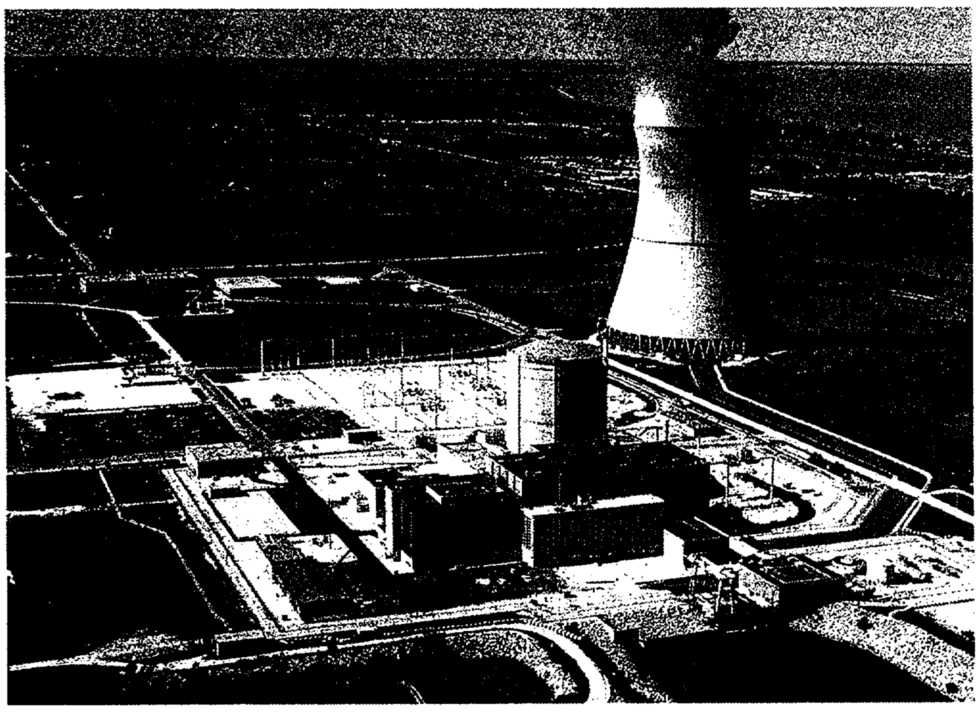
- Understanding of Davis-Besse degradation mechanisms and rates
  - Physical evidence and laboratory demonstrations
- Industry proposal needs a sufficient technical basis for inspection methods and frequencies
- Staff is considering a generic communication to bridge from current situation to "permanent" requirements
- Staff & industry are working to develop technical basis for longer-term inspection requirements
- Has the Davis-Besse experience raised the bar for vessel head nozzle cracking acceptability (axial vs. circumferential cracking)?

Advisory Committee on Reactor Safeguards Update of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Closure Head Activities

June 5, 2002



Agenda



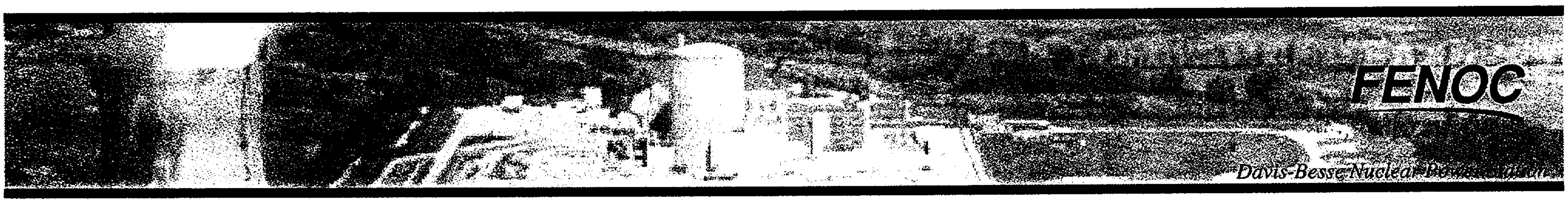
Introduction
- Jim Powers

Update of RPV Closure Head Field Activities
- Mark McLaughlin

RPV Closure Head Replacement
- Bob Schrauder

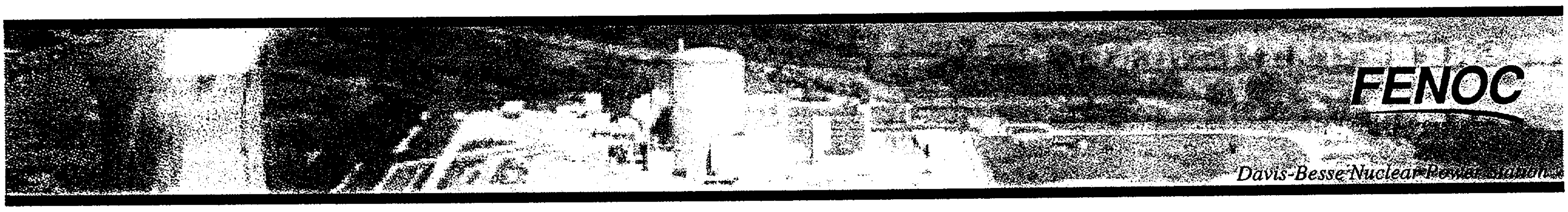
Root Cause Analysis
- Steve Loehlein

Concluding Remarks
- Jim Powers

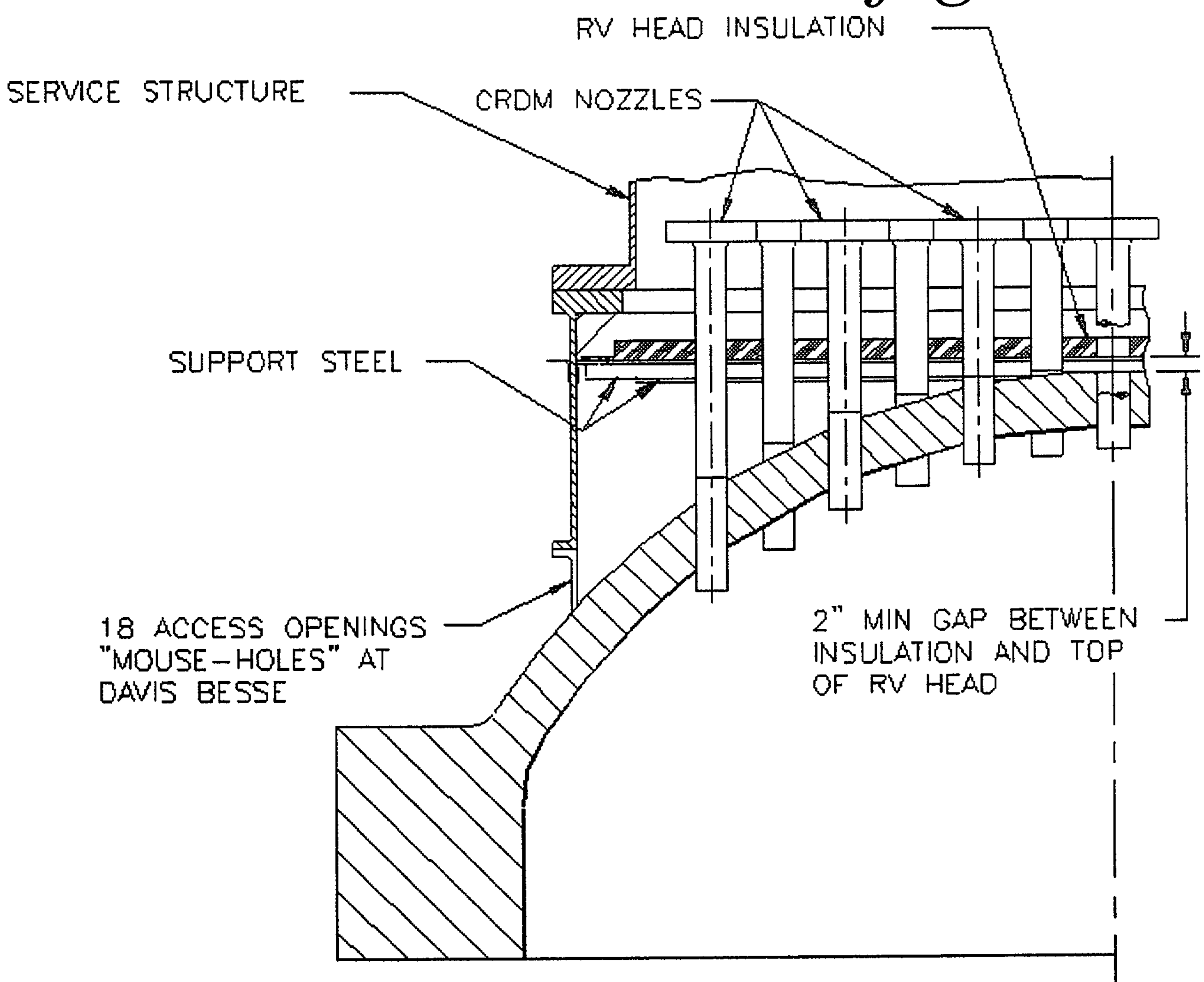


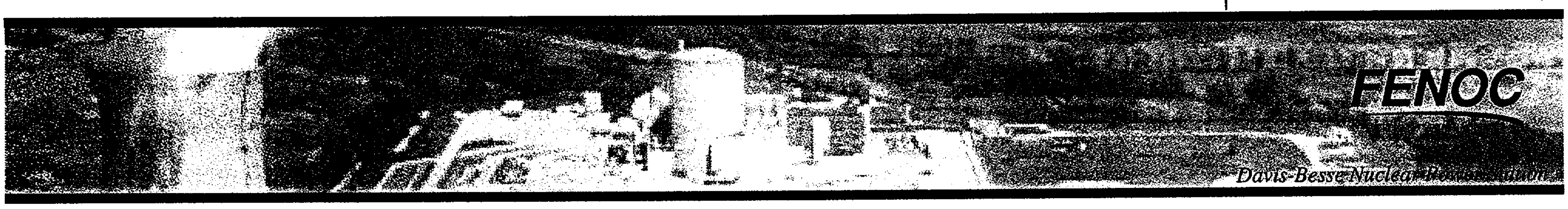
Update
of RPV Closure Head
Field Activities

Mark McLaughlin
Field Activities Team Leader



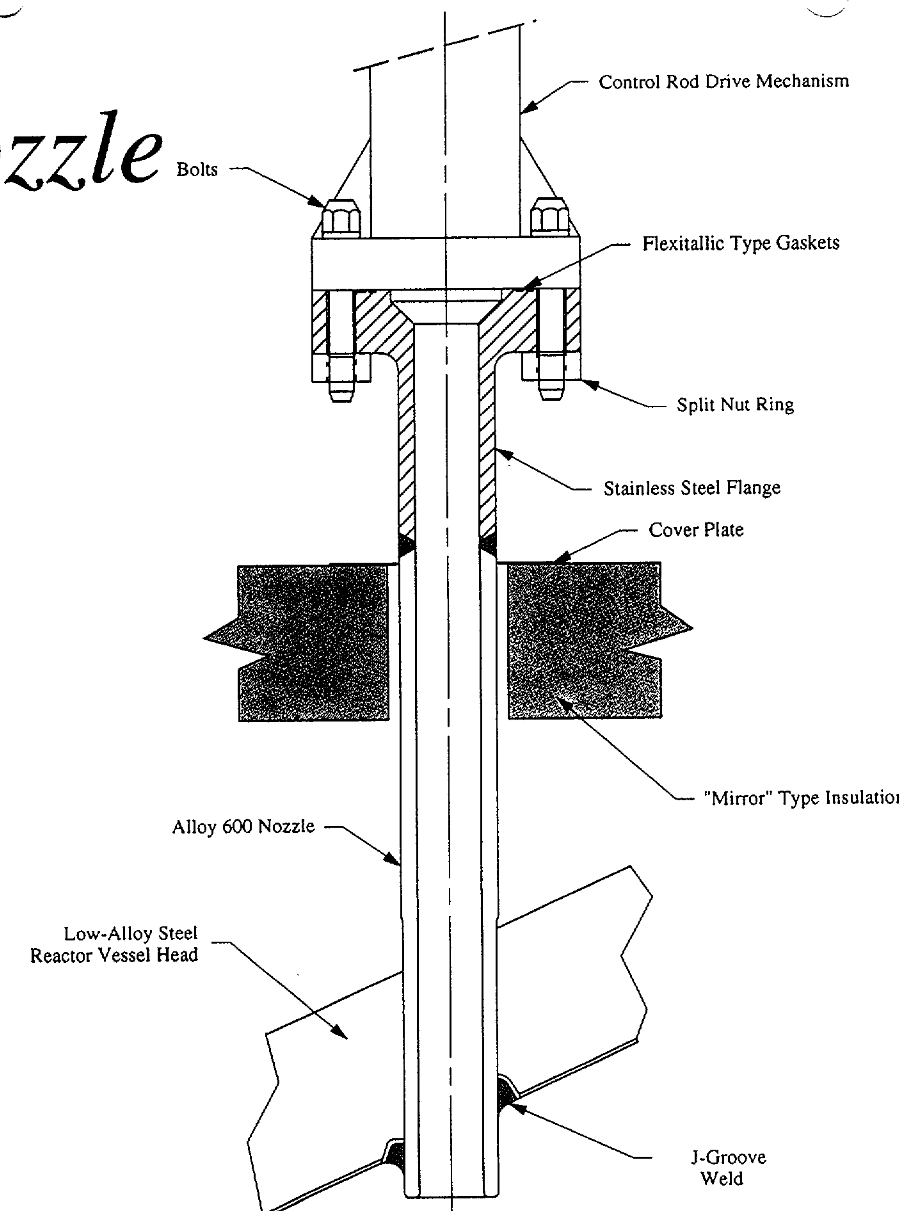
RPV Closure Head Configuration

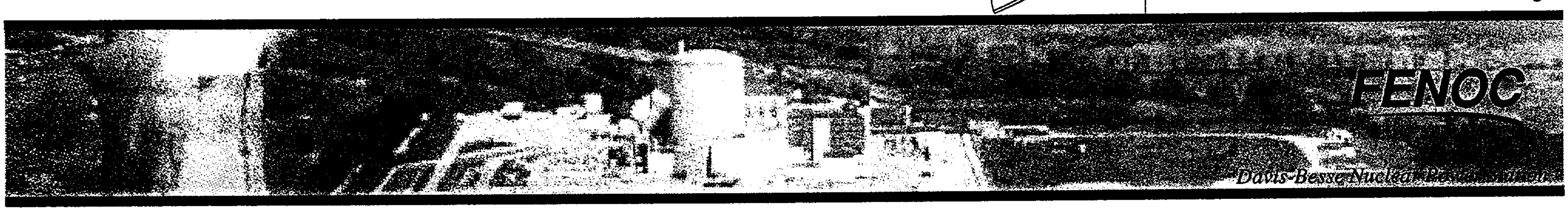




Control Rod Drive Nozzle

- Typical Control Rod Drive Nozzle (Babcock & Wilcox)

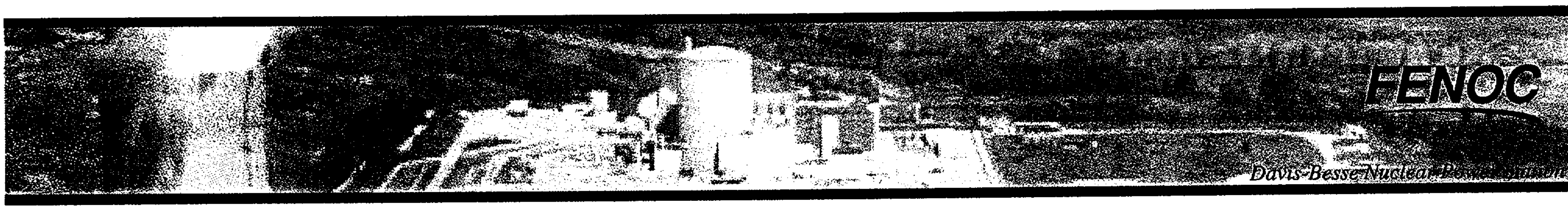




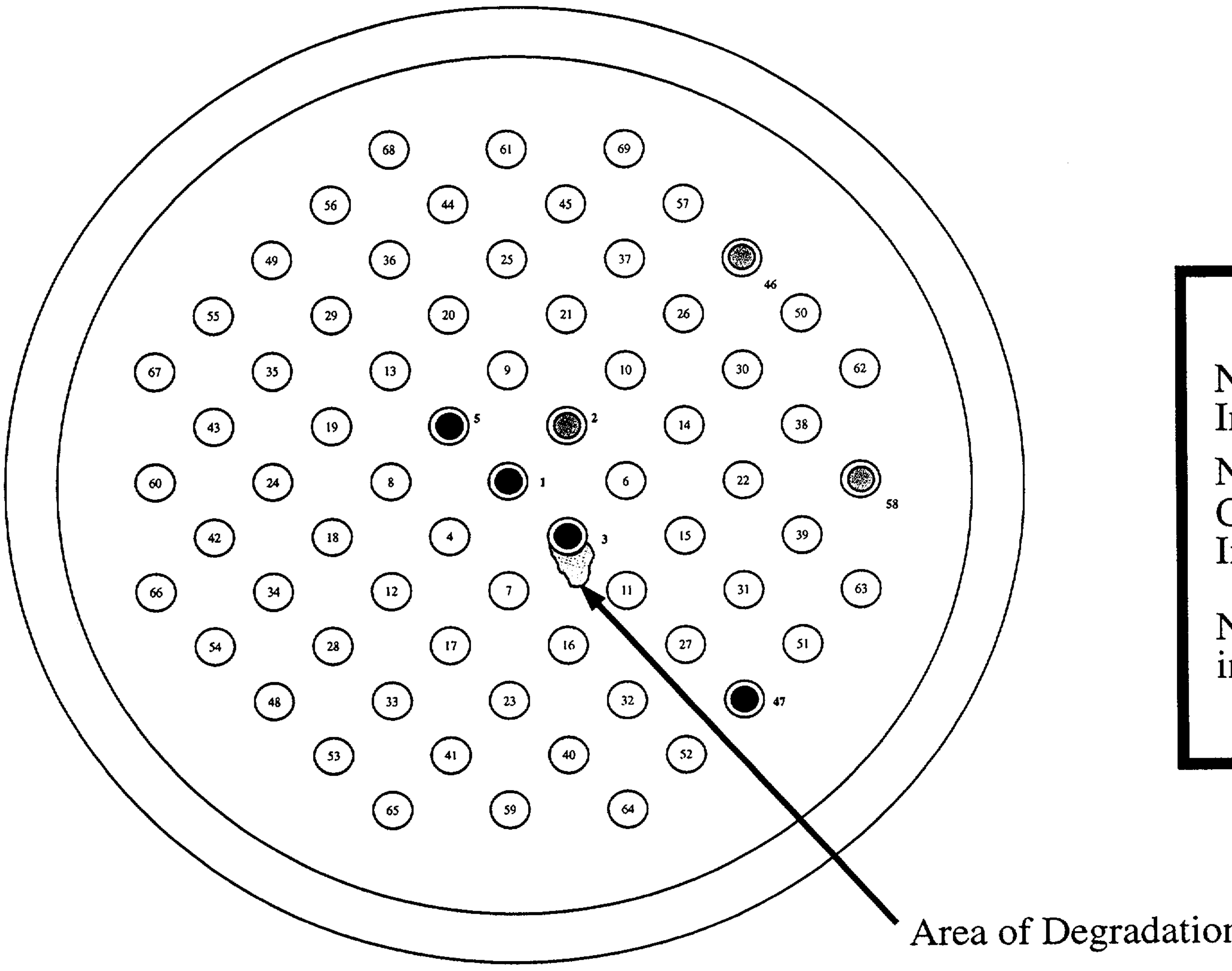
UT Examination Results

Nozzle #	Summary of Results
1*	9 Axial Flaws, 2 through-wall (TW)
2*	9 Axial Flaws, 1 Circumferential Flaw, 6 TW
3*	4 Axial Flaws, 2 TW
5*	1 Axial Flaw
46	No Flaw Indication
47	1 Axial Flaw
58	No Recordable Indications

