September 7, 2001

NOTE TO: Docket File

FROM: David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: HANDOUTS FOR MEETING WITH DUKE ENERGY CORPORATION (TAC Nos. MB2643, MB2644, AND MB 2645)

On September 7, 2001, the NRC staff participated in an open meeting with Duke Energy Corporation to discuss the Oconee Nuclear Station response to Bulletin 2001-01, *Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles.*

The handouts for that meeting are attached.

Docket Nos. 50-269, 50-270 and 50-287

NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetrations

Oconee Nuclear Station Response

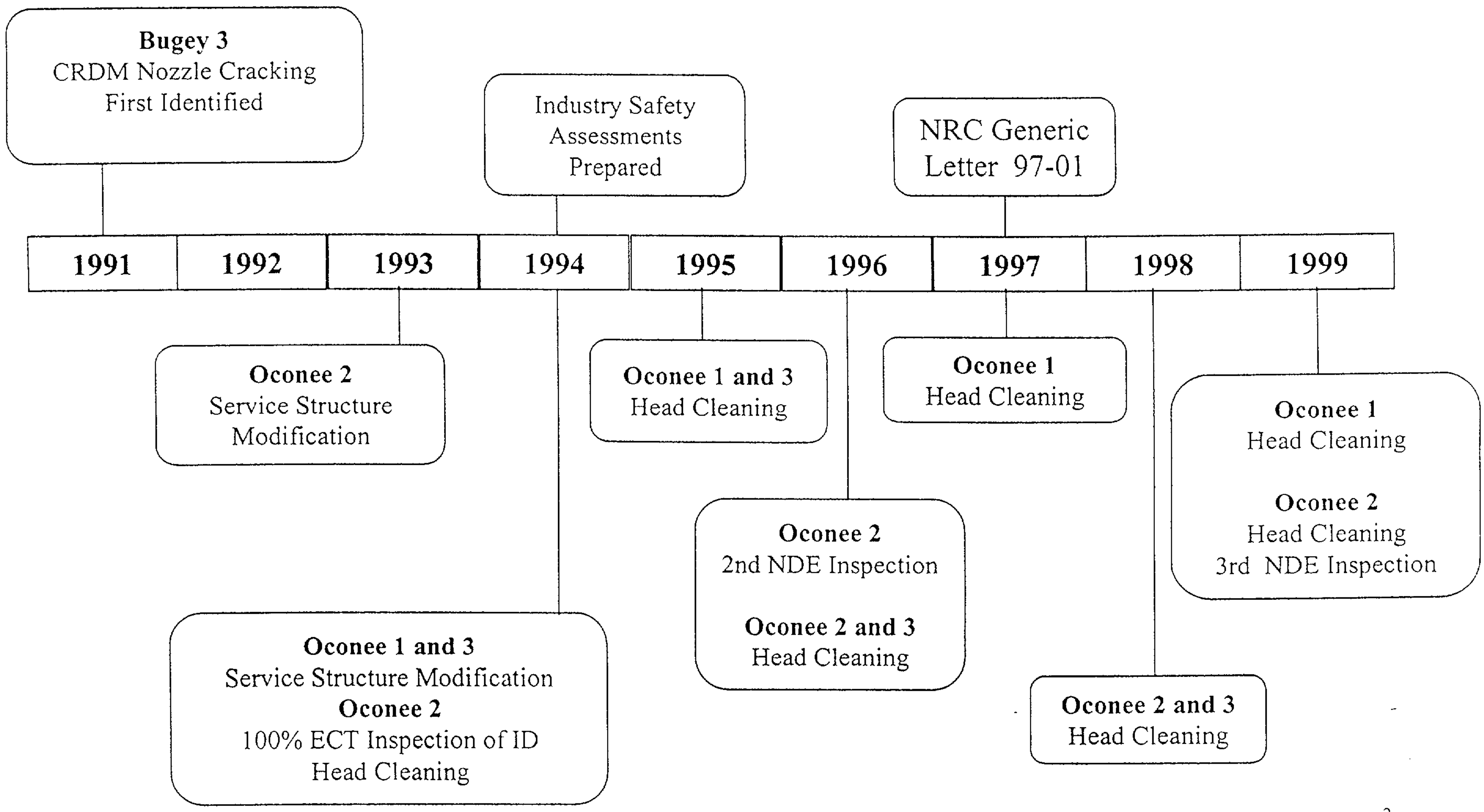
Bill McCollum, Duke Energy
Mike Robinson, Duke Energy
Stanley Levinson, Framatome ANP, Inc.

