U.S. Nuclear Regulatory Commission Meeting
with Nuclear Energy Institute and Material Reliability Program

Thursday, June 7, 2001
9:00 a.m. - 12:00 noon
Commissioner's Hearing Room

Purpose: To discuss the Material Reliability Program's (MRP) interim report and response to NRC questions on control rod drive mechanism cracking, and potential NRC generic communications on this issue.

Success: Obtain information from NEI/MRP to determine the need for proceeding with a generic communication and the type and content of such. Inform NEI/MRP and other external stakeholders where the staff is in the regulatory process.

Introduction:	Jake Zimmerman, NRC	9:00 - 9:10 a.m.
Opening Remarks:	Brian Sheron, NRC	9:10 - 9:25 a.m.
Discussion of MRP Interim Report and Response to NRC Questions:	Larry Mathews, MRP	9:25 - 10:15 a.m.
- Break -		10:15 - 10:25 a.m.
Cont. Discussion of MRP Interim Report and Response to NRC Questions:	Larry Mathews, MRP	10:25 - 11:45 a.m.
Closing Comments:	NRC/MRP/NEI	11:45 a.m. - 12:00 noon

Additional information on Generic Activities on PWR Alloy-600 Weld Cracking may be found on the NRC web site at http://www.nrc.gov/NRC/REACTOR/ALLOY-600/index.html.

The NRC staff will be available immediately following the meeting to speak with members of the public.

ATTACHMENT 1

NRC Meeting with Nuclear Energy Institute and Material Reliability Program on Control Rod Drive Mechanism Cracking Issue

Thursday, June 7, 2001
9:00 a.m. - 12:00 p.m.
Commissioner's Hearing Room

Name	Organization/Title	Phone Number/Email
Jake Zimmerman	NRC/NRR/DLPM - Lead Project Manager	(301) 415-2426, jiz@nrc.gov
Brian Sheron	NRC/NRR/ADPT - Associate Director	(301) 415-1274, bws@nrc.gov
Jack Strosnider	NRC/NRR/DE - Division Director	(301) 415-3298, jrs2@nrc.gov
Keith Wichman	NRC/NRR/DE/EMCB - Section Chief	(301) 415-2757, krw@nrc.gov
Allen Hiser	NRC/NRR/DE/EMCB	(301) 415-1034, alh1@nrc.gov
Mark Reinhart	NRC/NRR/DSSA/SPSB - Acting Branch Chief	(301) 415-1185, fmr@nrc.gov
Gene Carpenter	NRC/NRR/DE/EMCB	(301) 415-2169, cec@nrc.gov
Roger Huston	Licensing Support Services / Principal	703-671-9738, Roger@licensingsupport.com
Larry Mathews	Southern Nuclear, Mgr. ITS	(205) 992-7729, lkmathews@southernco.com
Altheia Wyche	SERCH Licensing / Bechtel	(301) 228-6401, awyche@bechtel.com
Dennis Weakland	FENOC	724-682-5958 dweakland@firstenergycorp.com
Chuck Rice	SCE&G	(803) 345-4491 crice@scana.com
D. Labott	PSEG	856-339-1094
Eric Schoonover	SCE	949-368-2234 schoonej@songs.sce.com
Martin Murphy	CCNPPI	410 495 2544 martin.c.murphy@ccnppi.com
Jim Bennetch	Dominion Generation	(804)-273-3169/Jim_Bennetch@dom.com
David Lounsbury	PSEG Nuclear	856-339-3906
Lee Abramson	NRC/RES/PRAB	(301)415-6180, LXA@NRC.GOV
George Rombold	AmerGen	610-765-5516 george.rombold@exeloncorp.com
Robert Hermann	SIA	540-710-6717 RHermann@adelphia
Steve Fyfitch	FRA-ANP	412-264-1610/SFyfitch@framatech.com
Karl Haslinger	Westinghouse	860-285-2606/Karl.H.Haslinger@us.westinghouse.com