* Heat number M3935 material



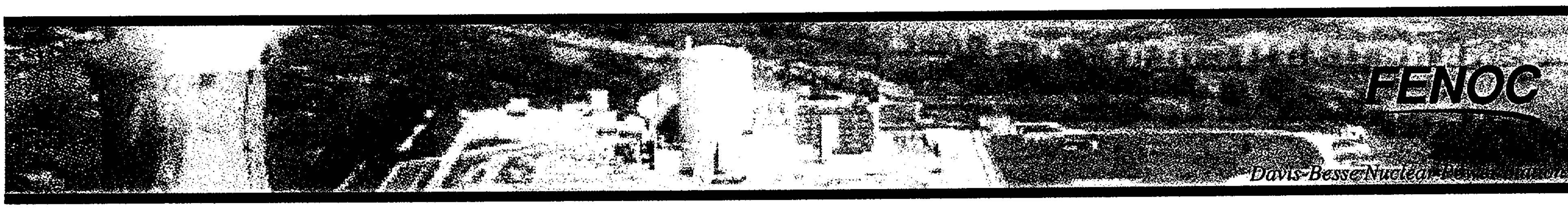
Facts of Discovery



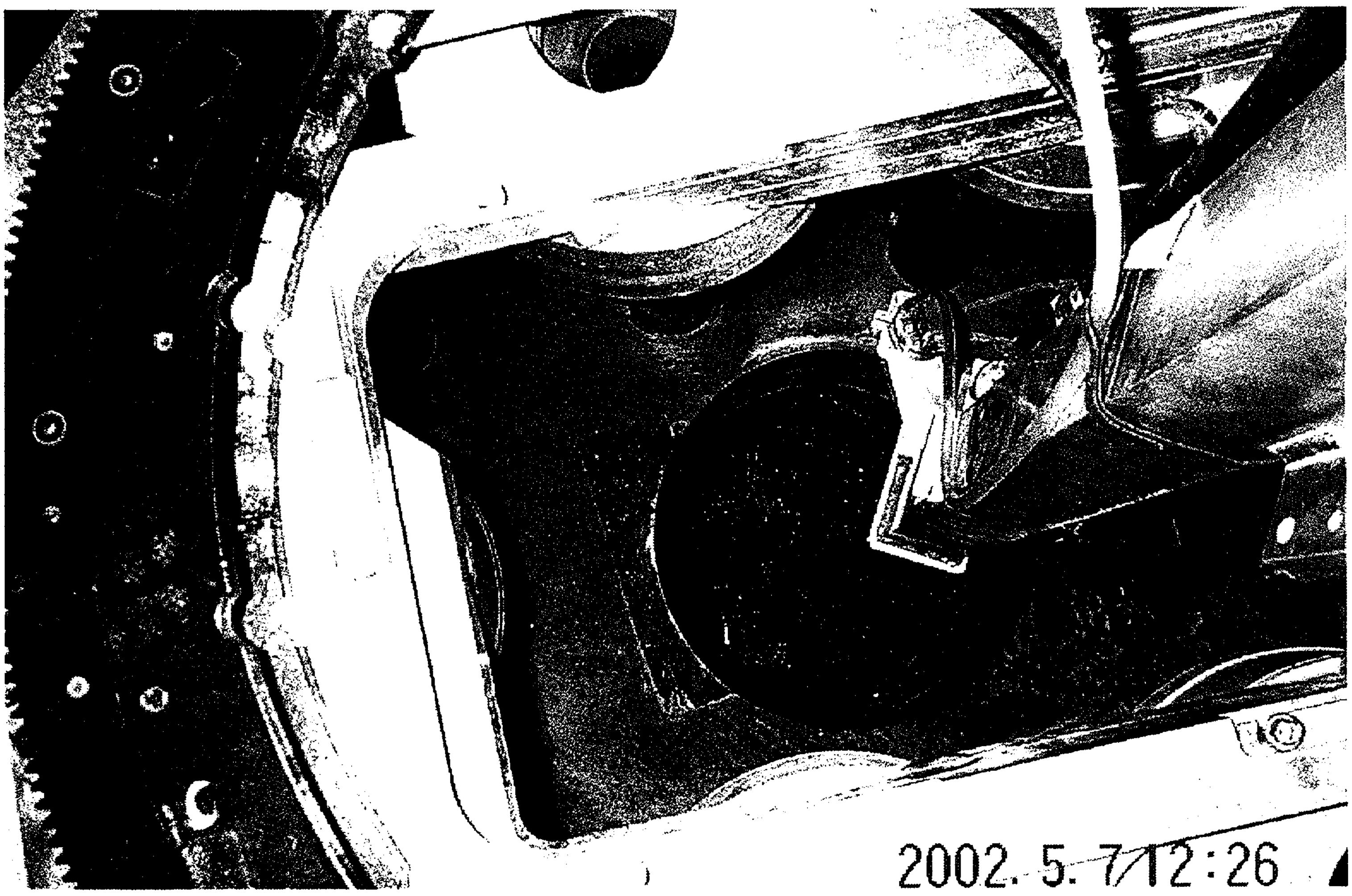
Nozzle with Axial Indication - ●

Nozzle with Axial and Circumferential Indication – ◉

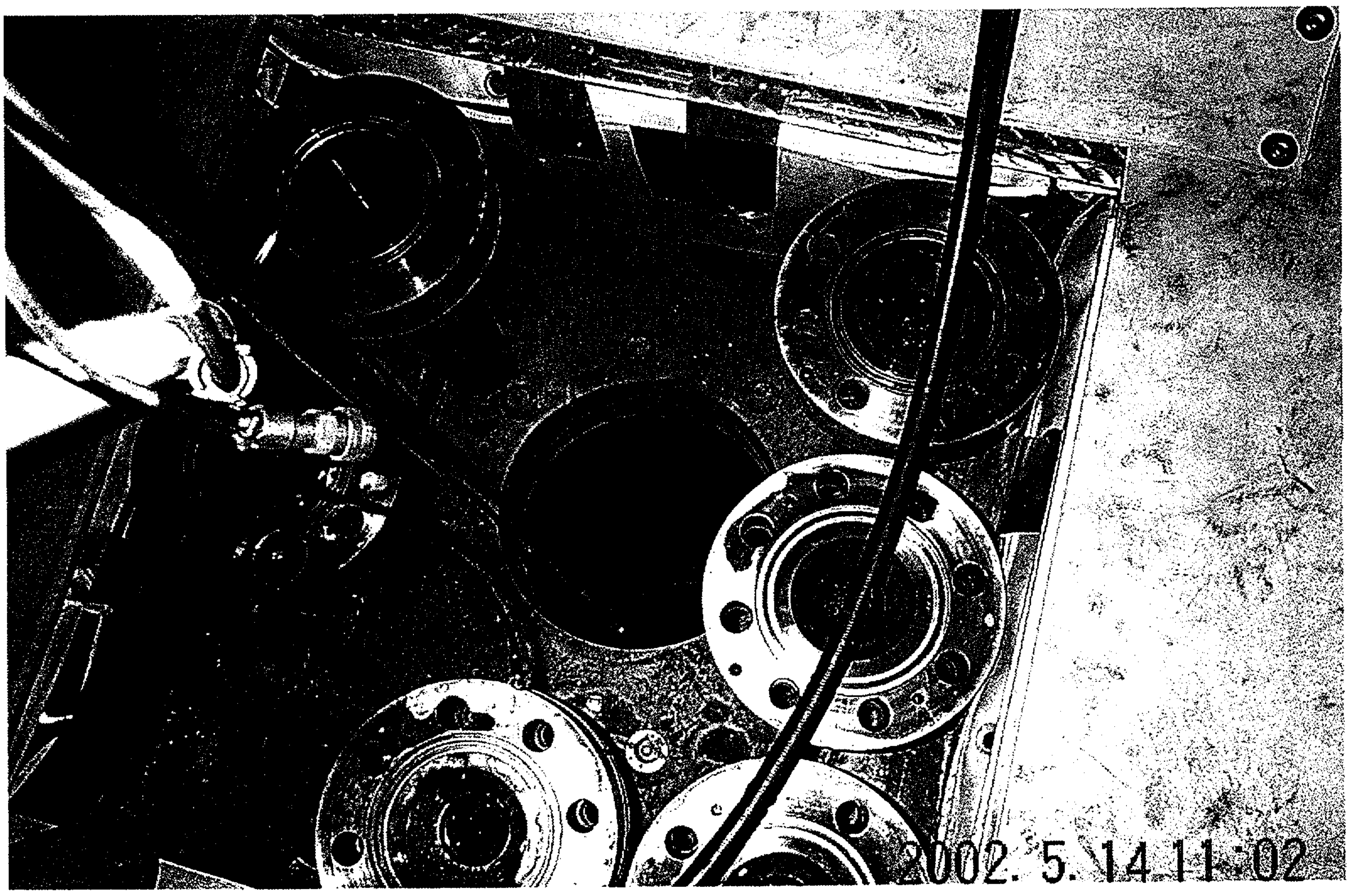
No flaw indication - ◎



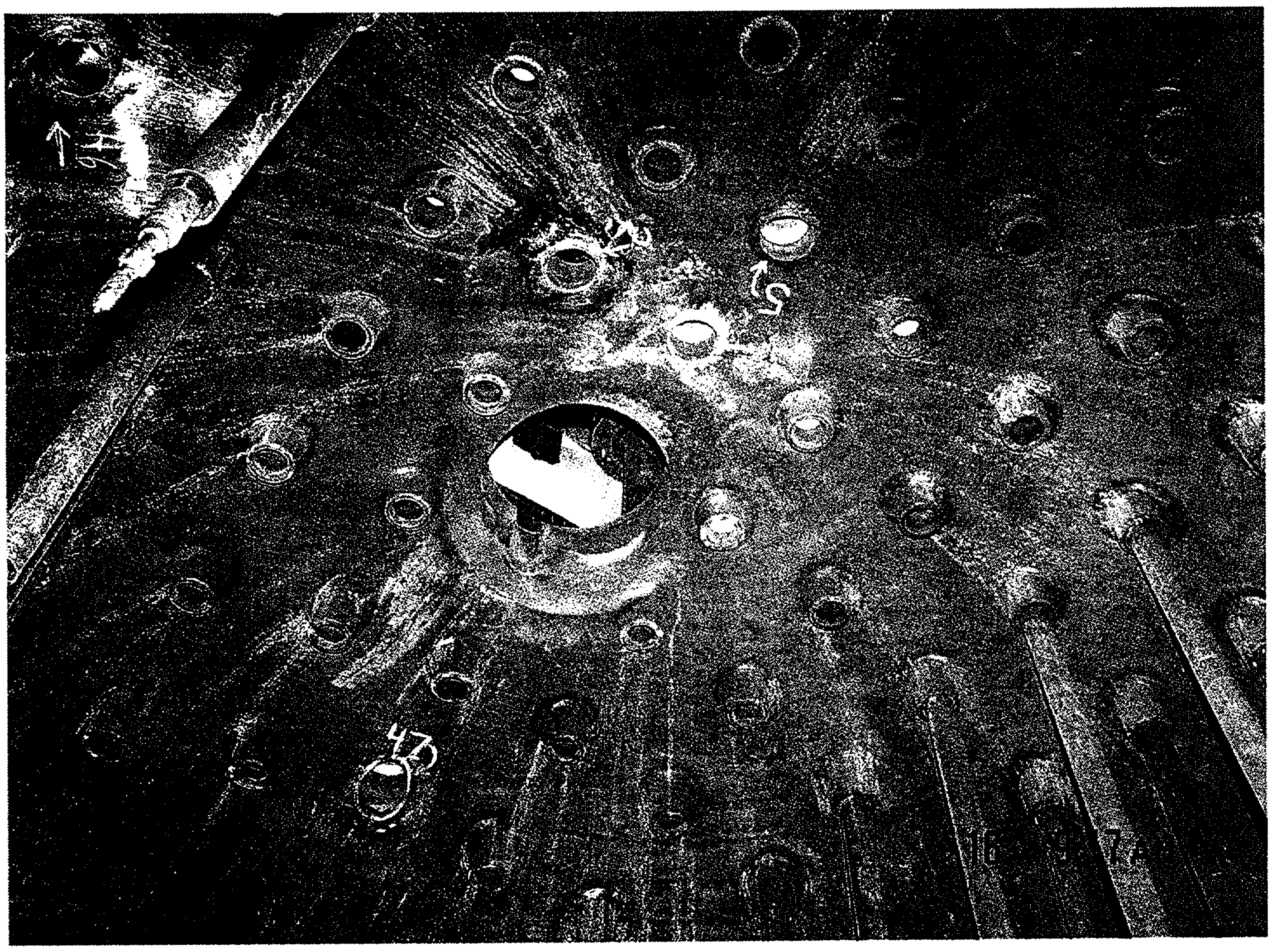
Abrasive Water Jet

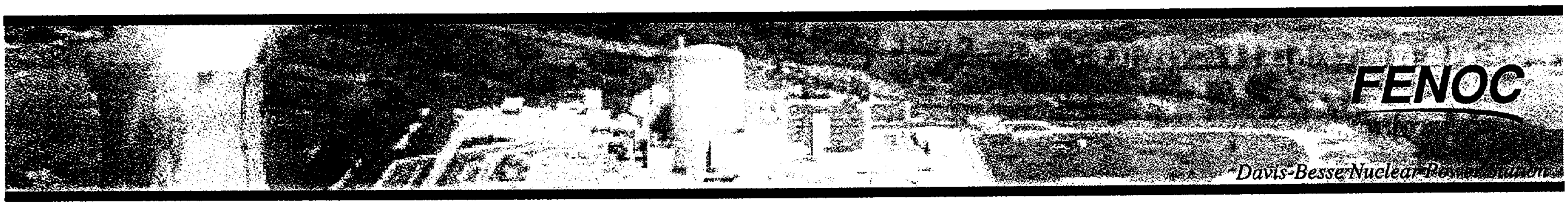


Area Removed from RPV Closure Head



Underneath RPV Closure Head

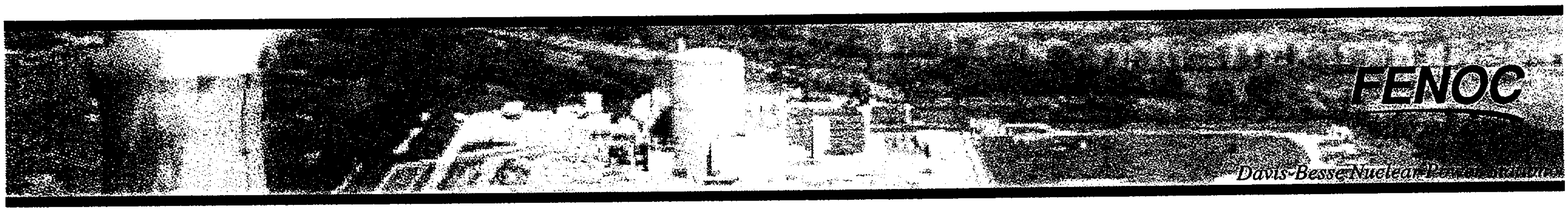




RPV Closure Head Cutout

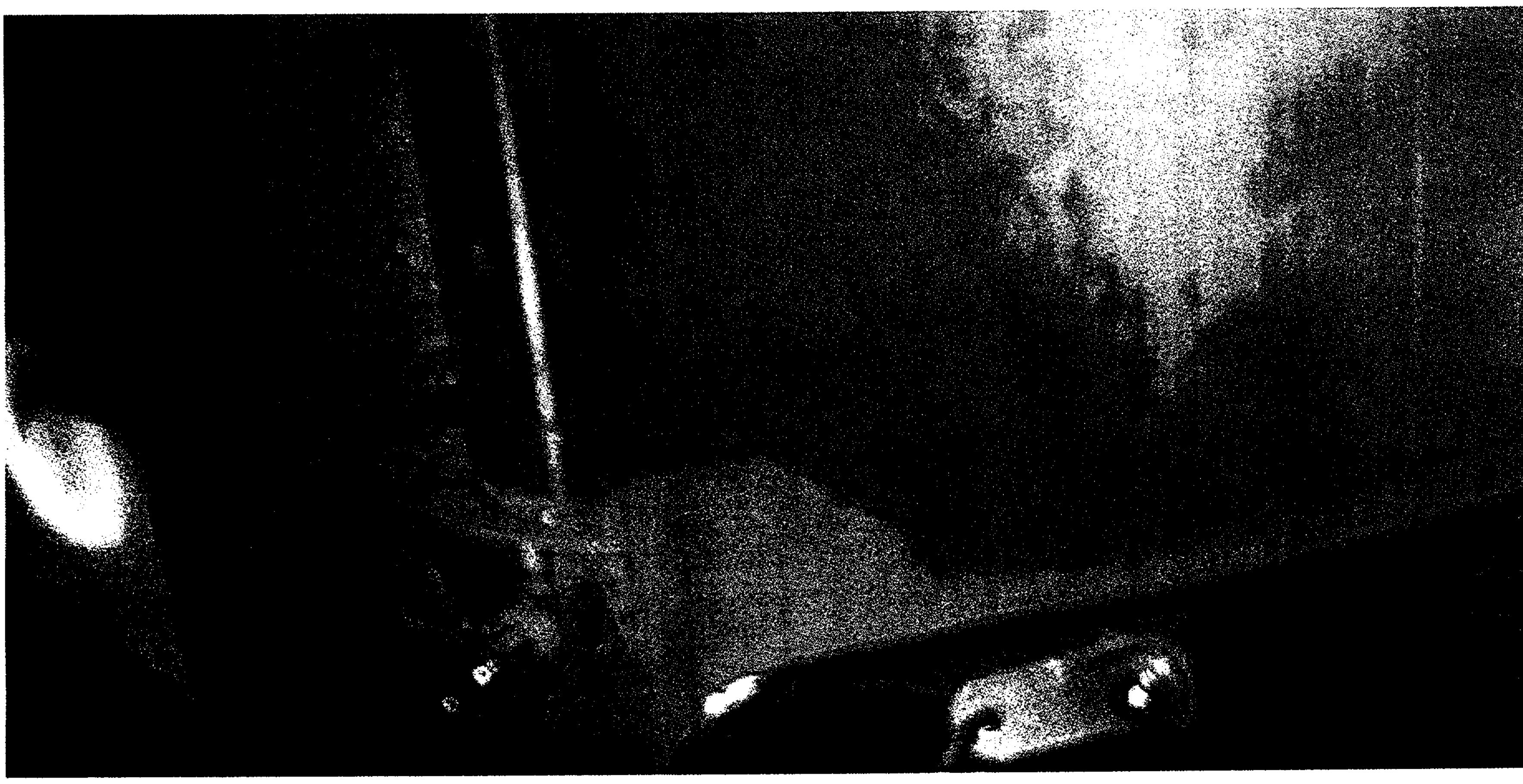
Sample Plan

- Phase 1
  - Corrosion products/boric acid deposits from top of head
  - Deposits scraped from CRD nozzle 3 below the flange
  - Draft report issued for Davis-Besse review
- Phase 2
  - Corrosion products/boric acid deposits from nozzle 2 removal
- Phase 3
  - Nozzle 3 and nozzle 3 corrosion area
  - Nozzle 2



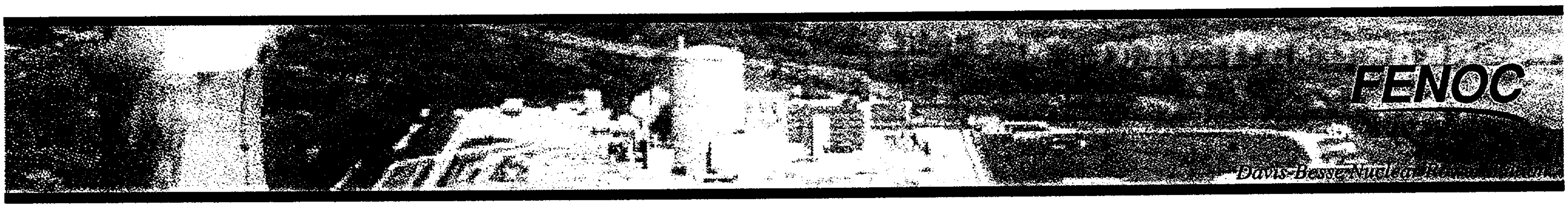
RPV Closure Head Cutout

Nozzle 3 Cutout Cladding Interface



Reactor Pressure Vessel Closure Head (RVPCH) Replacement

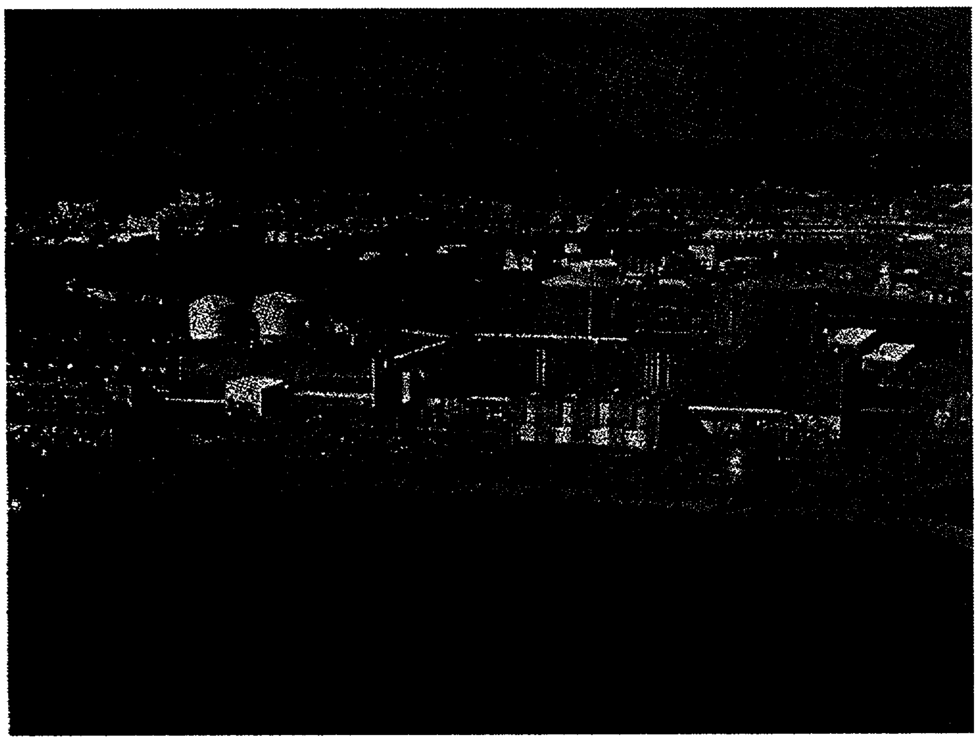
Bob Schrauder
Engineering Services



RPVCH Replacement Considerations

- Evaluated several replacement options
  - Repair existing RPVCH
  - Fabricate new RPVCH
  - Purchase existing RPVCH

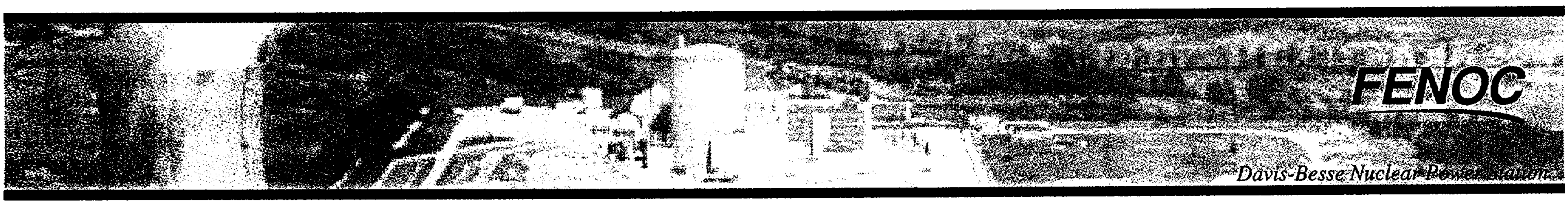
RPVCH Replacement Considerations



- The Midland RPVCH is
  - Similar in design to the Davis-Besse RPVCH
  - Readily available
  - Not contaminated