Office of Nuclear Reactor Regulation
Washington, DC
September 7, 2001

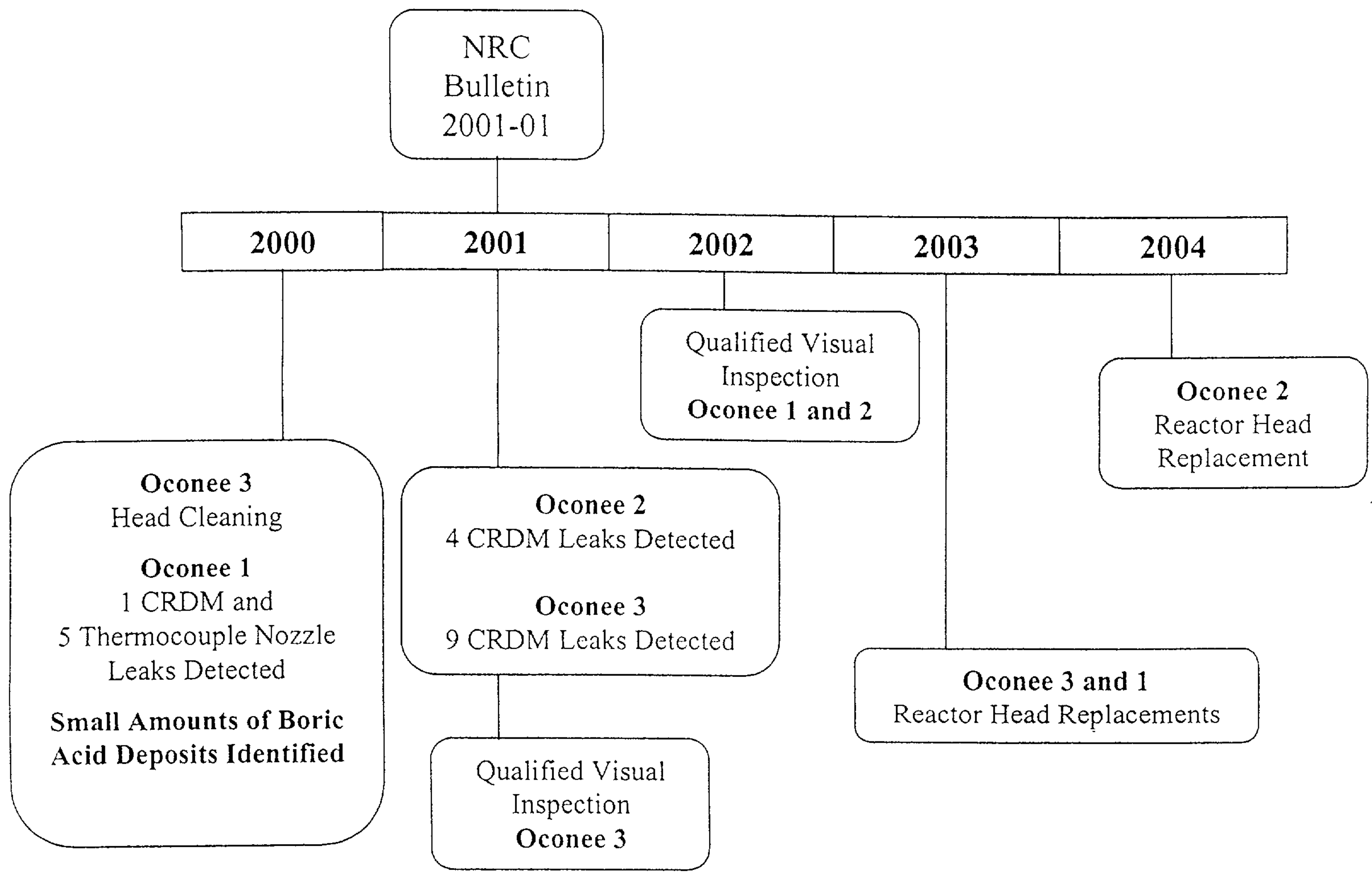
Meeting Agenda

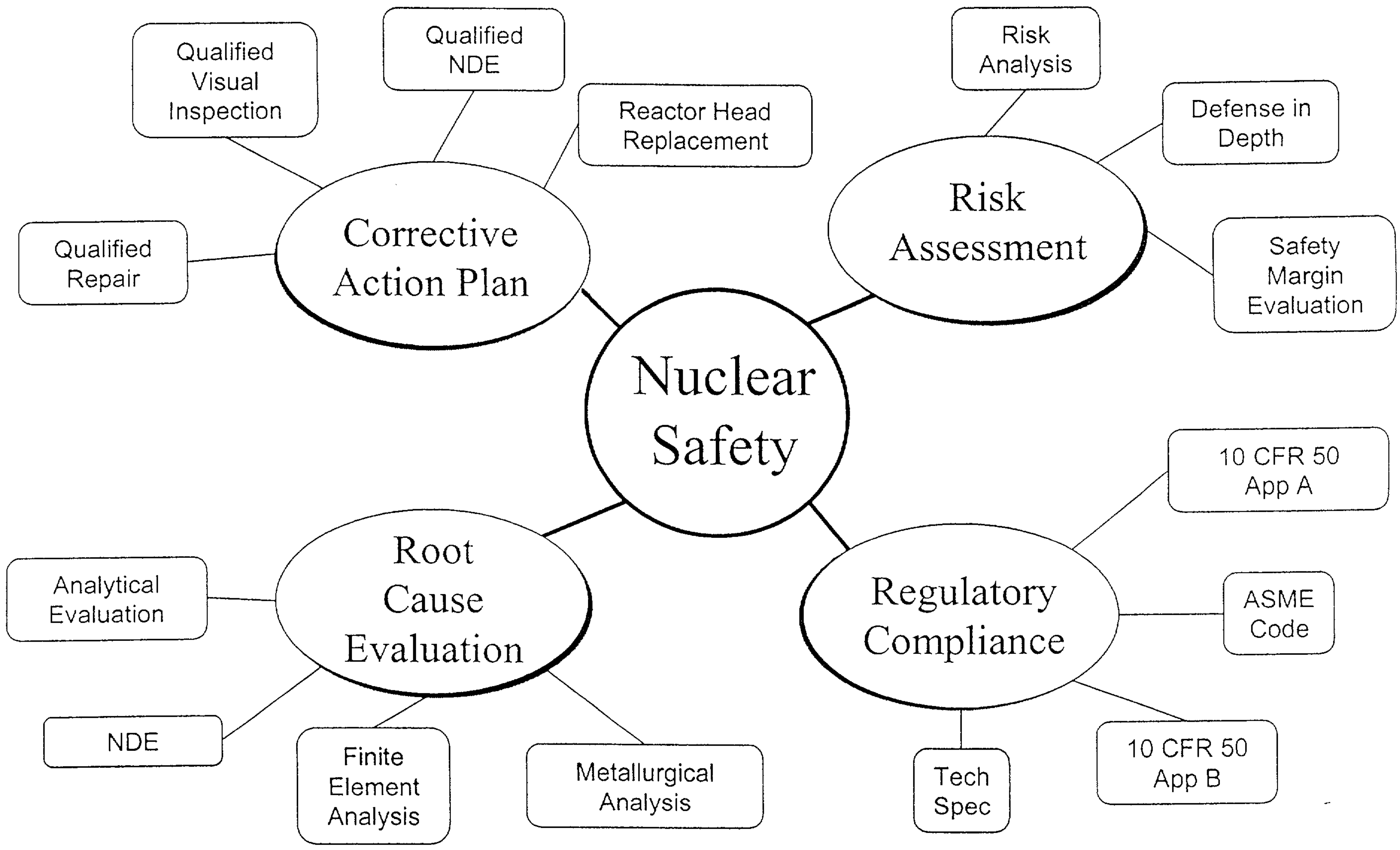
♦ Opening Comments	Bill McCollum
♦ Background	Mike Robinson
♦ Generic Bulletin 2001-01 Basis	Mike Robinson
» Root Cause Evaluation	
» Corrective Action Program	
» Risk Assessment	Stanley Levinson
» Regulatory Compliance	Bill McCollum
♦ Conclusion	Bill McCollum

Background


Bugey 3
CRDM Nozzle Cracking
First Identified
Industry Safety
Assessments
Prepared
NRC Generic
Letter 97-01
1991
1992
1993
1994
1995
1996
1997
1998
1999
Oconee 2
Service Structure
Modification
Oconee 1 and 3
Head Cleaning
Oconee 1
Head Cleaning
Oconee 1
Head Cleaning
Oconee 2
Head Cleaning
3rd NDE Inspection
Oconee 2
2nd NDE Inspection
Oconee 2 and 3
Head Cleaning
Oconee 1 and 3
Service Structure Modification
Oconee 2
100% ECT Inspection of ID
Head Cleaning
Oconee 2 and 3
Head Cleaning

Background


NRC Bulletin 2001-01
2000
2001
2002
2003
2004
Oconee 3
Head Cleaning
Oconee 1
1 CRDM and
5 Thermocouple Nozzle
Leaks Detected
Small Amounts of Boric Acid Deposits Identified
Oconee 2
4 CRDM Leaks Detected
Oconee 3
9 CRDM Leaks Detected
Qualified Visual Inspection
Oconee 3
Qualified Visual Inspection
Oconee 1 and 2
Oconee 3 and 1
Reactor Head Replacements
Oconee 2
Reactor Head Replacement


Nuclear Safety
Corrective Action Plan
Qualified Visual Inspection
Qualified NDE
Reactor Head Replacement
Qualified Repair
Risk Assessment
Risk Analysis
Defense in Depth
Safety Margin Evaluation
Root Cause Evaluation
Analytical Evaluation
NDE
Finite Element Analysis
Metallurgical Analysis
Regulatory Compliance
10 CFR 50 App A
ASME Code
10 CFR 50 App B
Tech Spec

Root Cause Evaluation

- Formal root cause evaluation completed for the discovery of VHP leakage on each Oconee unit as required by Duke procedures
- Primary water stress corrosion cracking (PWSCC) is the driving mechanism for the cracking discovered at Oconee
  - » High Residual Stress - Combination of manufacturing and operational stresses
  - » PWSCC is the result of a combination of metallurgy, primary water chemistry and stress factors
  - » Multiple fuel cycles are required for an axial crack to potentially begin to present nuclear safety concerns

Root Cause Evaluation (continued)