ATTACHMENT 2

NRC Meeting with Nuclear Energy Institute and Material Reliability Program on Control Rod Drive Mechanism Cracking Issue

Thursday, June 7, 2001
9:00 a.m. - 12:00 p.m.
Commissioner's Hearing Room

Name	Organization/Title	Phone Number/Email
Meggalen W. Weston	ACRS	415-3151 mww@nrc.gov
JIM CHUNG	NRC	
L.N. OLSHAN	NRC	415-1419
RALPH LANDRY	NRC	415-1140
Ramin Assa	NRC	415-8206
Michelle Snell	NRC	415 1840
Dwight Snowberger	NRC	415-2007
Bob [illegible]	NRC	415-8200
Jim Clifford	NRC	415-1430
Les Cupidon	NRC	415-6366
Matthew L. Mitchell	NRC	415-3303
Jay Collins	NRC	415-1038
Bill Bateman	NRC	
FAROUK ELTAWILA	NRC	415-5741
MOTOHISA FUJITA	KANSAI ELECTRIC POWER	202-659-1138
JAMES MEDOFF	NRC	301-415-2715, jxm@nrc.gov
Bill Campbell	Entergy Operations, Inc.	601-368-5307 wcamp63@entergy.com
Bill Gray	Framatome ANP	804-832-2783
A MARION	NEI	2027398080
Ross Telson	NRC	415-1175
Glenn White	Dominion Engineering, Inc.	703-790-5544; gwhite@domeng.com
Steve Hunt	Dominion Engineering, Inc.	703-790-5544, shunt@domeng.com

Name	Organization/Title	Phone Number/Email
John F. Hall	W	860-285-4762 john.f.hall@us.westinghouse.com
DAVID J. AYRES	W 8602854187	david.j.ayres@us.westinghouse.com
L. B. Marsh	NRC	LBM@NRC.GOV
TOM HARRISON	McGRAW-HILL	202-383-2165 THARR@MH.COM
Cayetano Santos	NRC/RES/DET/MEB	301-415-6004 CXS3@NRC.GOV
AVTAR SINGH	EPRI	650-855-2384/avsingh@epri.com

MRP RPV Penetration Interim Safety Assessment

Presentation
June 7, 2001

MRP RPV Penetration Interim Safety Assessment - Meeting Agenda

- Introductory Remarks
- Background
- Response to NRC Questions
  - Leakage Detection
  - Time-Temperature Histogram
  - Circumferential Crack Growth
  - Loose Parts
  - Risk Assessment
- Conclusions
- Future Actions
- NRC Feedback

Background

- Leaks detected from CRDM nozzles in four B&W design plants (also leaks from several Oconee 1 T/C nozzles)
- Interim Safety Assessment submitted May 18, 2001
  - The three Oconee units and ANO-1 are among the lead units in the US based on time at temperature
  - Leaks were found by routine visual inspections while the nozzles and welds were well within required structural margins
  - Leakage should also be detectable in other plants
  - Several other lead units with long operating times and high head temperatures have performed inspections from above and below the head without any significant findings
  - A CRDM nozzle ejection is an analyzed event in plant FSARs
- NRC identified several questions on May 25, 2001

NRC Questions

- Leak detection (Section 3.0)
  - Effect of initial interference fit on leak detection
- Time-temperature histogram (Section 4.0)
  - Effect of activation energy on predictions
  - Benchmarking against foreign plant inspections
  - Ten year inspection criterion
- Growth rate of circumferential cracks (Sections 4.0 & 5.0)
  - Time until Oconee 3 would have reached allowable flaw size
  - Effect of crack growth rates on histogram
- Loose parts (Additional Questions)
- Risk assessment (Additional Questions)

Safety Assessment Status

- The Interim Safety Assessment was prepared to demonstrate safety of operating plants
- Additional effort is ongoing in several areas
  - Analysis associated with the Final Safety Assessment
  - Visual inspections of the reactor vessel top head surface for all plants coming down for Fall 2001 refueling outages
  - Research into improved inspection and repair technology
  - Risk assessment
- Results will be factored into the Final Safety Assessment