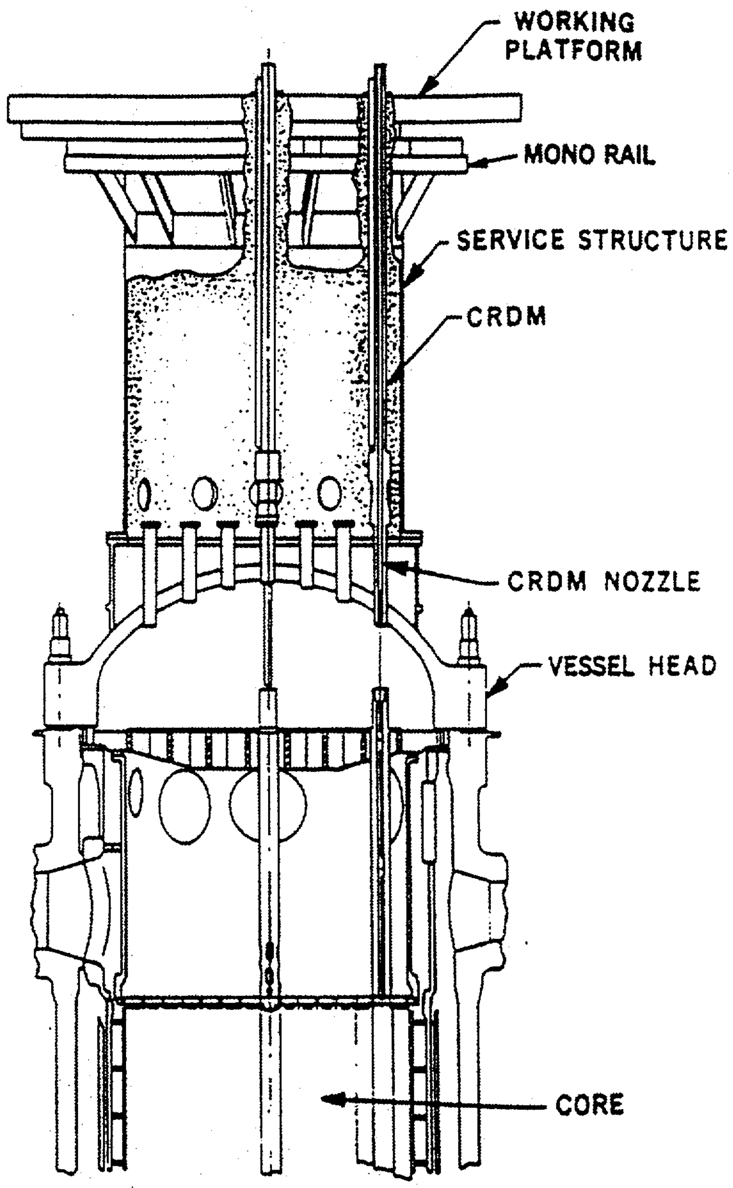
Replacement RPVCH

- Midland RPVCH was fabricated by Babcock and Wilcox
  - Manufactured to ASME Boiler & Pressure Vessel Code Section III, Code Class A, 1968 Edition, Summer 1968 Addenda
  - Accepted by Consumers Power and an Authorized Nuclear Inspector as an acceptable ASME component
  - Hydrostatically tested at 3125 psig per ASME Code Requirements



Replacement RPVCH

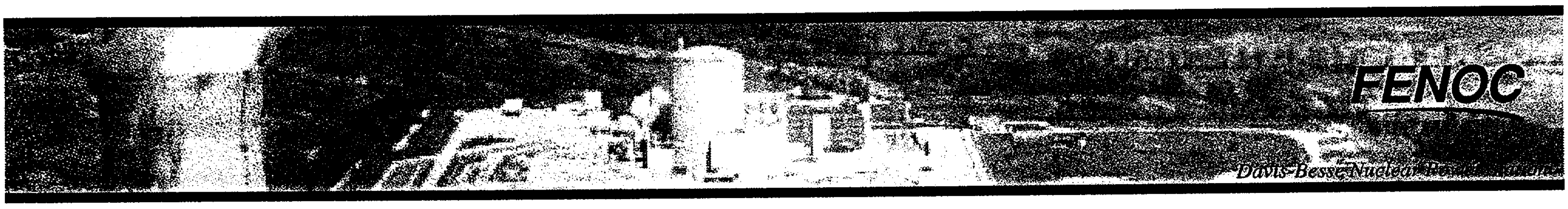
- Framatome-Advanced Nuclear Power (FRA-ANP) has purchased Midland RPVCH and is compiling/validating the ASME Code Data Package
- FRA-ANP is reconciling the Midland RPVCH against Davis-Besse design requirements
- FRA-ANP activities are governed by their safety-related Quality Assurance program, including 10CFR21 reporting



Replacement RPVCH Design

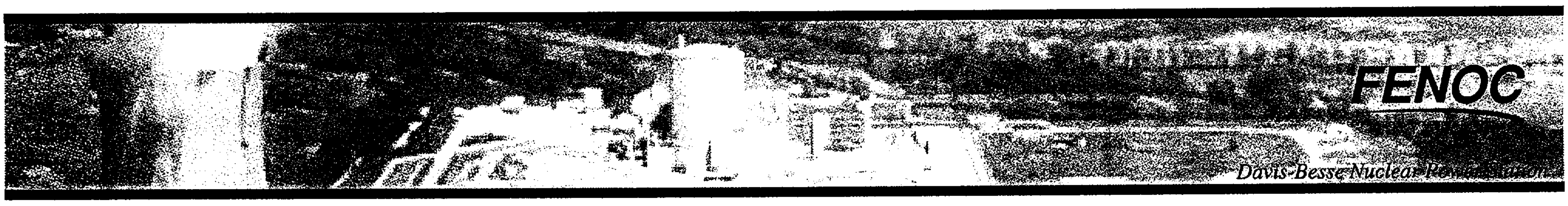
Replacement RPVCH Comparison to Davis-Besse RPVCH

	Davis-Besse	Midland
Material of Construction		
Closure Head	SA-533, GR B Cl 1	Same
Closure Head Flange	SA-508, Cl 2	SA-508-64, Cl 2
CRDM Nozzle	Inconel SB-167	Same
CRDM Flange	SA-182, F-304	Same
Design		
Pressure	2500 psig	Same
Temperature	650 degree F	Same

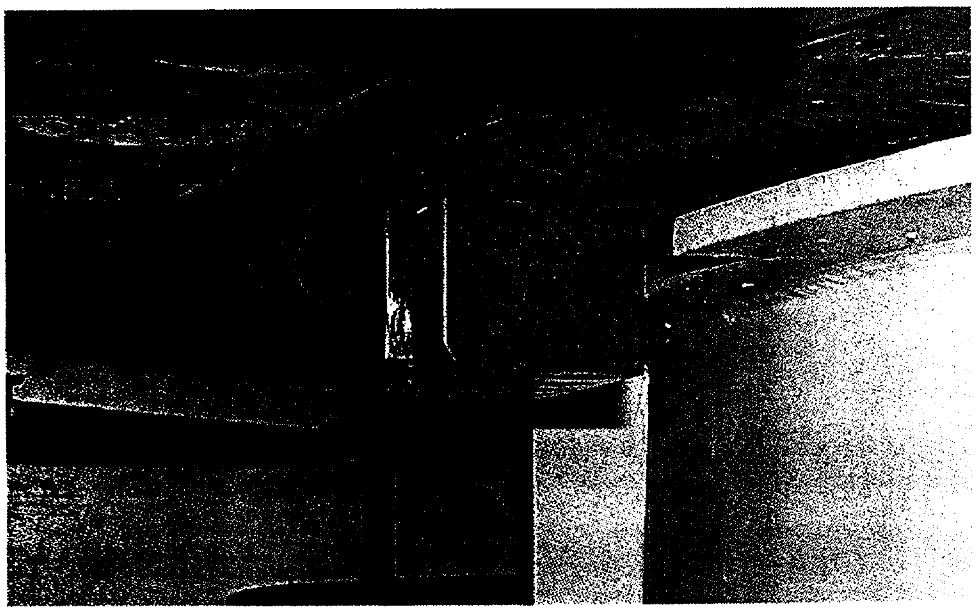


Replacement RPVCH CRD Nozzles

- Midland's Control Rod Drive (CRD) nozzles are similar to Davis-Besse
  - 68 Nozzles: Material Heat M7929
  - 1 Nozzle: Material Heat M6623
- Alignment of control rods to RPVCH nozzles is consistent with original Davis-Besse design