- Finite Element Analysis Performed for Oconee
  - » Confirms axial cracks are the preferred crack orientation, if cracking occurs
    - 77 of 90 cracks characterized at Oconee are axial in direction
    - Axial cracking precedes the circumferential cracking
  - » Results predict high axial stresses at uphill and downhill locations which could support circumferential crack growth
- J-groove Weld Cracking Can Occur
  - » Hoop stresses are dominant
  - » Axial cracking is the most likely orientation
  - » Oconee experience supports this conclusion

Root Cause Evaluation (continued)

- Alloy 600 CRDM Nozzle Material
  - » Flaw tolerant and not prone to gross or sudden failures
  - » Material degradation typically through small observable leakage events with significant structural margins
  - » Over 35 Alloy 600 (PWSCC suspected but not confirmed) leakage events in United States alone since 1986
    - Common elements of most occurrences:
      - Small amount of leakage
      - Discovered during system/plant walk downs

Root Cause Evaluation (continued)

- Visual Inspection
  - » Visual inspection of the reactor head is a reliable means of identifying through wall cracking for Oconee
    - Proven effective by the identification of control rod drive mechanisms (CRDM) nozzles with through wall cracks
    - Measures have enabled identification and discrimination of small boric acid deposits
      - Removed old boron deposits from Oconee reactor heads
      - Visual access from multiple angles of each reactor head penetration (VHP) nozzle
        - » Service structure modifications
        - » Elevated insulation design

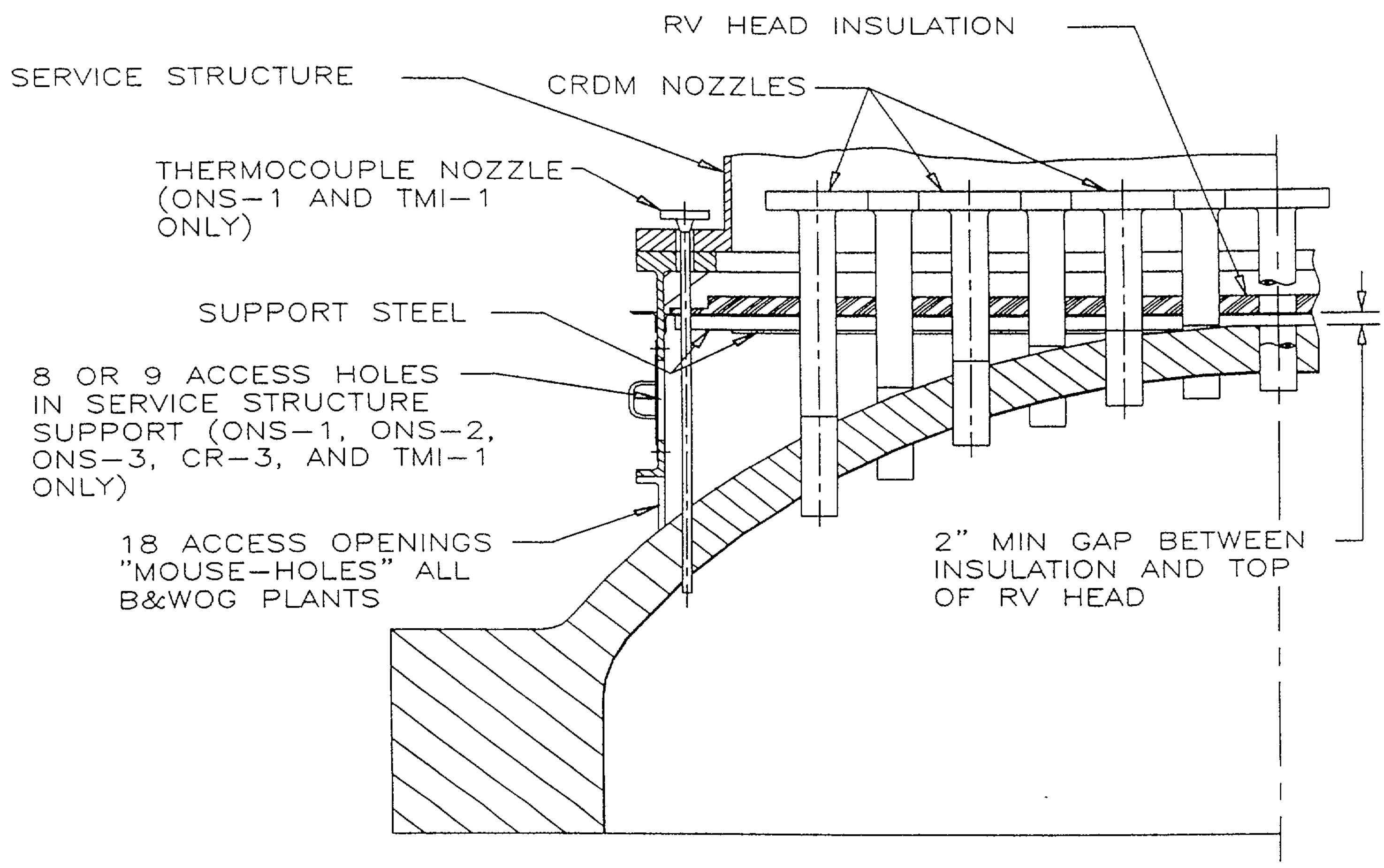
Root Cause Evaluation (continued)