Response to Questions Related to Leakage Detection

- Issues to be discussed
  - ONS and ANO-1 leakage "bounding" for industry (3-1)
  - Actual interference fits at ONS and ANO-1 (3-2)
  - Effect of operating conditions on initial interference (3-3)
  - Effect of nozzle ovalization on fit (3-3)
  - Effect of vessel flange rotation on fit (3-3)
  - Leakage modeling (3-4)
  - Gap plugging (3-5)

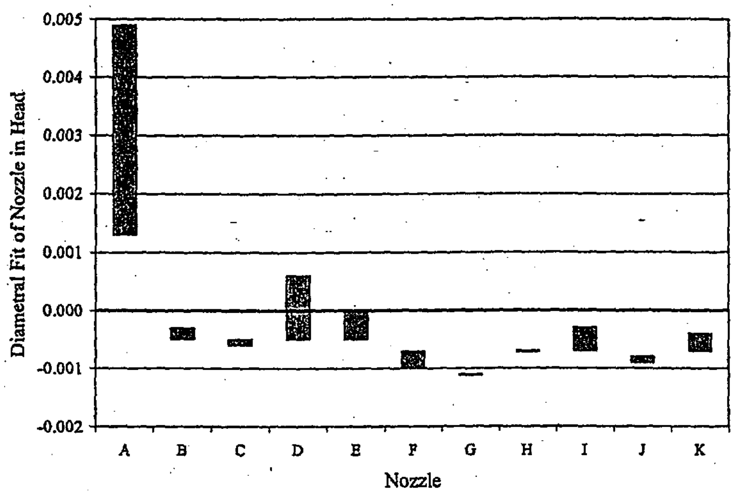
Leakage Detection
"Bounding Leakage"

- Oconee and ANO-1 detected leakage, but
  - Some other plants have greater interference fits (see Table 3-2 of Interim Safety Assessment)
- Leakage should be detectable at most other penetrations given similar cracks
  - No significant cracking was found in 26 additional "non-leaking" Oconee 1 and 3 CRDM nozzles. This is consistent with all through-wall cracks in CRDM nozzles leading to observable leaks
  - Interference fits at other plants are only slightly larger than Oconee and ANO-1
  - It is difficult to prevent leakage of 2,250 psi water without roll, hydraulic or explosive expansion or use of a sealant

Leakage Detection
Actual Fits at Oconee 1/3 and ANO-1

- Fabrication records for Oconee 1, Oconee 3 and ANO-1 vessel heads have been reviewed
- The following measurements were taken
  - ID of the hole in the vessel head at the top and bottom of the interference fit region
  - OD of the nozzle
- Results for the eleven leaking CRDM nozzles at Oconee 1, Oconee 3 and ANO-1 are shown on next slide
  - One nozzle had a clearance fit (gap)
  - Ten nozzles had at least one end within the specified diametral interference range of 0.0005 - 0.0015 inches. The average of the 20 measurement points for these nozzles was a 0.0006 inch diametral interference

Leakage Detection
Actual Fits at Oconee 1/3 and ANO-1



EPRI

Leakage Detection
Effect of Operating Temperature on Fit

- The hole diameter in the vessel head and outside diameter of the nozzle both increase with temperature
- The close match in thermal expansion to low-alloy steel was one of the reasons that Alloy 600 was originally selected for these applications
- Predictions depend on coefficient of thermal expansion given in the Code
  - Current Code - No differential expansion between Alloy 600 nozzle and vessel head
  - Earlier Codes - About 1 mil additional diametral interference from temperature effect