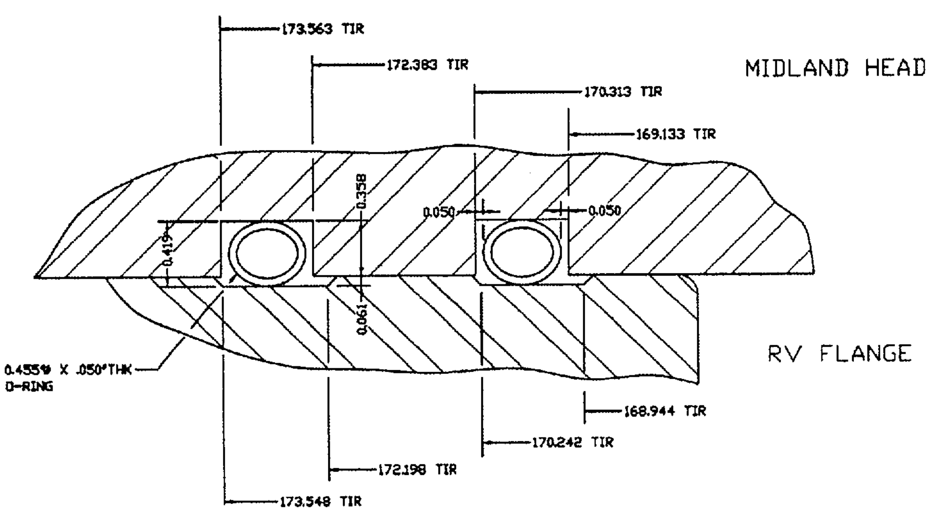


Replacement RPVCH

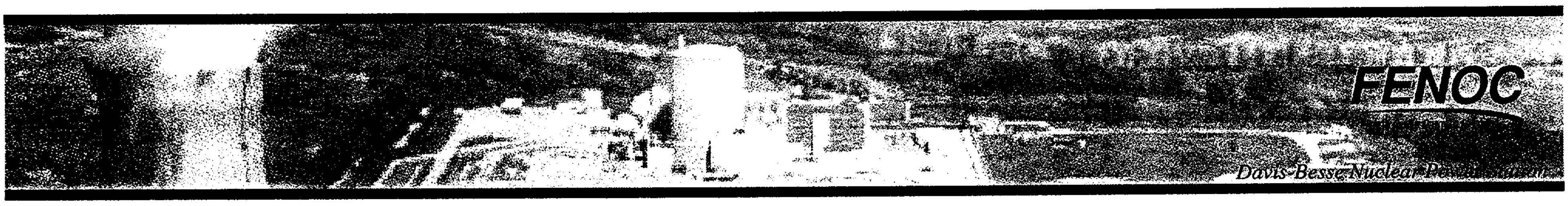


- Minor machining of 4 out of 8 vessel-to-head key-way surfaces is required
- The Midland CRDM flange indexing pin hole locations will be modified to match the proper Davis-Besse azimuth-orientation

Replacement RPVCH



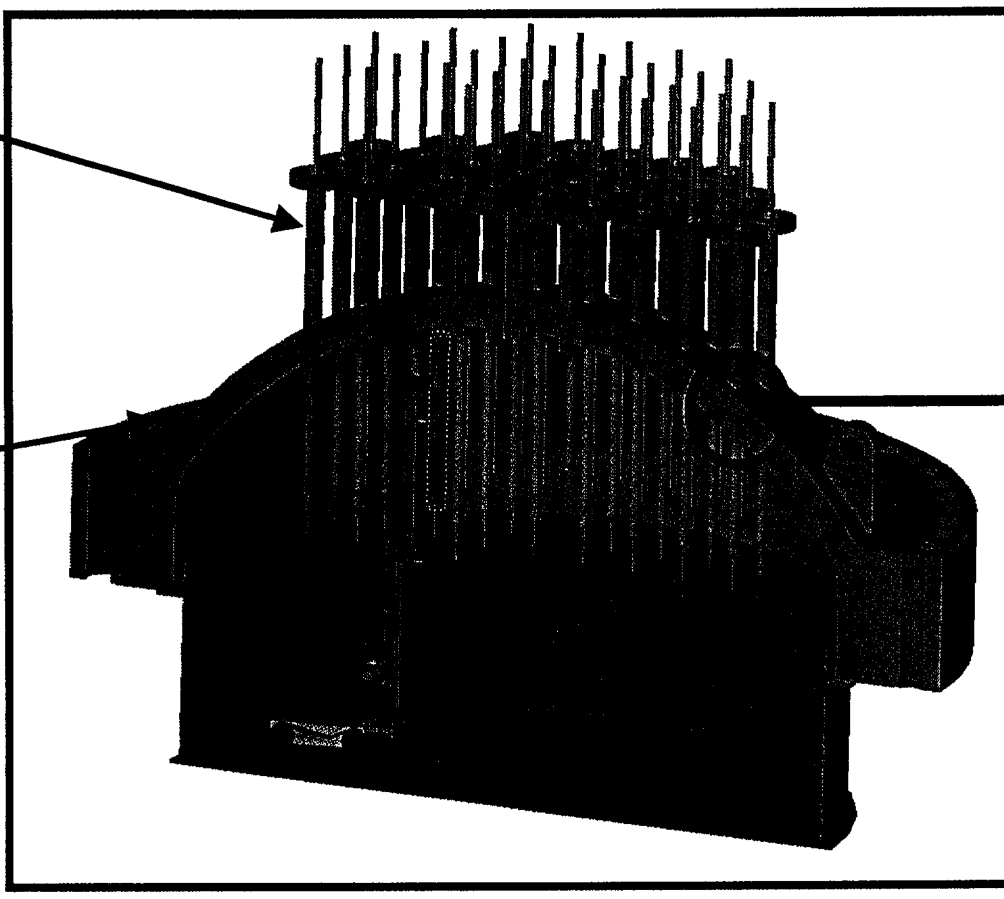
- Minor differences in RPVCH O-ring design
  - O-ring grooves are slightly different requiring the use of smaller diameter O-rings (0.455 in. vs 0.500 in.)
  - New O-rings will be installed

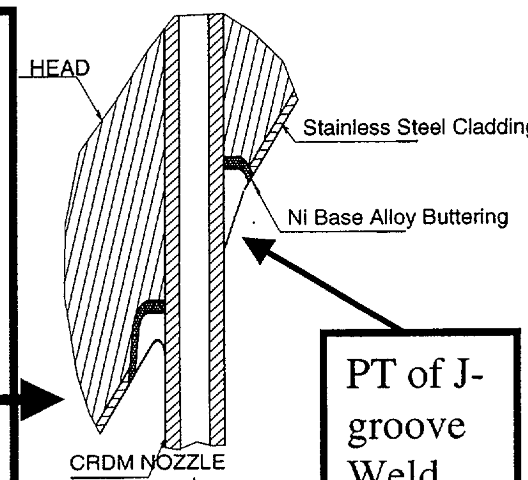


Examinations of Replacement RPVCH

RT and PT of CRD Nozzle to Flange Weld

RT, UT, and MT of Flange to Dome Weld





PT of J-groove Weld

UT and ET of CRD Nozzle

Examinations of Replacement RPVCH

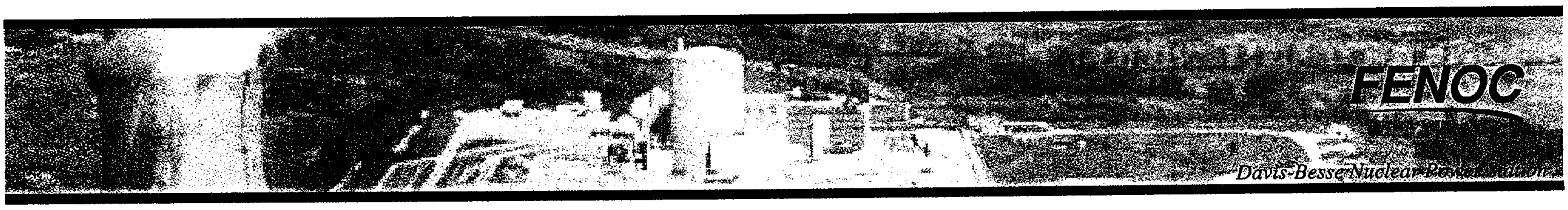
- Examinations to supplement ASME Code Data Package:
  - Visual examinations
  - Radiography (RT) of flange-to-dome weld
    - Lifting attachments prevented full coverage
  - RT of nozzle-to-flange welds
  - PT examination of the CRDM nozzle J-groove welds

Examinations of Replacement RPVCH

- Preservice Inspections
  - Magnetic Particle (MT) examination of flange-to-dome weld
  - Ultrasonic (UT) examination of flange-to-dome weld
  - Liquid Penetrant (PT) examination of peripheral CRDM nozzle-to-flange welds

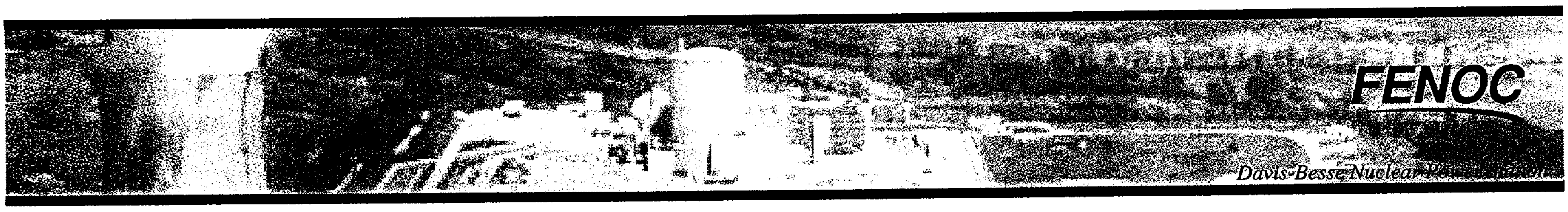
Examinations of Replacement RPVCH

- Additional Non-Destructive Examinations
  - Chemical smears
  - Baseline UT of CRD nozzles
  - Eddy Current Testing (ET) of CRD nozzles



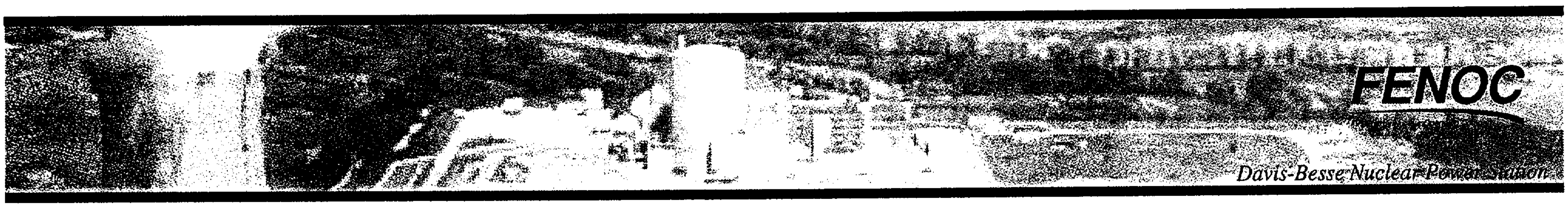
Installation of the Replacement RPVCH at Davis-Besse

- Davis-Besse Containment Building will require temporary access opening
- Original RPVCH will be moved outside Containment Building for storage and/or disposal
- Davis-Besse Service Structure will be used
- Inspection ports will be installed on replacement support skirt



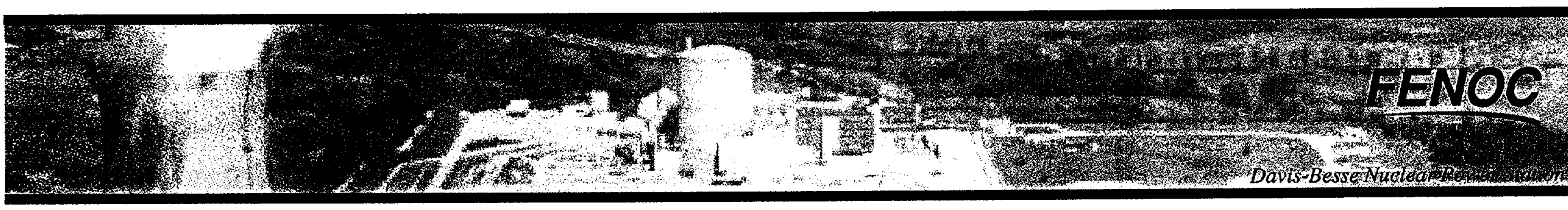
Installation of the Replacement RPVCH at Davis-Besse (continued)

- Original Davis-Besse control rod location and core configuration will be used
  - Existing CRD Mechanisms will be used
  - CRD Mechanisms nozzle flange split nut ring modification will be performed
  - Upgraded gasket design will be incorporated



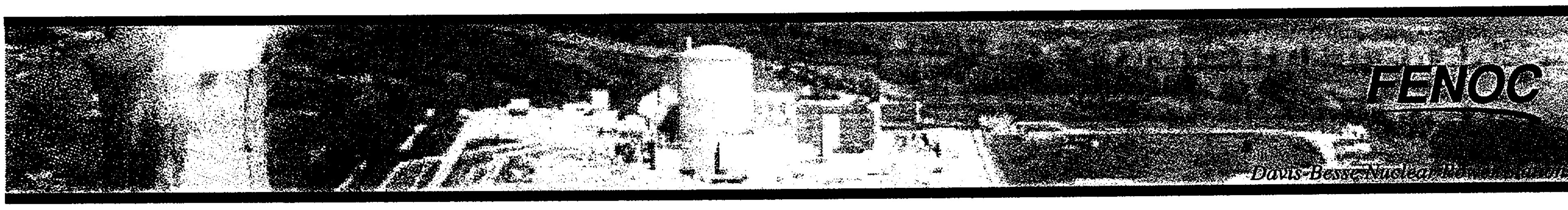
RPVCH Planned Post-Installation Activities

- Fill and vent RCS
  - Perform visual inspection for leakage
- Bring plant to normal operating temperature and pressure using Reactor Coolant Pump heat
  - Perform visual inspection for leakage
- Perform control rod drop time testing per Technical Specifications



NRC Approvals Identified to Date

- 10 CFR 50.55a approvals
  - Existing request RR-A2 for flange-to-dome weld volumetric examination
  - Existing request RR-E4 for VT-2 visual examination of containment building access opening following restoration
- No Technical Specification changes



Root Cause Investigation

Steve Loehlein
Root Cause Investigation Team Leader

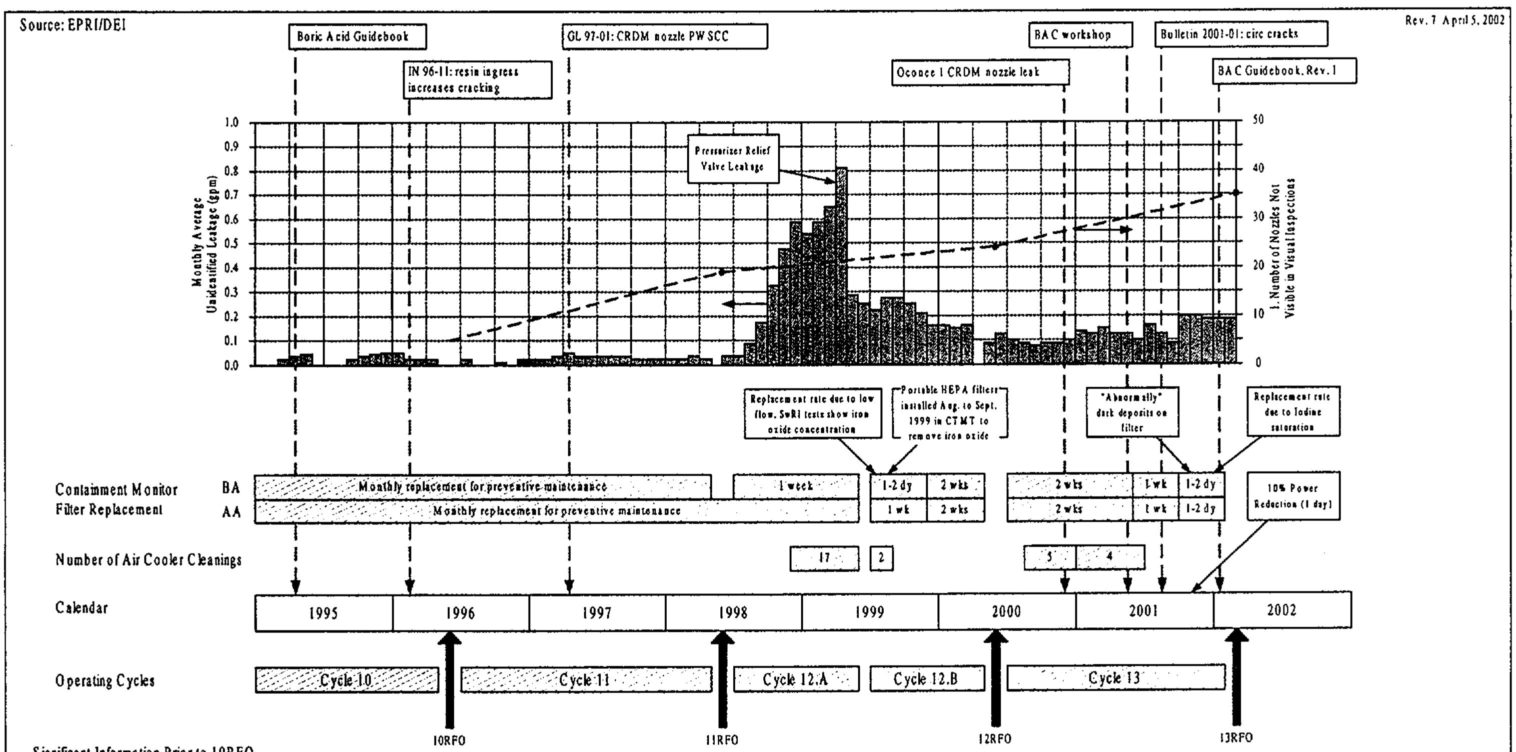
Key Questions

- Was there a new mechanism that caused this degradation?
- Was there adequate guidance/knowledge available to have prevented the degradation to the RPV closure head?