- Summary
  - » PWSCC is the driving mechanism for the observed through wall cracking
  - » Initial crack orientation is axial
  - » Axial through wall cracks precede formation of circumferential cracks
  - » Through wall flaws can be detected by effective visual inspection of the reactor head

Corrective Actions

- Corrective action plan effectively manages and prevents conditions which could contribute to future CRDM nozzle leakage
  - » Qualified visual inspection identifies any suspect nozzles
  - » Non destructive examination (NDE) to characterize suspect nozzles
  - » ASME Code evaluation and/or repairs to disposition indications
  - » Reactor head replacement
- Qualified visual inspection can be relied upon to characterize condition of the VHP nozzles if certain conditions exist
  - » VHP nozzles are visually accessible
  - » A clean reactor head surface exists
  - » A proven leakage pathway exists

Schematic of B&W-Design Reactor Head, CRDM Nozzles, Thermocouple Nozzles, and Insulation



Corrective Actions (continued)

- Qualified Visual Inspection
  - » A bounding finite element analysis of the gap between the CRDM nozzle and the reactor head penetration was performed for the Oconee units
    - A 45° segment model included upper hemispherical head, upper closure flange, and CRDM tubes
    - Analysis performed using normal operating conditions
    - Material properties based on 1989 ASME Code and T=600°F
    - Analysis results conclude a leakage pathway exists for all but 1 Oconee CRDM nozzle
      - – Will volumetrically inspect at next refueling outage (Unit 1, Spring 2002)

Corrective Actions (continued)

- Qualified Visual Inspection Planned for Each Refueling Outage (RFO)
  - » Any nozzle not meeting the qualified visual acceptance criteria will be inspected to determine source of boron deposits and to characterize any cracks
    - Best available NDE technique
  - » Results of the NDE characterization will be used to decide if other supplemental inspections are necessary
    - Other available NDE data
    - Nature and extent of any cracking

Corrective Actions (continued)

- ASME Section XI Repairs of Suspect Nozzles
  - » All thermocouples have been plugged
  - » All suspect through wall cracked CRDM nozzles repaired
    - All cracks removed from suspect nozzles
    - Approved repair methods
      - ASME Section XI Code
      - NRC approved alternatives
  - » Repairs restored reactor coolant pressure boundary integrity prior to returning unit to service

Corrective Actions (continued)

- Oconee has partnered with Framatome-ANP for any future NDE inspections and repairs of CRDM nozzles
  - » Working closely on NDE techniques and delivery systems
  - » Numerous lessons learned from past NDE experiences
    - Reduce occupational exposure
    - Improve unit availability
    - Improve NDE equipment reliability

Corrective Actions (continued)

- Non Destructive Examination
  - » Techniques to examine OD of CRDM nozzle are under development - qualification & demonstration pending
    - Refinement of the OD ultrasonic techniques used at Oconee
    - Axial and circumferential OD ultrasonic test blade probe capability is being developed
    - Alternating Current Field Measurement (ACFM), surface exam

Corrective Actions (continued)

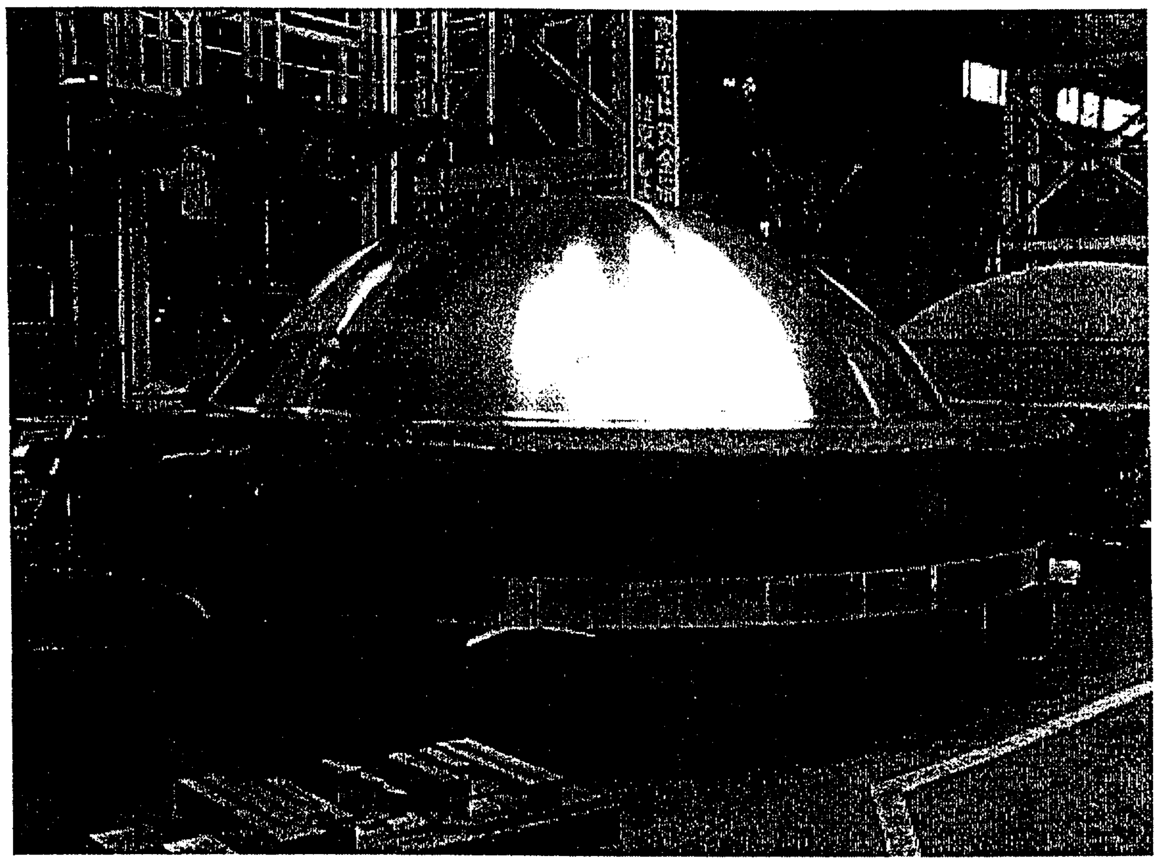
- Non Destructive Examination Limitations
  - » Current state NDE is effective to support a limited number of CRDM nozzle inspections during each RFO
    - Techniques currently lack demonstration
    - Delivery systems can result in significant occupational exposure and unit availability impacts
    - New systems lack field trials in production modes