EPRI

Leakage Detection
Effect of Operating Pressure on Fit

- The hole diameter in the vessel head and outside diameter of the nozzle both increase with internal pressure
- For example:
  - The Alloy 600 nozzle is a capped thick wall cylinder. For the typical case of 4.00" OD, 2.75" ID, 2250 psi internal pressure, and modulus of elasticity of 28.7 x10^6 psi at 600°F, the radial expansion is
    - ΔR_{nozzle} = 0.00024"
  - The low-alloy steel head is a sphere with a hole for the nozzle. For the case of 172" ID, 7" wall thickness, 2250 psi internal pressure, modulus of elasticity of 26.4 x10^6 psi at 600°F, and a stress concentration factor of 2 for a hole in a plate subjected to biaxial stresses, the radial expansion at the hole is
    - ΔR_{head} = 0.00201"

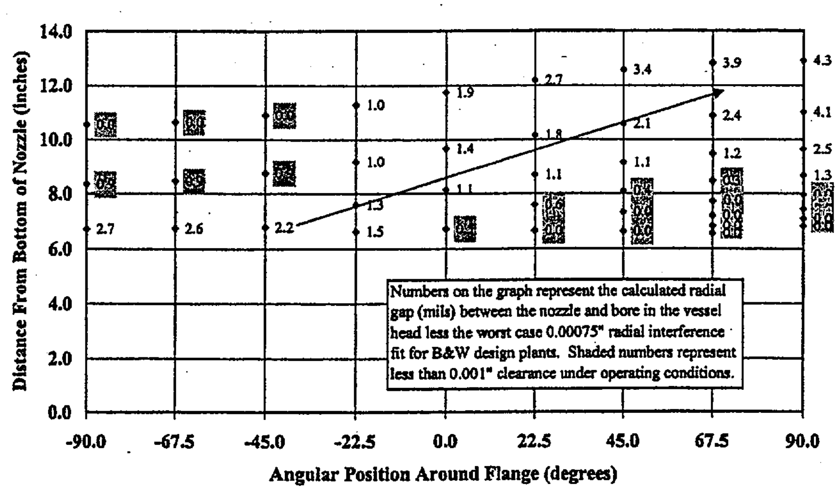
Leakage Detection
Effect of Operating Conditions on Fit

- Since temperature has only a small effect on the initial interference fit, the change in fit under operating conditions is primarily due to dilation of the vessel head under internal pressure
- For the example, the change in diametral fit due to pressure is approximately
  - ΔD = 2(0.00201" - 0.00024") = 0.0035"
  - The hole will open up further when the effect of reduced effective modulus due to the effect of multiple nozzles is considered
- Annular gaps are expected to open up for most nozzles under operating conditions

Leakage Detection
Effect of Nozzle Ovalization on Fit

- Finite element analyses show that outer row CRDM nozzles displace laterally and become slightly ovalized in the vessel head as a clearance opens up under operating conditions
- The displacement and ovalization reduce the leak path at some locations and increase the leak path at other locations
- Results for a typical CRDM nozzle are given in next slide
- The net effect is to create a spiral flow path which has less resistance than a uniform annular gap

Leakage Detection
Effect of Nozzle Ovalization on Fit



Leakage Detection
Effect of Vessel Flange Rotation on Fit

- Previous finite element analyses show a minor (<20%) effect of flange tensioning and rotation on the ovality of the peripheral nozzles
- The effect is small

Leakage Detection
Leakage Modeling

- Leak rates from head penetration SCC are very small
- Modeling of leak rates for these cracks and geometries would be very complex and may not be productive

Leakage Detection
Gap Plugging

- Experience shows that gap plugging from boron or corrosion deposits did not prevent detection of leakage at Oconee and ANO-1
  - Observed for a range from cold gap to 1.1 mil interference fit
- Boron not expected to plug gaps because liquid behind plug should dissolve boron
  - Field experience (valve body to bonnet or packing leaks) has shown accumulation on outside
- Corrosion requires oxygen (see *Boric Acid Corrosion Guidebook*, EPRI TR-104748)
  - If oxygen can migrate into the annulus, then water can leak out, especially given a 2,250 psi driving pressure