Key Conclusions

- The degradation to the RPV closure head was caused by Primary Water Stress Corrosion Cracking (PWSCC) of the Control Rod Drive (CRD) nozzle which led to leaks that were undetected allowing boric acid corrosion to occur
- The existing guidance/knowledge was adequate for preventing unacceptable RPV closure head degradation from CRD nozzle leaks

Timeline of Key Events



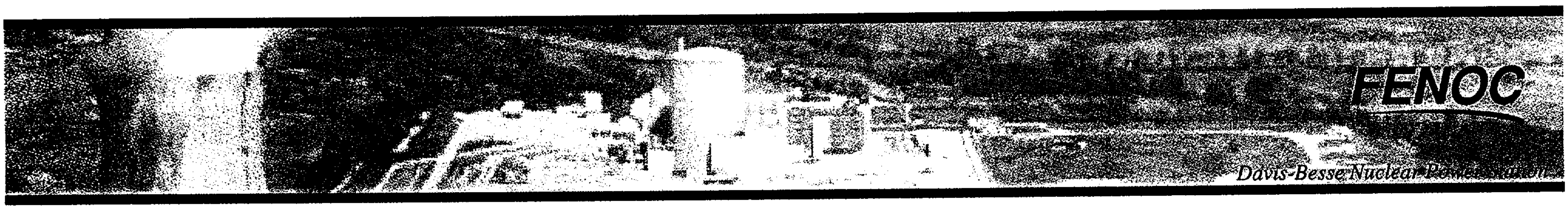
Significant Information Prior to 10RFO

1) Nozzle 66 gasket replaced at 7RFO and 8 RFO		1) Nozzle 31 gasket leaks but not changed 2) BA is powder and white	1) Replaced gaskets on flanges 3, 5, 6, 11, 31 2) "Red" deposits under nozzle 3 flange 3) Bottom of nozzle 3 flange not inspected due to hard BA	1) No leakage from flange gaskets 2) "Red" deposits under nozzle 3 flange 3) Difficult to get camera in place due to hard, brown, boric acid	CRDM Flange Conditions
1) Head flange clean in 1994	1) BA flowing towards mouseholes in S-E quadrant	First indication of red colored boric acid deposits from mouseholes in quadrant of S-E quadrant	Significant flow of red colored boric acid deposits from mouseholes, piling to 6" deep behind studs in S-E quadrant	1) Significant flow of boric acid deposits from mouseholes in S-E quadrant 2) BA deposits extend around entire flange 1" deep max	RPV Flange Conditions
1) BA deposits dripping through insulation at 8RFO 2) Head "cleaned" at 8RFO 3) Head not inspected during 9RFO	1) Visual inspection 65/69 nozzles 2) Minor BA deposits around outer nozzles of RV Head 3) BA deposits at center of head	1) Vsual inspect 50/69 nozzles 2) Loose BA piling up behind outer nozzles 3) BA brittle, breaks easily	1) Visual inspect 45/69 nozzles 2) BA definitely red and hard to break up (needed crowbar) "lava-like"	1) Visual inspect 34?/69 nozzles N-1: leak [Nozzles 5 & 47 cracked] N-2: leak, circ & minor wastage N-3: leak & major wastage	RPV Head Conditions

Figure 26. Timeline of Key Events Related to Reactor Vessel Head Boric Acid Corrosion

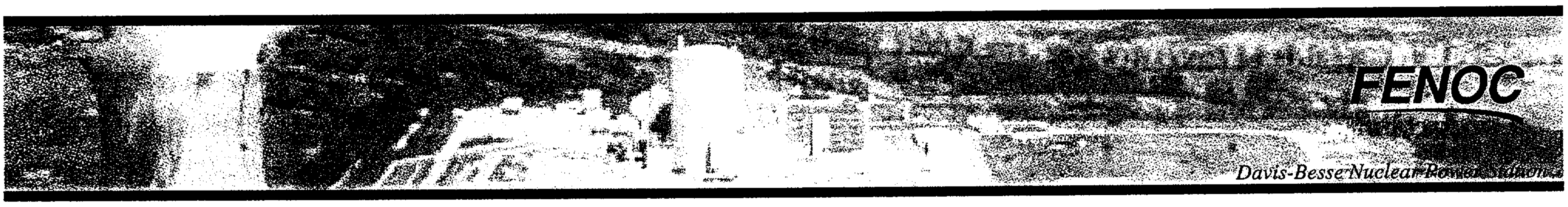
Conclusions Regarding Identified Cracking

- Cracking mechanism is PWSCC
  - Flaw characteristics found at Davis-Besse are similar to other plants with confirmed PWSCC
  - No factors indicating sulfide-induced intergranular stress corrosion cracking (IGSCC) due to chemistry transients
  - No other cracking mechanism deemed credible

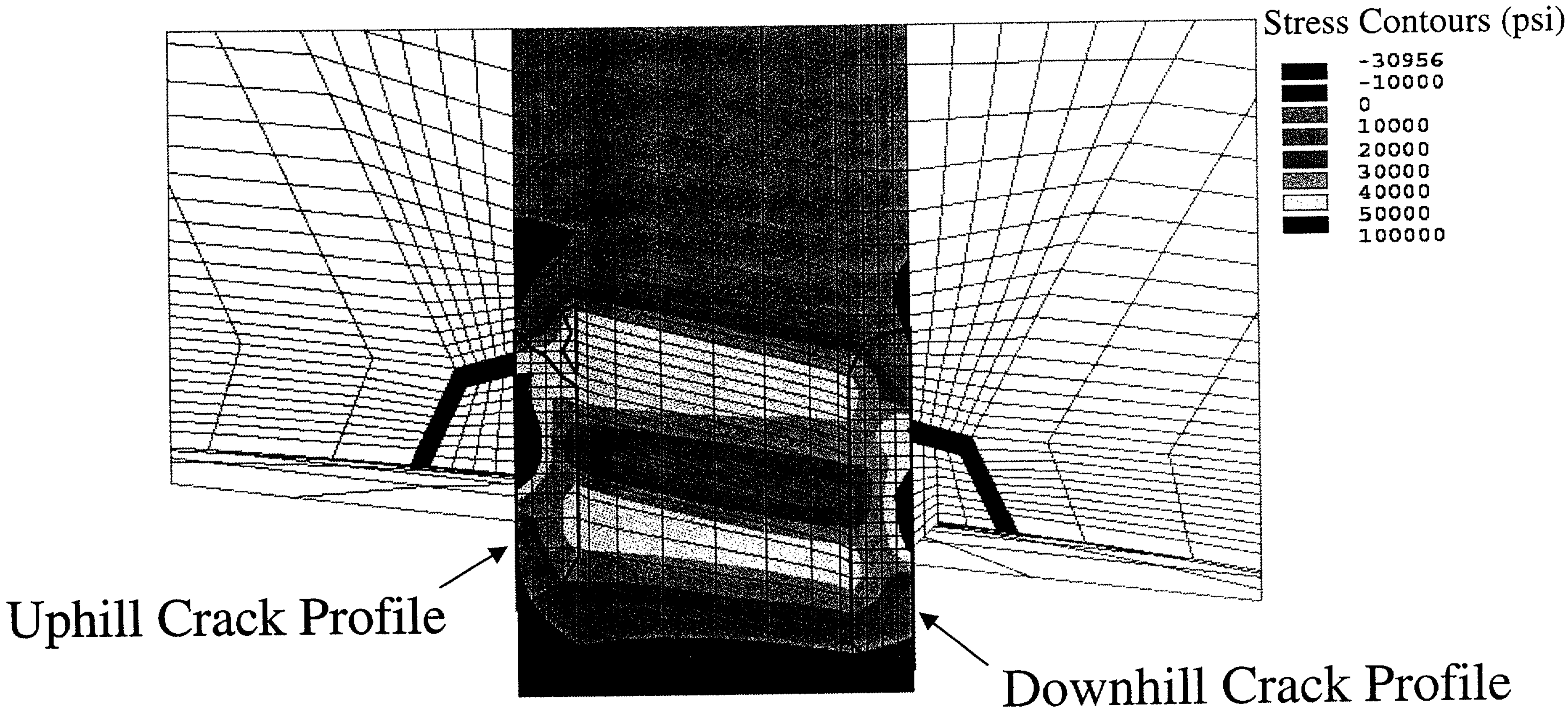


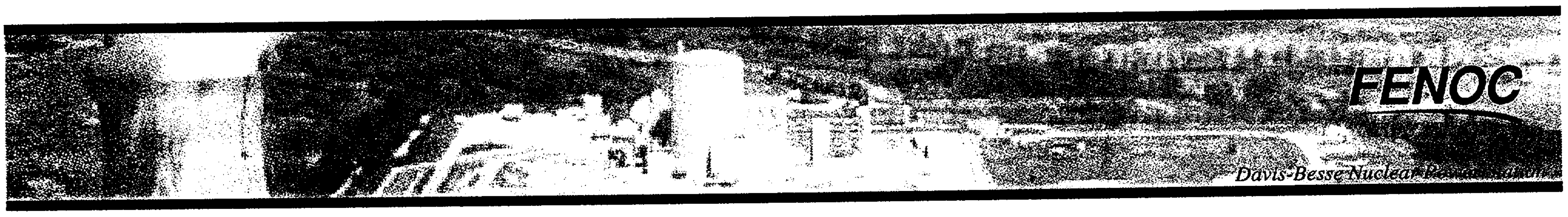
Estimated Crack Propagation Timeframe

- Longest through-wall cracks estimated to have initiated in 1990 (+/- 3 years)
- Estimated time for flaw to propagate through-wall is 4-6 years
- Consistent with proposed EPRI Material Reliability Program crack growth rate curve



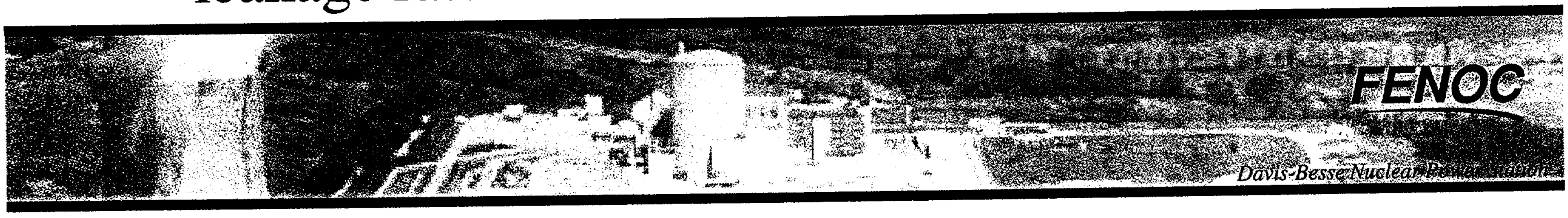
Davis-Besse Nozzle #3
Operating Hoop Stress and Axial Crack Profiles





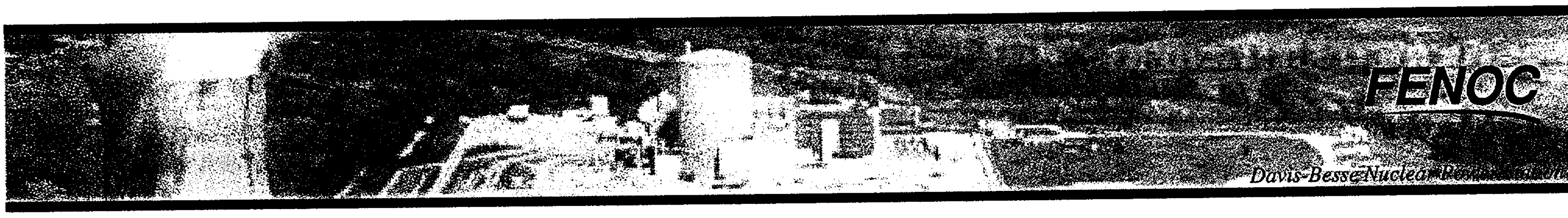
Leakage From Cracked Nozzles

- Through-wall cracking in nozzle or J-groove weld leads to leaks into annulus region
- Leakage rate is a function of crack length above J-groove weld and degree of cracking through the weld
- Leakage rate increases significantly as crack lengthens above the J-groove weld due to increase in crack width
- Previous industry observations indicated very low leakage rates

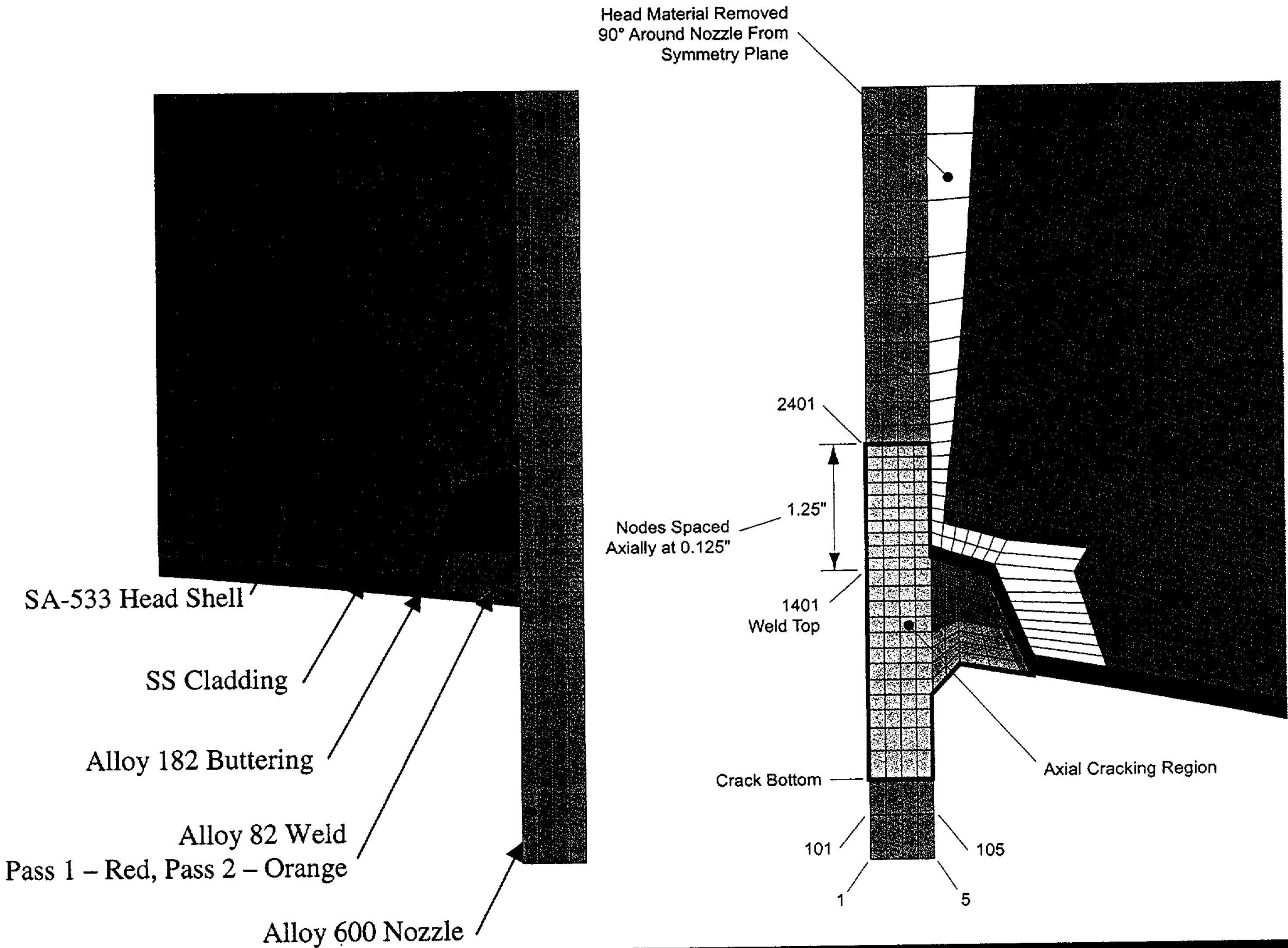


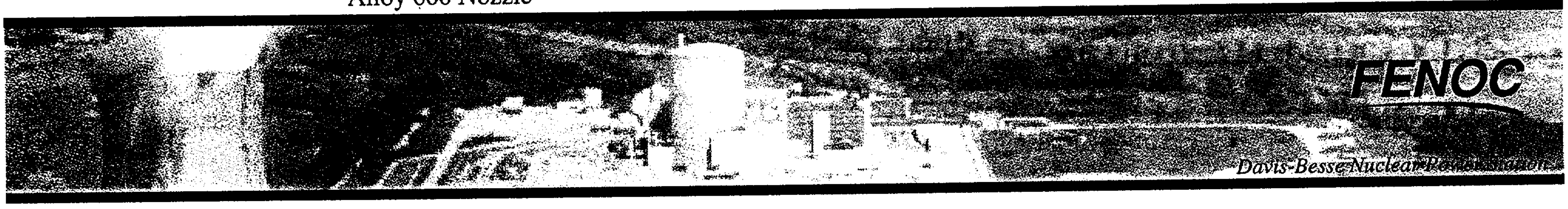
Davis-Besse
Leakage Rate from Cracked Nozzle

- Davis-Besse axial cracks above weld were longer than reported from other plants (1.1 inches for nozzle 2 and 1.2 inches for nozzle 3)
- Analytical leakage predictions yield wide range of results (.025 to >1 gpm) depending on method and assumed geometry used
- Estimated leak rate based on boric acid deposits and unidentified leakage are in the range of 0.04 to 0.2 gpm