Corrective Actions (continued)

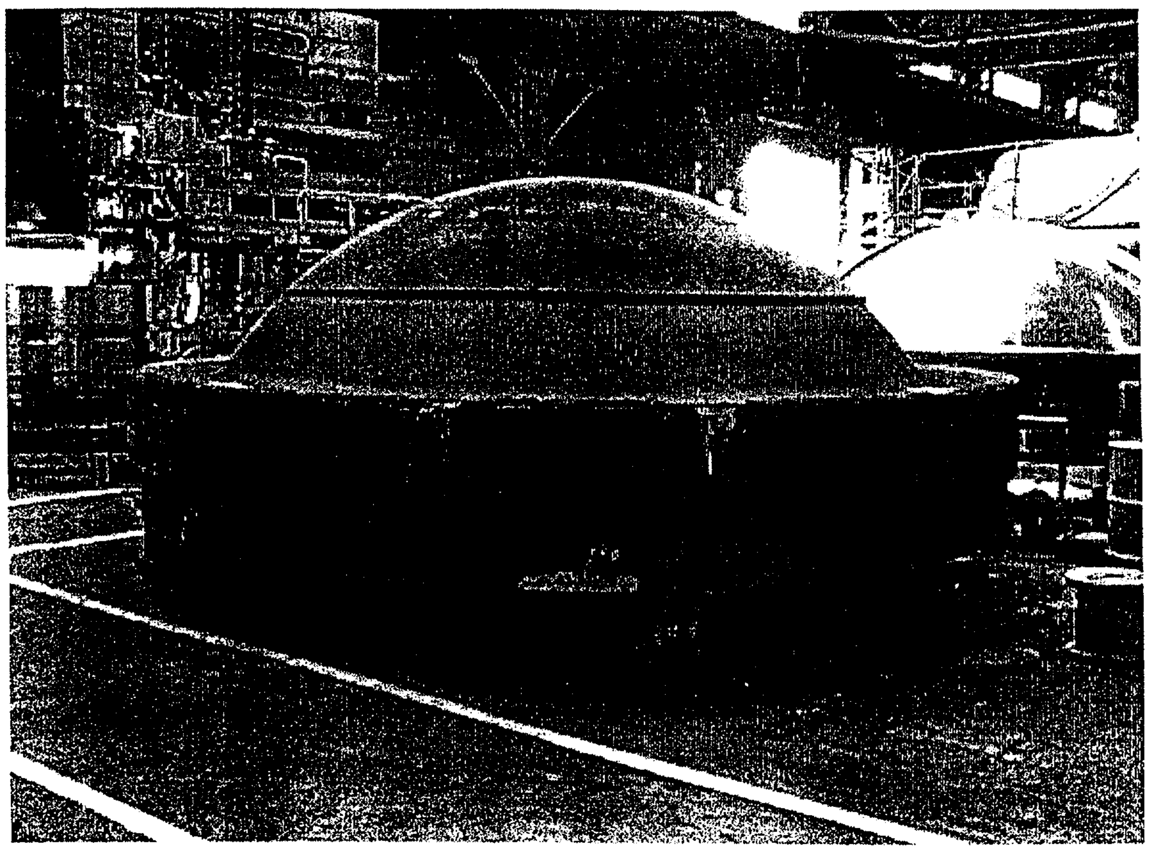
- Reactor Head Replacement
  - » Corrects potential recurrence of Alloy 600 through wall cracking in VHP nozzles for remaining life of plant
  - » Uses upgraded Alloy 690 CRDM nozzles that are more highly resistant to PWSCC
  - » To be inspected using 100% volumetric inspection of CRDM nozzles