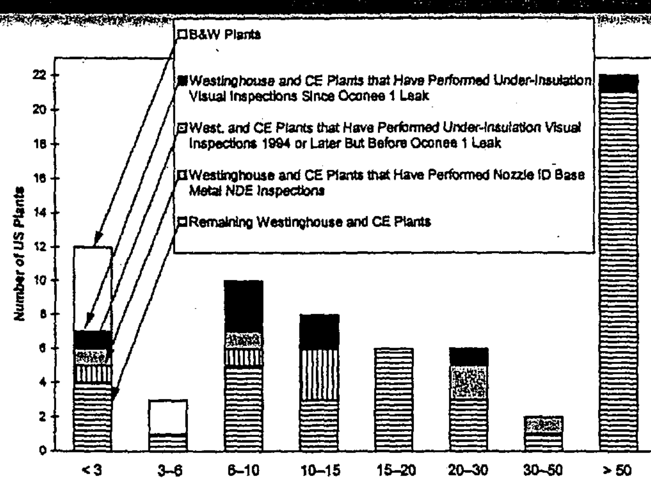
Response to Questions Related to Time-Temperature Histogram

- Issues to be discussed
  - Effect of activation energy on predictions (4-1)
  - Benchmarking against inspections of foreign plants (4-3)
  - Basis for 10 year inspection criterion (4-4)
  - Basis for not inspecting plants in top group that have previously performed some inspections (4-5)

Time-Temperature Histogram
Background

- The time-temperature model groups plants according to the time (EFPY) required for each unit to reach the equivalent effective time at temperature as Oconee 3 at the time the above-weld circumferential cracks were discovered in February 2001
- The reference date for the time-temperature assessments is March 1, 2001
- The industry standard activation energy of 50 kcal/mole for PWSCC initiation in Alloy 600 material was used to normalize plant operating time to a head temperature of 600°F

Time-Temperature Histogram
Chart in Interim Safety Assessment



Effective Full Power Years (EFPYs) from 3/1/2001 until Reaching Oconee 3 EFPYs at Time of its Spring 2001 Outage, Normalized For Differences in RV Head Temperature Using the Arrhenius Relationship and a Standard Activation Energy of 50 kcal/mole

Time-Temperature Histogram
Table in Interim Safety Assessment

Activation Energy = 50 kcal/mole

Status		Assessment Groups							
		< 3 EFPYs	3-6 EFPYs	6-10 EFPYs	10-15 EFPYs	15-20 EFPYs	20-30 EFPYs	30-50 EFPYs	> 50 EFPYs
B&W Plants		5	2	0	0	0	0	0	0
West./CE Plants	Head Surface Visual[1,3]	1 (1)	0	3 (1)	2 (1)	0	1 (1)	0	1 (0)
	Head Surface Visual[2,3]	1 (0)	0	1 (0)	0	0	2 (2)	1 (1)	0
	Nozzle ID NDE	1	0	1	3	0	0	0	0
	Remaining Plants	4	1	5	3	6	3	1	21
Totals		12	3	10	8	6	6	2	22

NOTES:
[1]Visual inspection of top surface of head under insulation since Oconee 1 CRDM nozzle leak in December 2000
[2]Visual inspection of top surface of head under insulation 1994 or later but before Oconee 1 leak
[3]The number in parenthesis indicates the number of plants that have performed complete under-insulation visual inspections of all nozzles. The remaining under-insulation visual inspections were of between 3% and 34% of the CRDM, CEDM and ICI nozzles.

Time-Temperature Histogram
Effect of Activation Energy

- 50 kcal/mole is a best-estimate value for PWSCC initiation in Alloy 600 (e.g., EPRI NP-7493) for temperature range of interest (550-610°F; 285-320°C)
- EDF has published a lower experimentally determined activation energy for PWSCC initiation in Alloy 600 reactor head nozzle material for the temperature range of interest (44 kcal/mole)

Time-Temperature Histogram
Effect of Activation Energy (cont.)