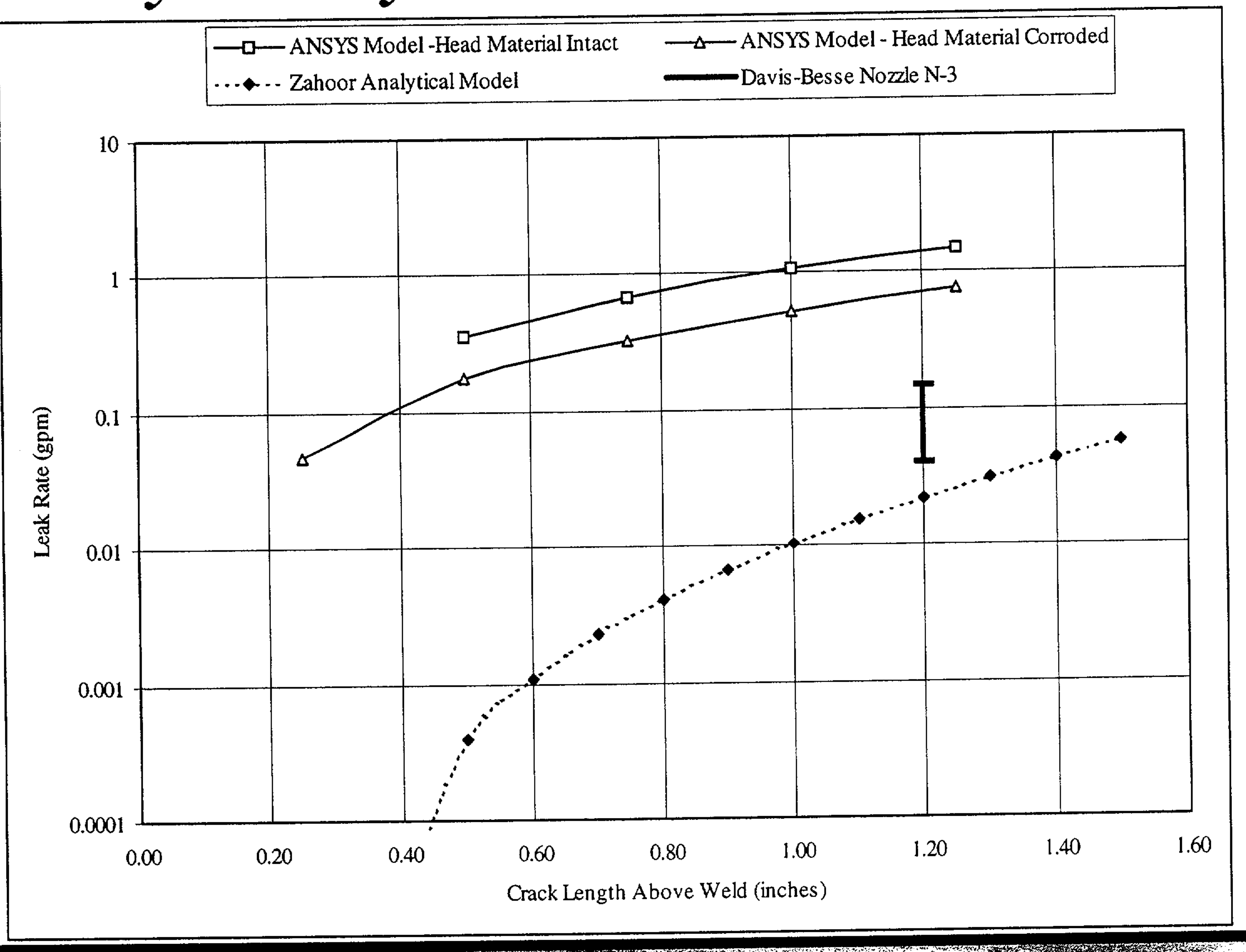


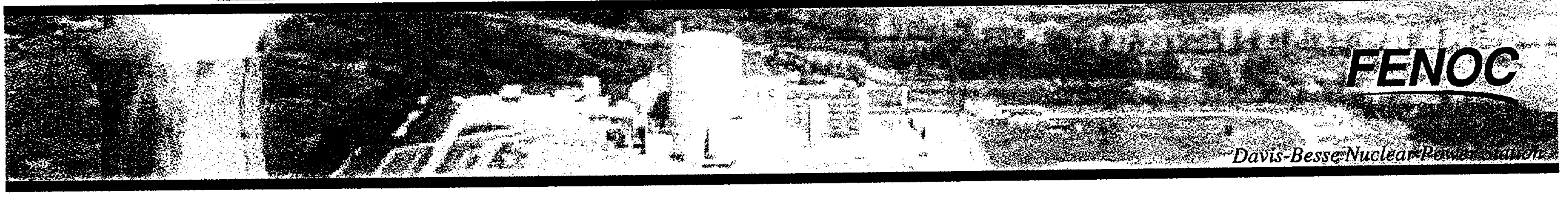
Nozzle 3 Crack Finite Element Model





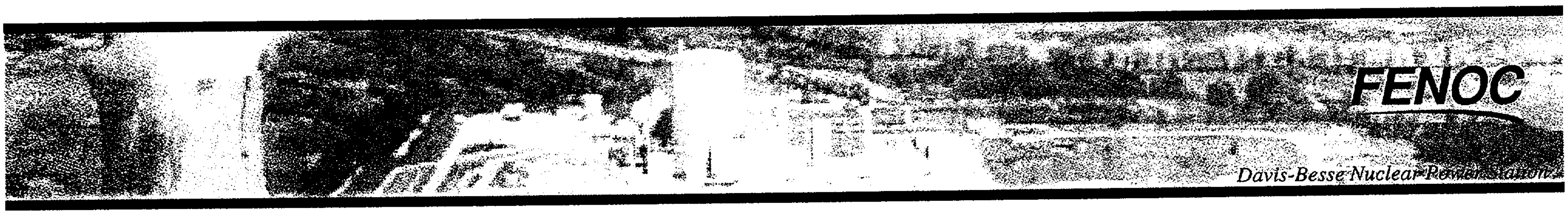
Analytically Predicted Leak Rates





Source of Corrosion

- Degradation at nozzle 2 and 3 is due to boric acid corrosion
- Boric acid corrosion is a known mechanism capable of producing such significant degradation
- There is a history of boric acid corrosion incidents on RPV heads in the industry



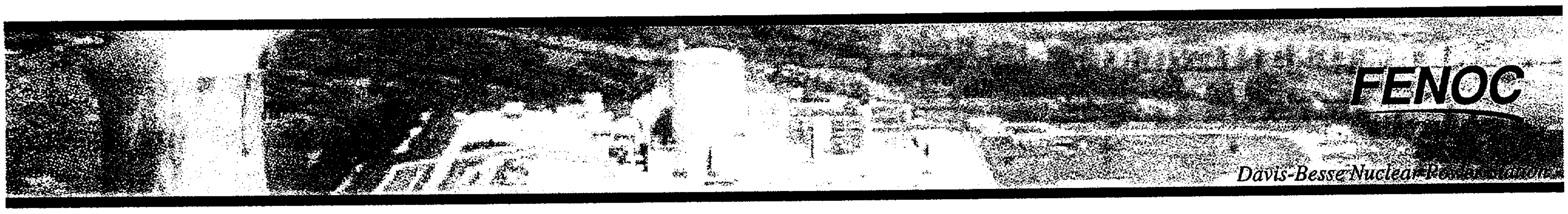
Degradation Sequence

Stage 1 - Crack Initiation Progression

Stage 2 - Minor Weepage / Latency Period

Stage 3 - Deep Annulus Corrosive Attack

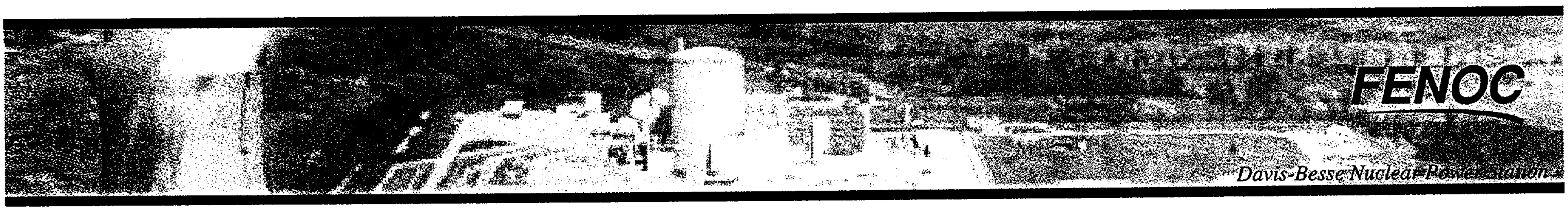
Stage 4 - General Boric Acid Corrosion



Stage 1

Crack Initiation Progression

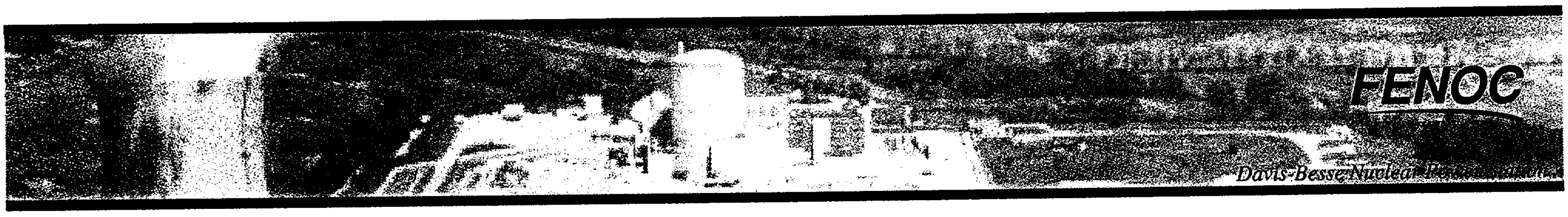
- Nozzle 3 cracks resulted from PWSCC
- Cracks grew at rate consistent with industry data
- RCS leakage miniscule



Stage 2

Minor Weepage/Latency Period

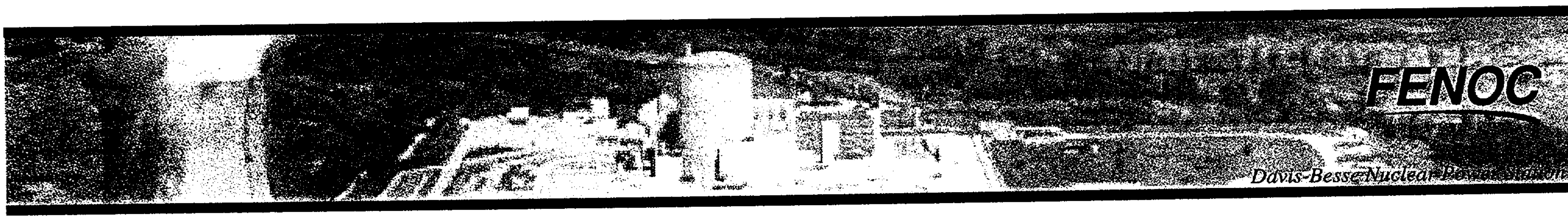
- Leakage entered annulus between Alloy 600 nozzle and low alloy steel RPV closure head
- Fit allowed capillary flow path
- Latency period could involve several mechanisms (e.g., steam cutting, galvanic corrosion, crevice corrosion, and flow accelerated corrosion)
- Annular gap increased due to localized corrosion resulting in leakage flow (residual and dry steam) reaching surface
- Leak rate controlled by number of cracks and size of cracks (length and width)



Stage 3

Deep Annulus Corrosive Attack

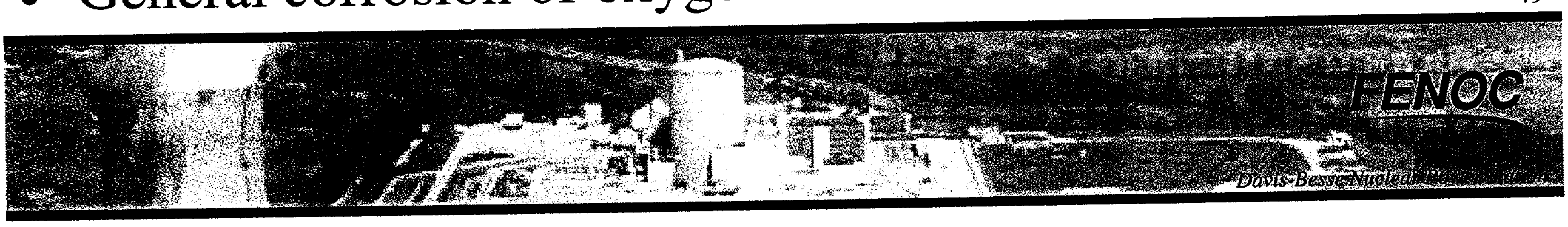
- Oxygen penetration in annulus increased due to decreasing velocity and differential pressure in annulus
- Preferential corrosion occurred in the vicinity of crack (consistent with EPRI-6 test)
- Exiting steam mass flow from annulus region not sufficient to wet surrounding areas
- Nozzle 2 progressed to this stage



Stage 4

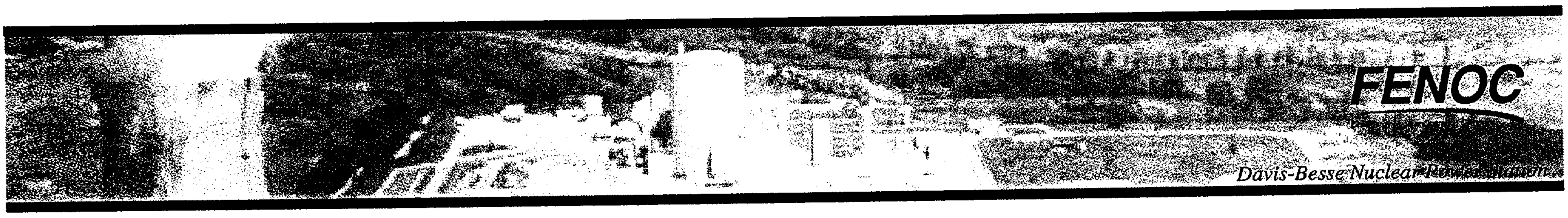
General Boric Acid Corrosion

- Corrosion progression limited by crack growth rate and leakage through crack
- Annulus flooded with moist steam
- Boric acid accumulates on head
- Increased leakage provides localized cooling of head allowing greater wetted area
- Affected area governed by thermodynamics and material properties (e.g., viscosity, density, slope)
- General corrosion of oxygenated surface



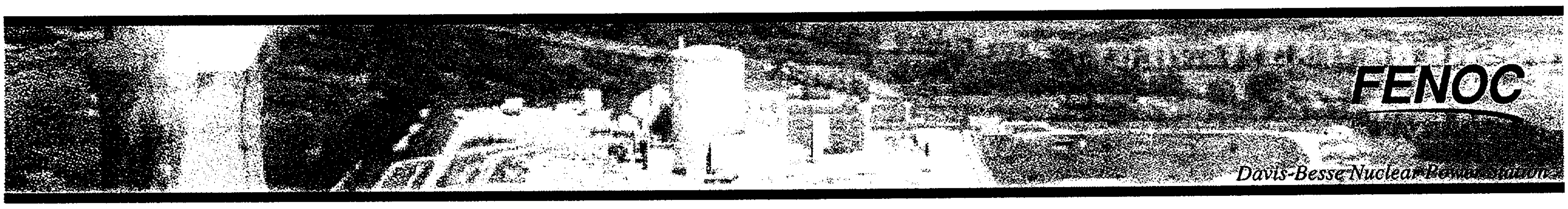
Corrosion Rates From Industry Testing

- EPRI and industry testing (effect of boric acid on low alloy steel) demonstrates corrosion rates of 0.6 to 5.0 inches per year
- EPRI - 6 Test
  - Tests performed using deaerated, high-temperature water (600° F)
  - Orientation, geometry and materials simulated RPV head nozzles
  - Flow rates of 0.01 and 0.10 gpm used in test



Davis-Besse
Estimated Reactor Vessel Closure Head Corrosion Rates

- 4 years of stage 4 corrosion
- Maximum radial progression ~7 inches
- Average rate ~2 inches per year
- Lateral direction corrosion rate ~1/2 that of axial direction
- Consistent with EPRI Boric Acid Corrosion Guidebook

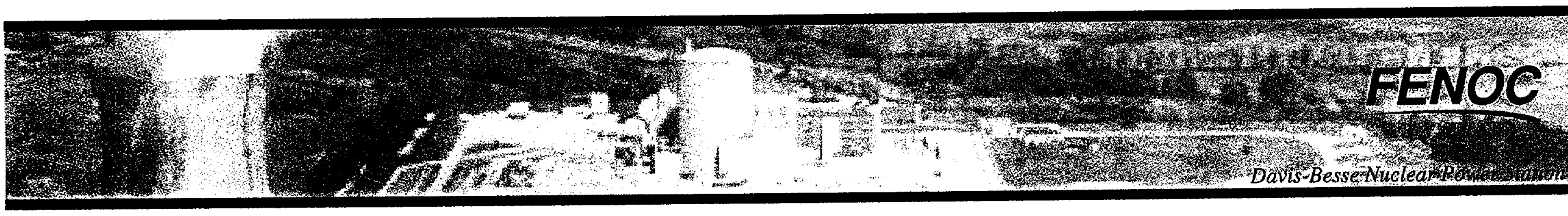


Root Cause Summary

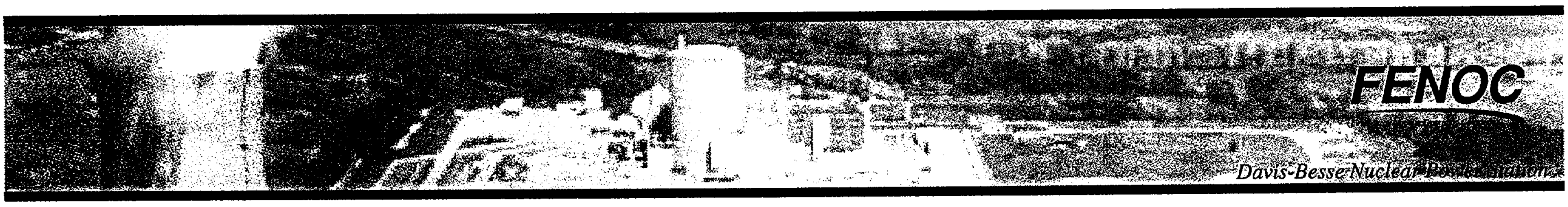
Inadequate inspection of the RPV closure head prevented early detection of nozzle leakage,

resulting in prolonged boric acid corrosion

and significant degradation.