Corrective Actions (continued)

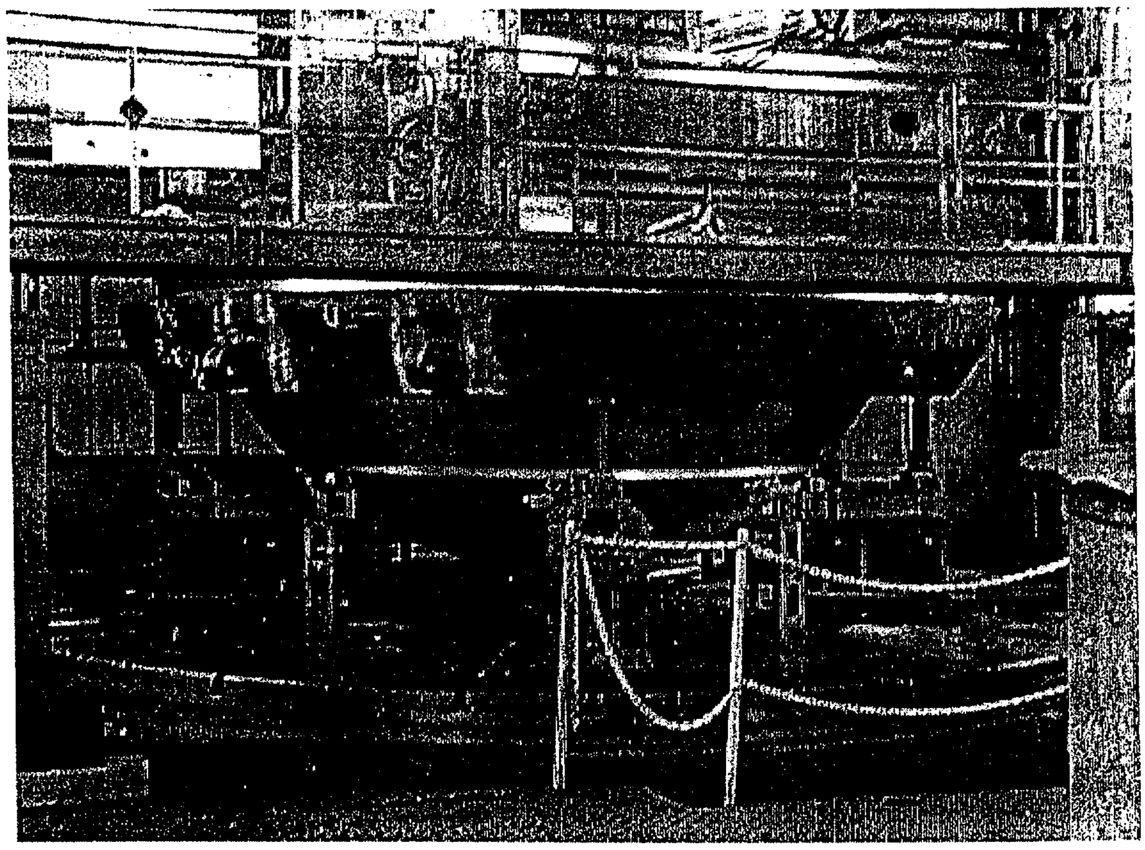
- Reactor Head Replacement Schedule
  - » Corrects condition at first available opportunity
  - » 24 month fabrication and delivery schedule for new heads
    - Receive first head in Cambridge in February 2002
    - Machine J-grooves in first head in July 2002
    - CRDM welds on first head in September to December 2002
  - » Within two 18-month cycles of CRDM through wall cracking discovery
    - Unit 3 - Spring 2003; Unit 1 - Fall 2003; Unit 2 - Spring 2004