- A sensitivity study for the results of the plant assessments was performed
- The effect is small, as shown below:

Activation Energy	*Assessment Groups*							
	< 3 EFPYs	3–6 EFPYs	6–10 EFPYs	10–15 EFPYs	15–20 EFPYs	20–30 EFPYs	30–50 EFPYs	> 50 EFPYs
50 kcal/mole	**12**	**3**	**10**	**8**	**6**	**6**	**2**	**22**
40 kcal/mole	**12**	**4**	**14**	**9**	**4**	**3**	**2**	**21**

Time-Temperature Histogram
Benchmarking Against Foreign Plants

- The new time-temperature histogram has not been benchmarked against foreign plants
- It is not a predictive model, but a way to rank US plants relative to Oconee 3

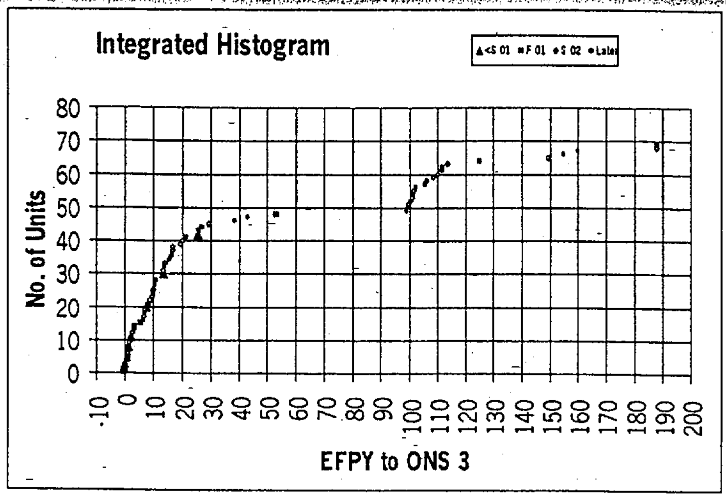
Time-Temperature Histogram
Ten-Year Period

- 10 Year Period for Near-Term Inspection
  - The ten year period for recommending visual inspections of the top of the vessel head for small amounts of leakage similar to that observed at Oconee and ANO-1 was selected to provide some margin for uncertainties
  - The ten year period will be re-assessed based on results of upcoming outages

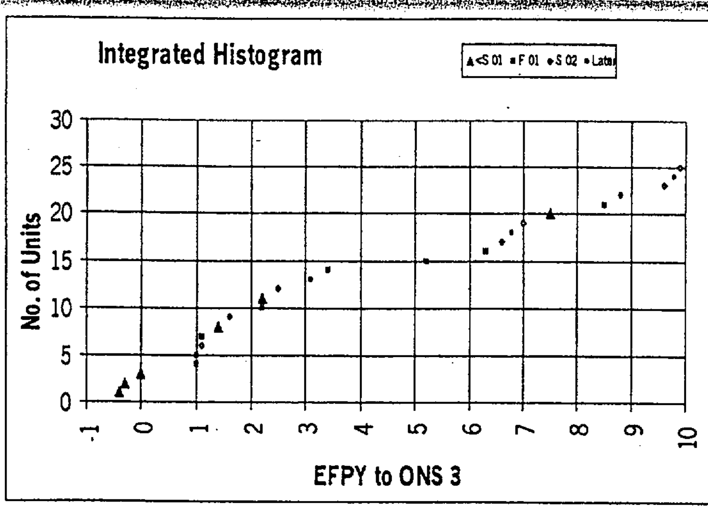
Time-Temperature Histogram
Reinspections

- The recommendations for top of the head visual inspections apply regardless of any previously performed inspections from the top or bottom of the head