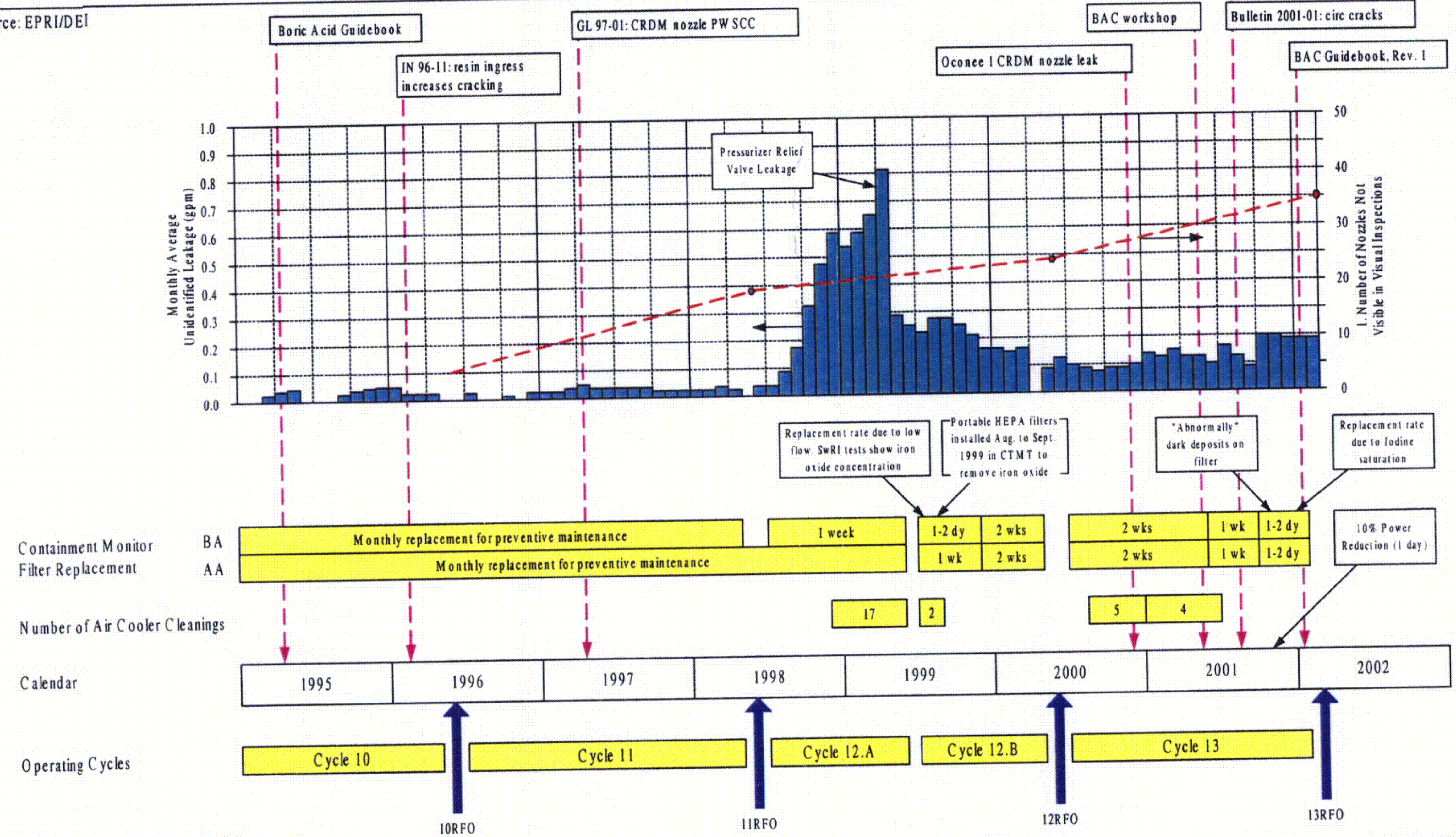
Concluding Remarks



FENOC

Source: EPRI/DEI

Rev. 7 April 5, 2002



Significant Information Prior to 10RFO

1) Nozzle 66 gasket replaced at 7RFO and 8 RFO		1) Nozzle 31 gasket leaks but not changed 2) BA is powder and white	1) Replaced gaskets on flanges 3, 5, 6, 11, 31 2) "Red" deposits under nozzle 3 flange 3) Bottom of nozzle 3 flange not inspected due to hard BA	1) No leakage from flange gaskets 2) "Red" deposits under nozzle 3 flange 3) Difficult to get camera in place due to hard, brown, boric acid	CRDM Flange Conditions
1) Head flange clean in 1994	1) BA flowing towards mouseholes in S-E quadrant	First indication of red colored boric acid deposits from mouseholes in quadrant of S-E quadrant	Significant flow of red colored boric acid deposits from mouseholes, piling to 6" deep behind studs in S-E quadrant	1) Significant flow of boric acid deposits from mouseholes in S-E quadrant 2) BA deposits extend around entire flange 1" deep max	RPV Flange Conditions
1) BA deposits dripping through insulation at 8RFO 2) Head "cleaned" at 8RFO 3) Head not inspected during 9RFO	1) Visual inspection 65/69 nozzles 2) Minor BA deposits around outer nozzles of RV Head 3) BA deposits at center of head	1) Vsual inspect 50/69 nozzles 2) Loose BA piling up behind outer nozzles 3) BA brittle, breaks easily	1) Visual inspect 45/69 nozzles 2) BA definitely red and hard to break up (needed crowbar) "lava-like"	1) Visual inspect 34?/69 nozzles N-1: leak [Nozzles 5 & 47 cracked] N-2: leak, circ & minor wastage N-3: leak & major wastage	RPV Head Conditions

Figure 26. Timeline of Key Events Related to Reactor Vessel Head Boric Acid Corrosion

C01

Safety Margin Assessment of Davis-Besse Head Wastage Condition

Presented by:
Dr. Nathaniel G. Cofie
Structural Integrity Associates
June 2002

Structural Integrity Associates, Inc.

Summary of Analysis

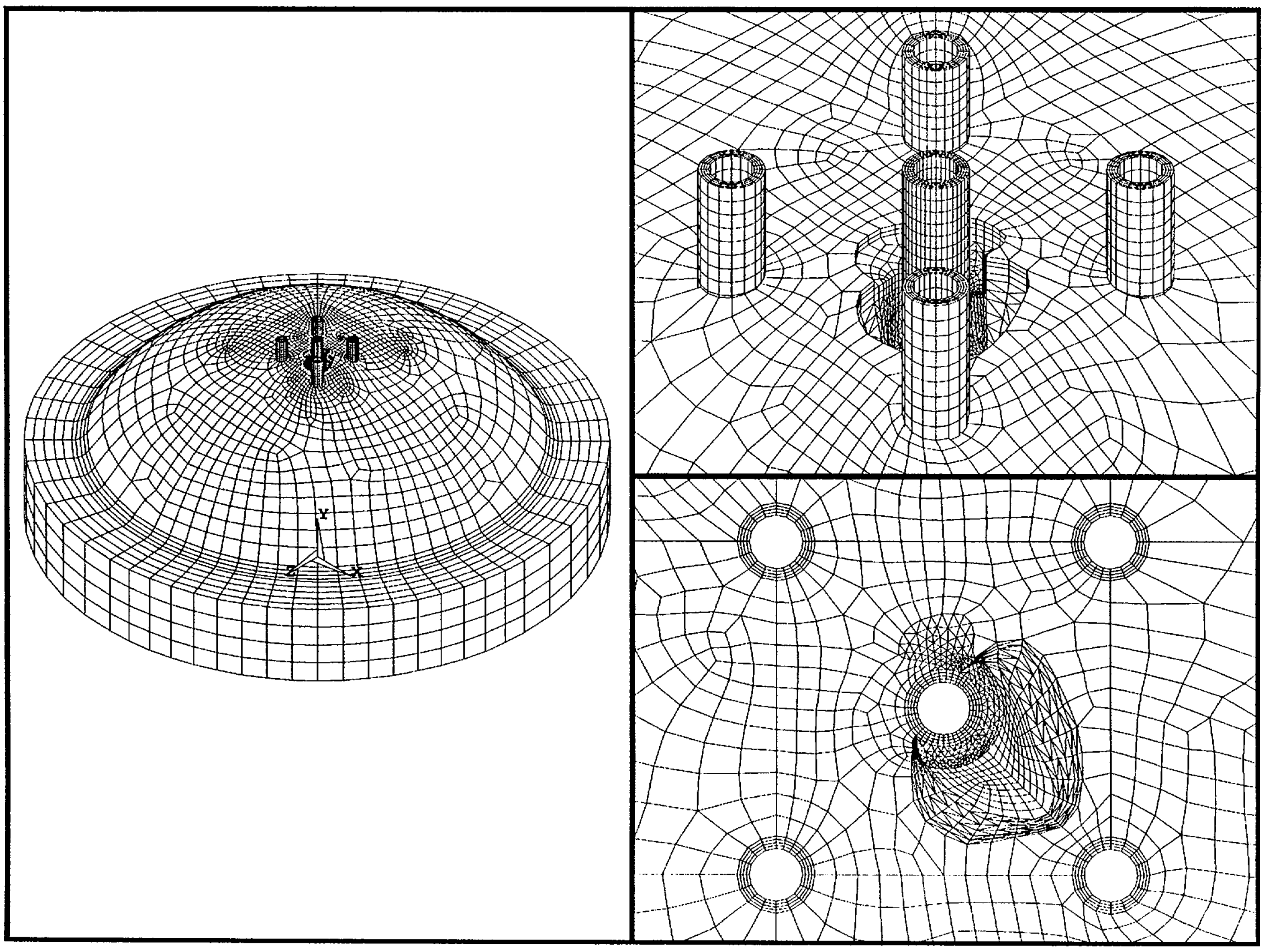
- **Three-Dimensional Finite Element Model**
- **Entire Head, Damaged Nozzle and Adjacent Nozzles Modeled**
- **Incremental Elastic-Plastic, Large Strain Analysis performed**
- **Conservative Stress-Strain Curve used in analysis**
    - Uniform elongation limited to 11.15%
- **Conservative failure criterion applied to analysis**
    - Failure assumed to occur if a row of elements has strains > uniform elongation

Summary of Analysis (cont'd)

- Predicted failure pressure is 5600 psi (> 2 times normal operating pressure) for average clad thickness of 0.297 in. Predicted failure pressure is 4600 psi for minimum measured clad thickness of 0.24 in.

- Analysis procedure and failure criterion compared against physical disk burst tests to demonstrate that burst pressure predictions are conservative

FEM of Davis-Besse Head Wastage Condition



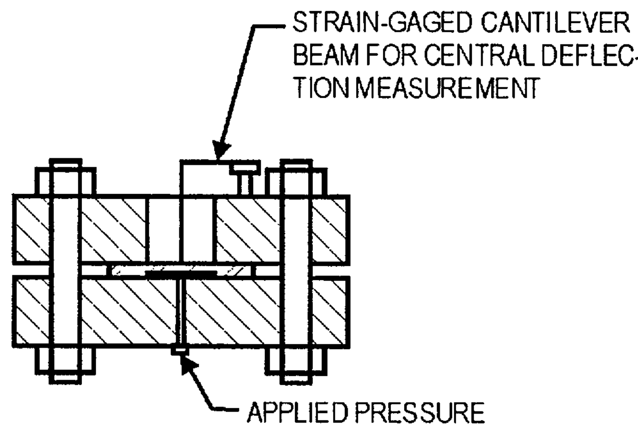
Analysis Cases and Results

Load Case	Predicted Pressure @ 11% Strain	Predicted FEM Instability Pressure
Original footprint with 0.297 in. thick clad (20.5 in^2)	5600 psi	>8000 psi
Original footprint with 0.24 in. thick clad (20.5 in^2)	4600 psi	>4800 psi
Enlarged footprint with 0.24 in. thick clad (self-similar) (41 in^2)	>2750 psi	>4000 psi

Stress-Strain Data for Type 308 Weld Metal

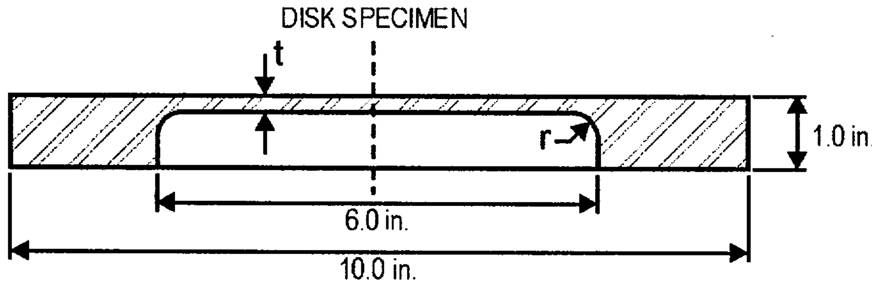
Reference	YS ksi	UTS ksi	Elong %	RA %	Matl Type
NUREG/CR-6235	20.8	62	38.4	70.8	Base
NUREG/CR-4538	22.2	67.3	39	70.8	Base
NUREG/CR-4538	22.8	68.8	40.5	70.8	Base
NUREG/CR-4687	20.1	65.2	53.8	71.3	Base
EPRI NP-4768	23.1	61.3	47	74	Base
EPRI NP-4768	24.8	62.6	45	70	Base
EPRI NP-4768	33.2	72.7	42	67	Base
ASME 72PVP12	34	84	54	75	Base
		Ave.Base	**45.0**	**71.2**	
EPRI NP-4668	44.8	62.9	22	46	SAW
EPRI NP-4768	36	61.8	25	67	SAW
EPRI NP-4768	40.8	70.3	25	69	SAW
NUREG/CR-6098	37.4	68	26.4		SAW
NUREG/CR-6389	49.1	68.1	30	46	SAW
NUREG/CR-6389	45	67.1	33	42.4	SAW
NUREG/CR-6389	54.3	74	15.5	63	SAW
NUREG/CR-6389	51.8	71.8	13.7	54	SAW
NUREG/CR-4878	471	67.6	31.5	44.2	SAW
NUREG/CR-4878	28.3	67.5	34.5	47	SAW-Ann
		Ave.SAW	**25.7**	**53.2**	
EPRI NP-4668	45.7	65.1	26	58	SMAW
EPRI NP-4768	46.8	61.4	37	48	SMAW
EPRI NP-4768	49.4	64.7	35	46	SMAW
NUREG/CR-4878	40.8	70.3	24.8	68.6	SMAW
		Ave.SMAW	**30.7**	**55.2**	
NUREG/CR-4538	44.3	65.4	33	74.3	Weld
NUREG/CR-4538	42.2	64.3	30	72.9	Weld
		Ave.SAW&SMAW	**27.3**	**53.8**	

PVRC Disk Burst Test Specimens



SCHEMATIC ILLUSTRATION OF TEST SETUP

DISK SPECIMEN



GEOMETRY	THICKNESS (t)	FILLET RADIUS ®
A	0.25 in.	0.375 in.
B	0.125 in.	0.125 in.
C	0.125 in.	0.375 in.