Time-Temperature Histogram



Time-Temperature Histogram



Response to Questions Related to Circumferential Crack Growth

- Issues to be discussed
  - Crack growth rates in annulus environment after leak (4-5)
  - Time until Oconee 3 cracks would have reached critical flaw size (5-1)
  - Effect of crack growth rates on histogram (4-5)

Circumferential Crack Growth
Growth Rate in Annulus Environment

- Data are available from 5 sources for carefully controlled PWSCC tests of Alloy 600 and 182, using PWR conditions
- OD initiated cracking requires the presence of water or steam, so a pressure boundary leak is necessary
- The crevice region could contain some Oxygen from the containment atmosphere, but at temperature this Oxygen would be quickly consumed by reaction with the low alloy steel nearby
- This reaction, plus the extremely tight fit and the distance to the OD of the head, make a high Oxygen environment unlikely

Circumferential Crack Growth
Growth Rate in Annulus Environment

- Since the fluid will contain lithium hydroxide and boric acid, it will likely be similar to a controlled PWR environment
- Comparison of BWR and PWR crack growth rates for Alloy 600 and 182 shows that, at a given temperature, the growth rates are comparable
- Temperature is a stronger variable than environment for these materials

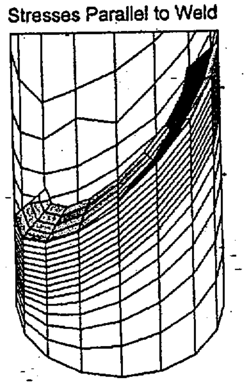
Circumferential Crack Growth
Margin for Oconee 3 Cracks

- Two Oconee 3 nozzles were cracked approximately 165°
- Stress analyses show that cracks initiated in a high stress region and propagated into a lower stress region (see next slides)
- The remaining time for Oconee 3 circ cracks to reach the allowable ligament was estimated to be 4 years, based on the modified Peter Scott model
- Efforts are underway to better define the stresses and stress intensities in the nozzle in the intact and cracked conditions

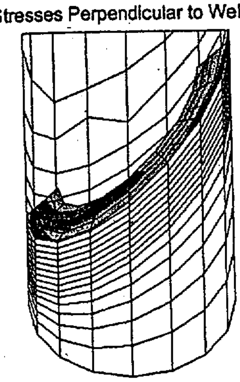
Circumferential Crack Growth
Intact Operating Condition Stresses

- Finite element analyses show stresses are highest at uphill side of peripheral nozzles where deep cracks were discovered

Stresses Parallel to Weld



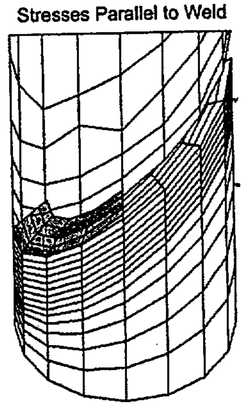
Stresses Perpendicular to Weld



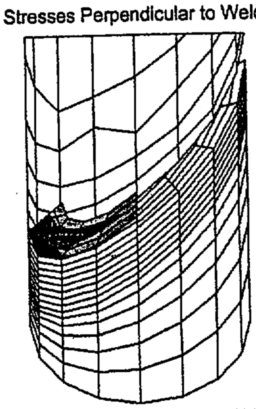
Circumferential Crack Growth
Stresses With 180° Through-Wall Crack

- With 180° crack, stresses are still low relative to uphill stresses in intact condition

Stresses Parallel to Weld



Stresses Perpendicular to Weld



Circumferential Crack Growth
Growth Rate Assumed for Histogram

- No crack growth was assumed for the histogram
- Top head visual inspections will be performed to detect leakage
- Oconee is at high end of time and temperature rankings
- Oconee 3 still had margin to Code allowables
- 10 years was picked to reasonably envelope uncertainties for near term recommendations

Loose Parts

- The potential for, and consequences of, loose parts in B&W designed plants such as Oconee and ANO-1 was described to the NRC on April 12, 2001
- While analyses for other plant designs have not been completed, results are expected to be similar
- Loose parts analyses will be included in final report