02055R0

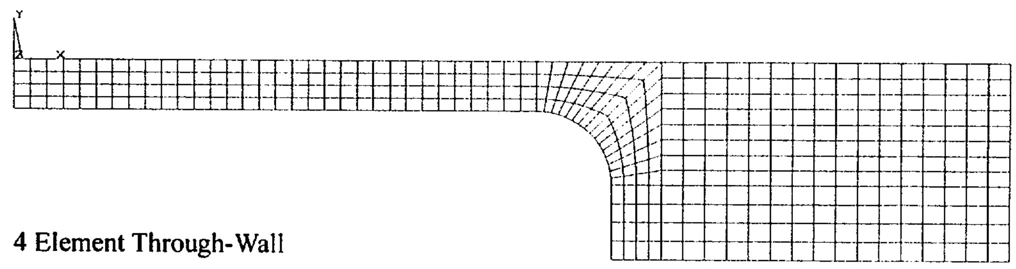
PVRC Disk Burst Test
Stainless Steel Material Properties

Modulus of Elasticity, E, e^6 psi:	28.3
Poisson's Ration, ν:	0.3

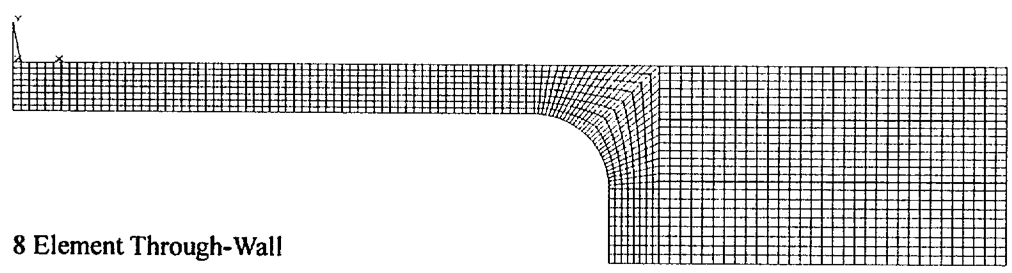
0.25 Y.S. (psi)	**S_{ult} (psi)**	**ε_{ult} (in/in)**	**Reduction In Area**	**A[1] (psi)**	**n[1]**
34,000	84,000	0.54	0.74	193,060	0.494

[1] Stress Strain Curve Assumed to be of form $\sigma = A (\varepsilon)^n$

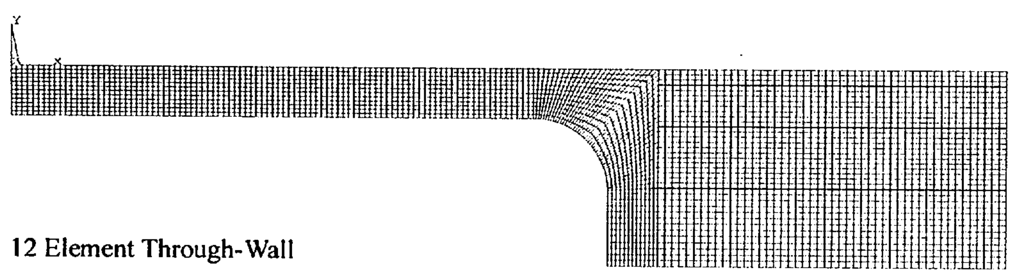
Axisymmetric FEMs of Disk Burst Specimens



4 Element Through-Wall

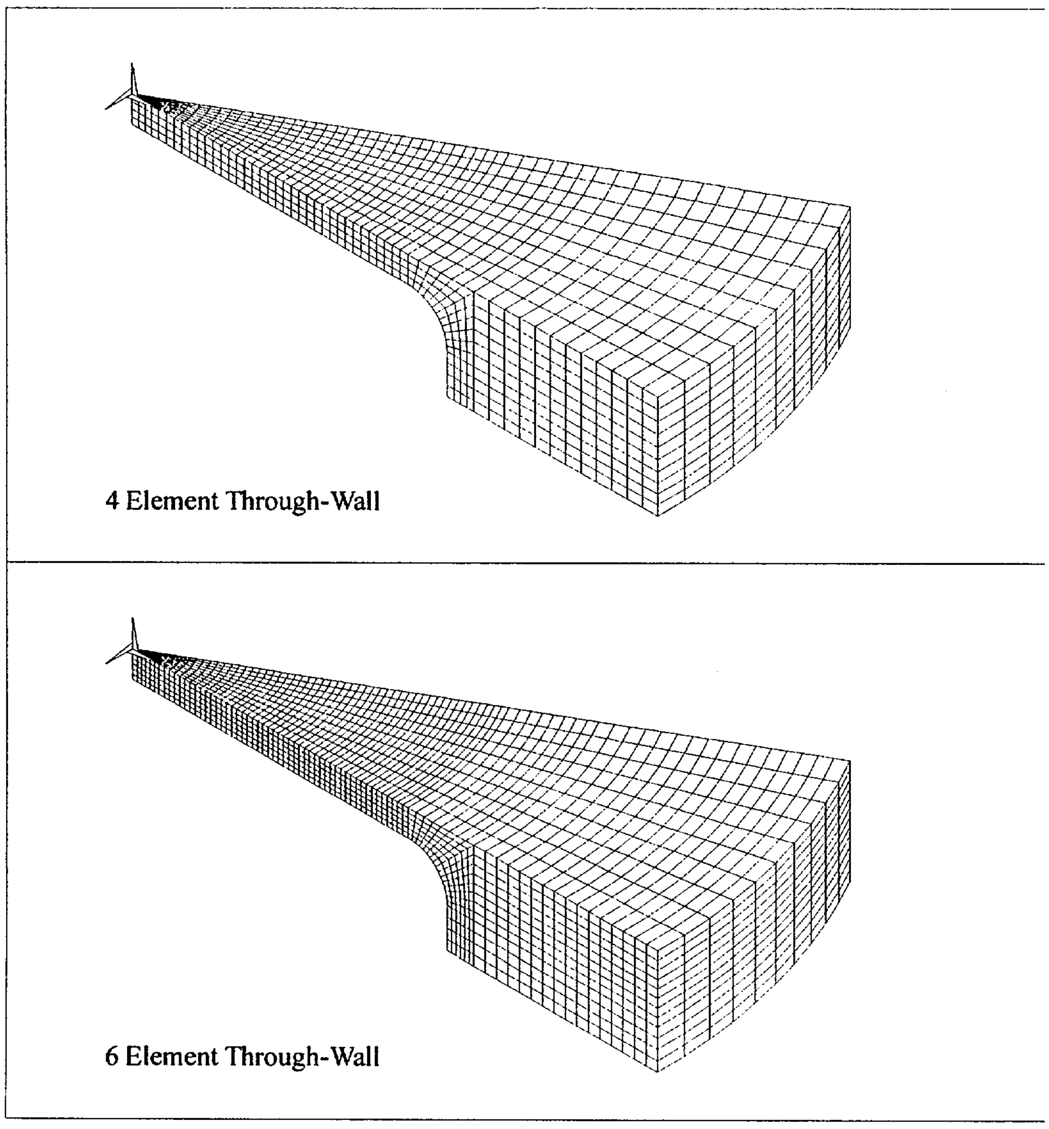


8 Element Through-Wall



12 Element Through-Wall

3-Dimensional FEMs of Disk Burst Specimens



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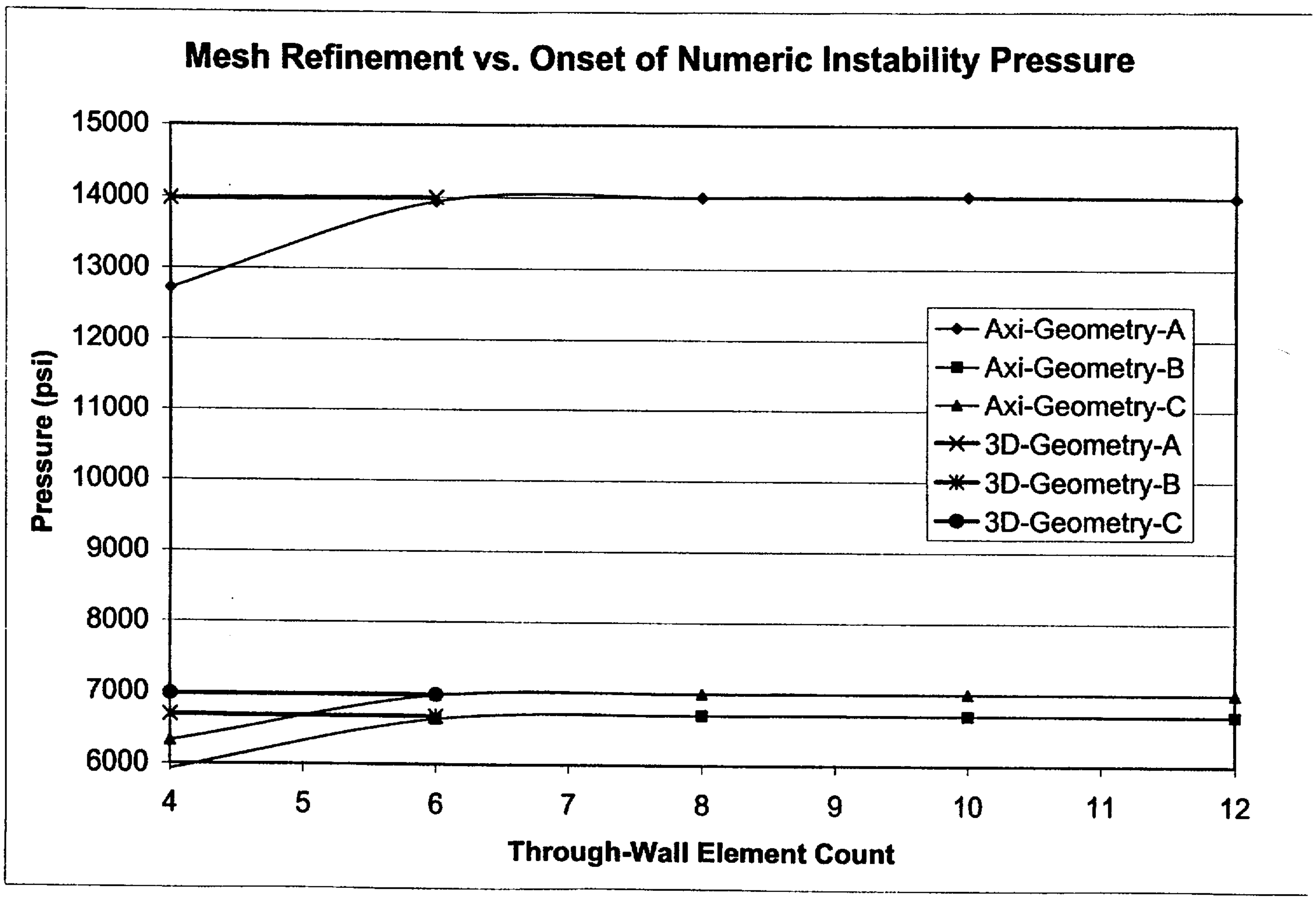
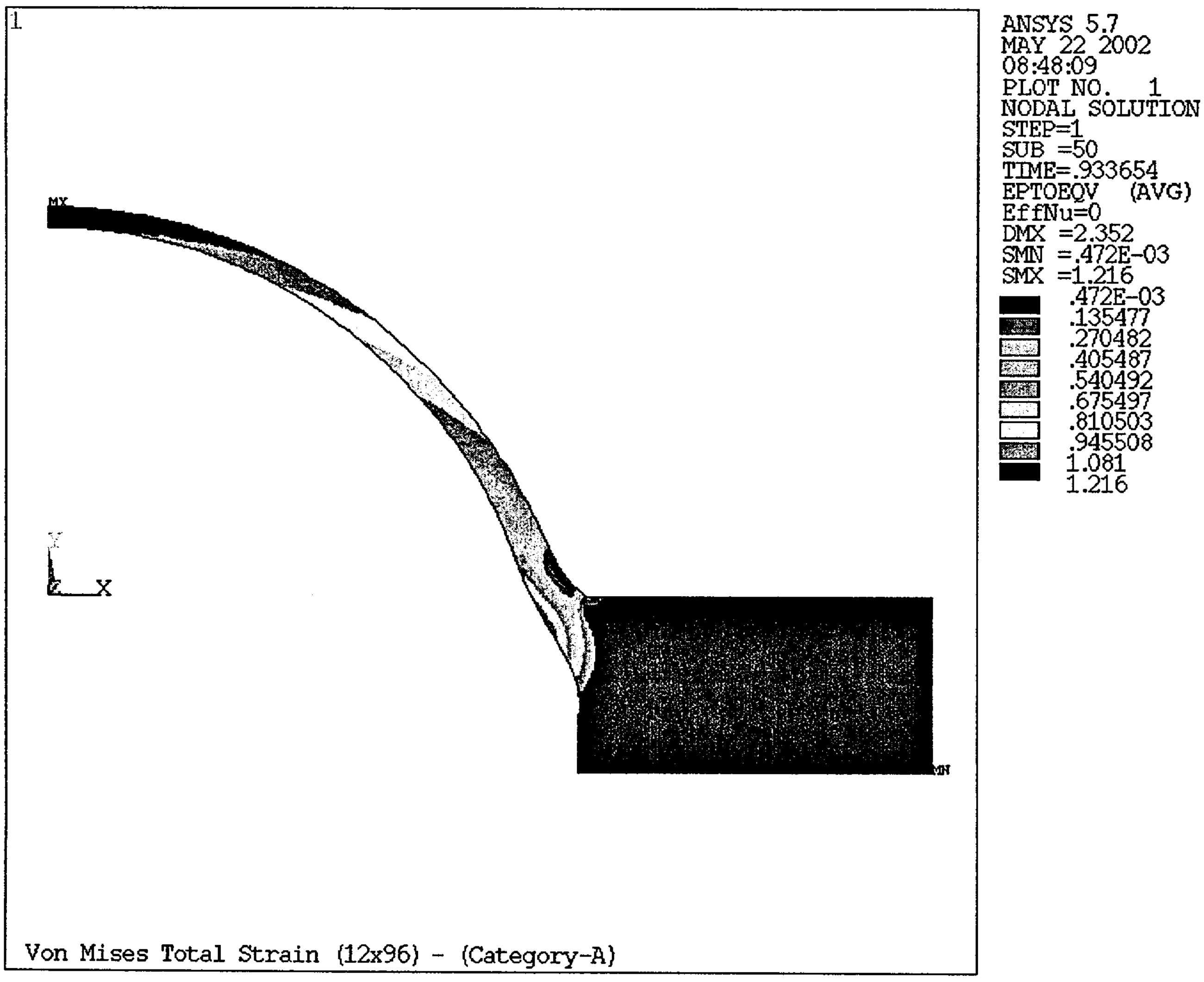
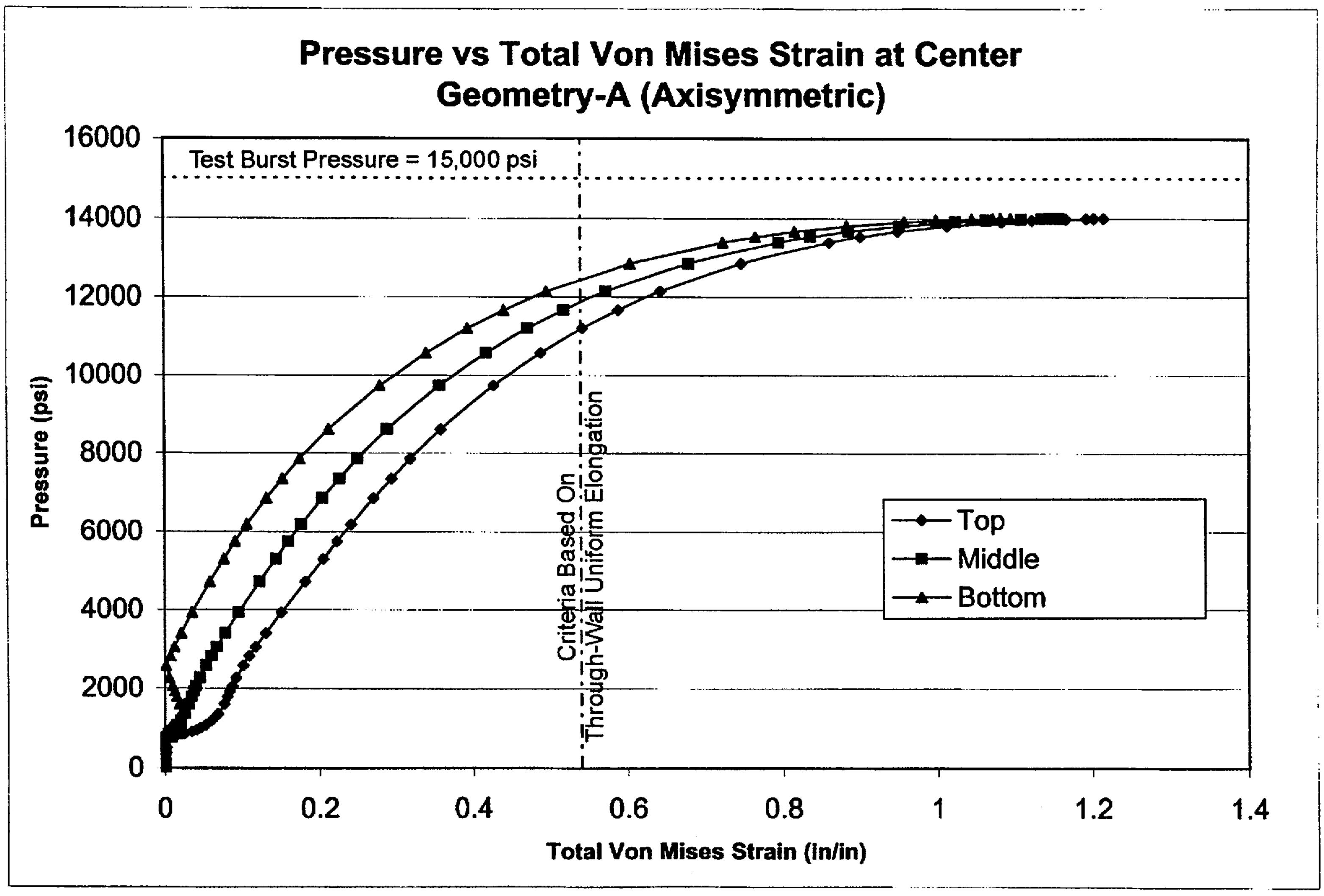
Demonstration of FEM Convergence on Disk Burst Specimens



Typical FEM Result on Disk Burst Specimen – Geometry A



Demonstration of Failure Criterion on Disk Burst Specimens



Failure Criteria Comparisons

Model Type	Model Geometry	Failure Pressure (psi)		
		Burst Test	Instability	Failure Criteria (Unif.Elong.)
Axisymmetric	A	15000	14005	~11000
Axisymmetric	B	6800	6694	~5500
Axisymmetric	C	7700	6997	~5750
3-Dimensional	A	15000	13997	~11000
3-Dimensional	B	6800	6671	~5500
3-Dimensional	C	7700	6974	~5750

Conclusion

The analysis procedure and failure criterion used in the Davis-Besse RPV head wastage evaluation is conservative compared with physical burst test results.