Risk Assessment
Background

- Risk calculations are in the planning stages
- Answering this question will require expertise from materials, LOCA and transient analysis, fuels, licensing, and risk
- The effort will include interaction with all PWR Owners groups, to ensure applicability to all plants
- One approach is to perform a qualitative risk analysis, following the same process successfully used for the Part Length CRDM issue

Risk Assessment
Accident Progression

- Failure of a CRDM housing is the same as a hot leg LOCA
  - Maximum break size for one housing is 4 inches
  - Failure of multiple housings would just be a larger LOCA
- LOCA behavior is bounded by design basis analyses
  - RCS will depressurize resulting in ECCS actuation
  - ECCS will inject borated water to maintain core cooling
  - Core internals will remain in a coolable geometry
- Control rod ejection may cause reactivity excursion
  - Rods are typically withdrawn during power operation, thus no substantial power excursion expected due to limited rod worth
  - Rod ejection during startup could result in a reactivity excursion that would be turned around by doppler and insertion of other control rods
    - Injection of borated water by the ECCS will assure core shutdown

Risk Assessment
Plant Response and Operator Actions

- Failure of a CRDM housing is the same as a hot leg LOCA
  - RCS will depressurize resulting in ECCS actuation
  - ECCS will inject borated water to maintain core cooling and assure core shutdown
  - Existing EOPs direct the operators to monitor plant critical safety functions to assure core shutdown and core cooling
- Operator actions for failure of multiple CRDM housings would be the same as a single housing failure
  - Monitoring plant critical safety functions provides operators direction and prioritized actions in case of multiple events, subsequent failures or consequential failures such as multiple rod ejection or ATWS
  - Control rod insertion and boration are used for reactivity control
  - ECCS injection is used for core cooling and inventory control

Risk Assessment
Secondary Effects

- Failure of a CRDM housing is the same as a hot leg LOCA
  - Secondary effect concerns would not differ from DBA LOCAs
  - Issues such as insulation blockage of the recirculation sump are not unique to failure of a CRDM housing and are being addressed outside of this issue
  - Operator actions described earlier will not change if damage to additional CRDMs occurs
  - EOPs for the operators are symptom based and do not require event identification. The EOPs provide operators with direction and prioritized actions in case of multiple events, subsequent failures or consequential failures

Risk Assessment
Conclusions

- Failure of a CRDM housing is the same as a hot leg LOCA
- Failure of multiple housings would just result in a larger LOCA
- LOCA behavior is bounded by design basis analyses
- Existing EOPs are symptom based, do not require event identification and direct the operators to monitor plant critical safety functions to assure core shutdown and core cooling
- Existing EOPs provide adequate directions for the operators to mitigate a transient induced by one or more CRDM penetration failures

RPV Penetration Summary

- Interim Assessment Provided to Staff May 18, 2001
  - Update of previous submittals to reflect recent events
    - Safety Assessments in 1993/4
    - Generic Letter 97-01 in 1997
- Conclusions
  - Axial PWSCC in CRDM nozzles does not impact plant safety
    - Bounded by previously submitted Safety Assessments (1993/94)
  - Reasonable assurance that other PWRs do not have circumferential cracking that would exceed structural margin
    - Oconee and ANO-1 in highest grouping based on effective time-at-temperature
    - Leaks discovered by routine visual inspection of top head surface
    - Leaks discovered with significant structural margin remaining
    - Several other plants in highest groupings have no evidence of leakage

Schedule

- The following schedule is planned
  - MRP Workshop - 6/13-14/01, Atlanta
  - Revised Inspection Recommendations - 6/30/01
  - Expert Panel on Crack Growth - First Meeting 7/01
  - Final Butt-Weld Safety Assessment - ~9/01
  - Inspections during Fall 2001 outages
  - Final RPV Penetration Safety Assessment - 12/01
  - Safety Assessment for all Alloy 600, 82 and 182 Locations - 12/01