BABCOCK & WILCOX CANADA
CONTRACT# 068S
FIRST RPVCH FORGING AT JSW
INSPECTION BY ANI COMPLETED
BEFORE OVERLAY CLADDING



BABCOCK & WILCOX CANADA
CONTRACT# 068S
SECOND RPVCH FORGING AT JSW
AFTER QUENCHING AND TEMPERING



BABCOCK & WILCOX CANADA
CONTRACT# 068S
THIRD RPVCH FORGING AT JSW
ROUGH MACHINING OF
INNER SURFACE

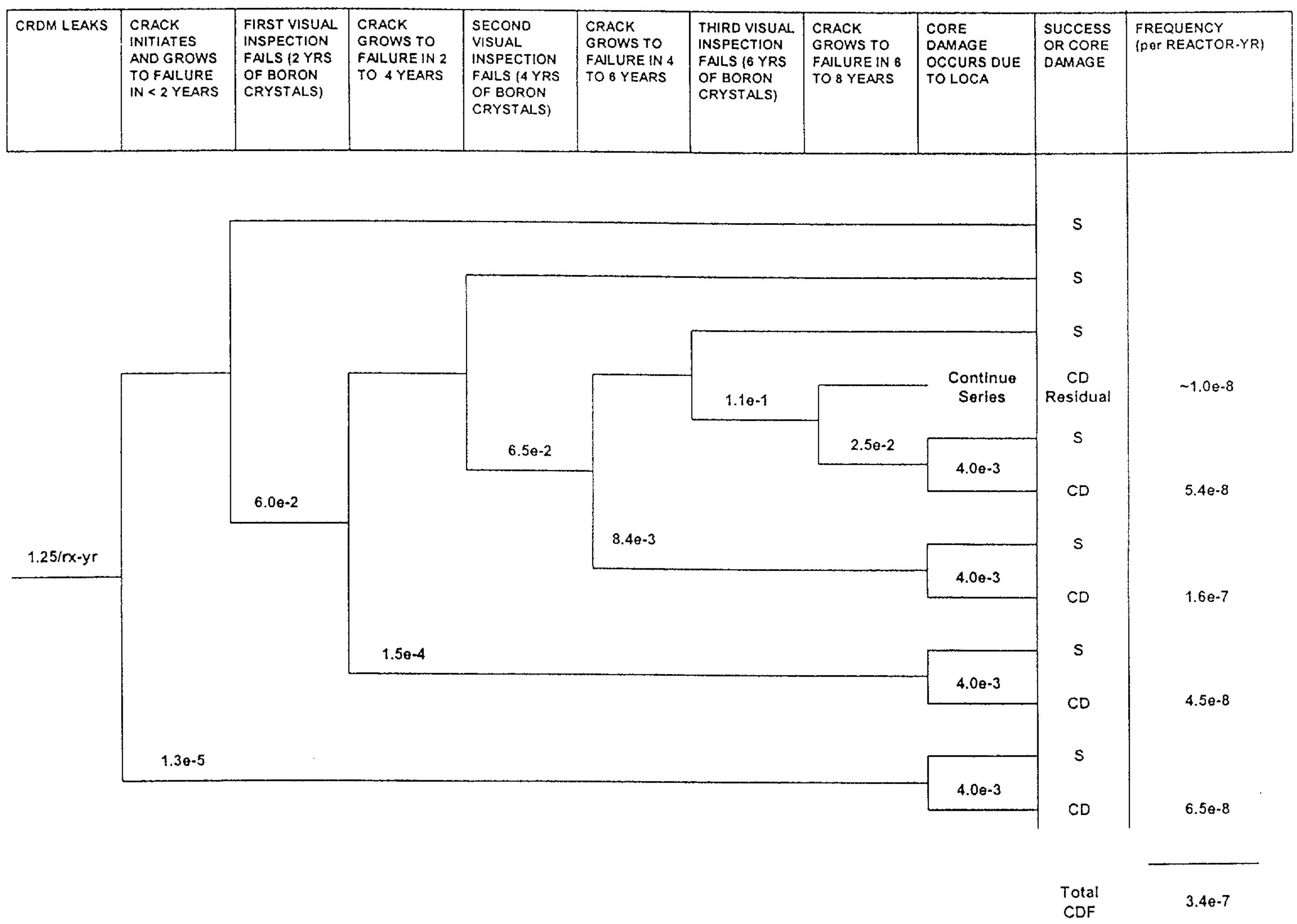
Corrective Actions (continued)

- Summary of Corrective Actions
  - » Qualified visual inspection of all VHPs to confirm structural and leak tight integrity
  - » Perform additional inspections to characterize any deposits and any through wall cracks
  - » Repairs made in accordance with ASME Code and NRC approved alternatives
  - » Reactor head replacement on first available opportunity basis

Risk Assessment

- Risk assessment performed for B&WOG/EPRI
- Based upon deterministic safety assessment performed for B&WOG/EPRI
- B&WOG risk assessment tailored for Oconee-specific risk assessment

Generic B&WOG Event Tree for OD PWSCC



Risk Assessment (continued)

- Initiating Event Frequency
  - » Weld or nozzle through wall axial cracks are included
  - » CHECWORKS predicts 0.52/rx-yr for ID initiated cracks (not including weld cracks)
  - » Experience = 1.25/rx-yr
    (15 leakers @ 4 plants, assume two 18-month fuel cycles)

Risk Assessment (continued)

- Probability of Undetected Through Wall Cracking
  - » Human Reliability Analysis employed
  - » Based on effective visual inspection
    - Leakage pathways confirmed, clean heads
  - » Uses Swain & Guttman and Human Cognitive Reliability Model

RFO	#1	#2	#3+
HEP	0.06	0.065	0.11

  - » Results in a conservative human error probability (HEP) considering future emphasis on qualified visual inspection

Risk Assessment (continued)

- Time to CRDM failure
  - » Crack growth model based on deterministic fracture mechanics analysis
    - Monte Carlo Simulation
    - Distributions from industry data or conservative assumption
    - Uncertainty in probabilistic fracture mechanics (PFM) data addressed via conservative assumptions

Risk Assessment (continued)

- Probability of OD Circumferential Crack Initiation
  - » OD crack assumed to initiate on 100% of CRDMs, once OD is wetted
  - » Zero time-to-initiation assumed, once OD is wetted
  - » Most conservative assumption used in lieu of supported crack models -- bounds uncertainty

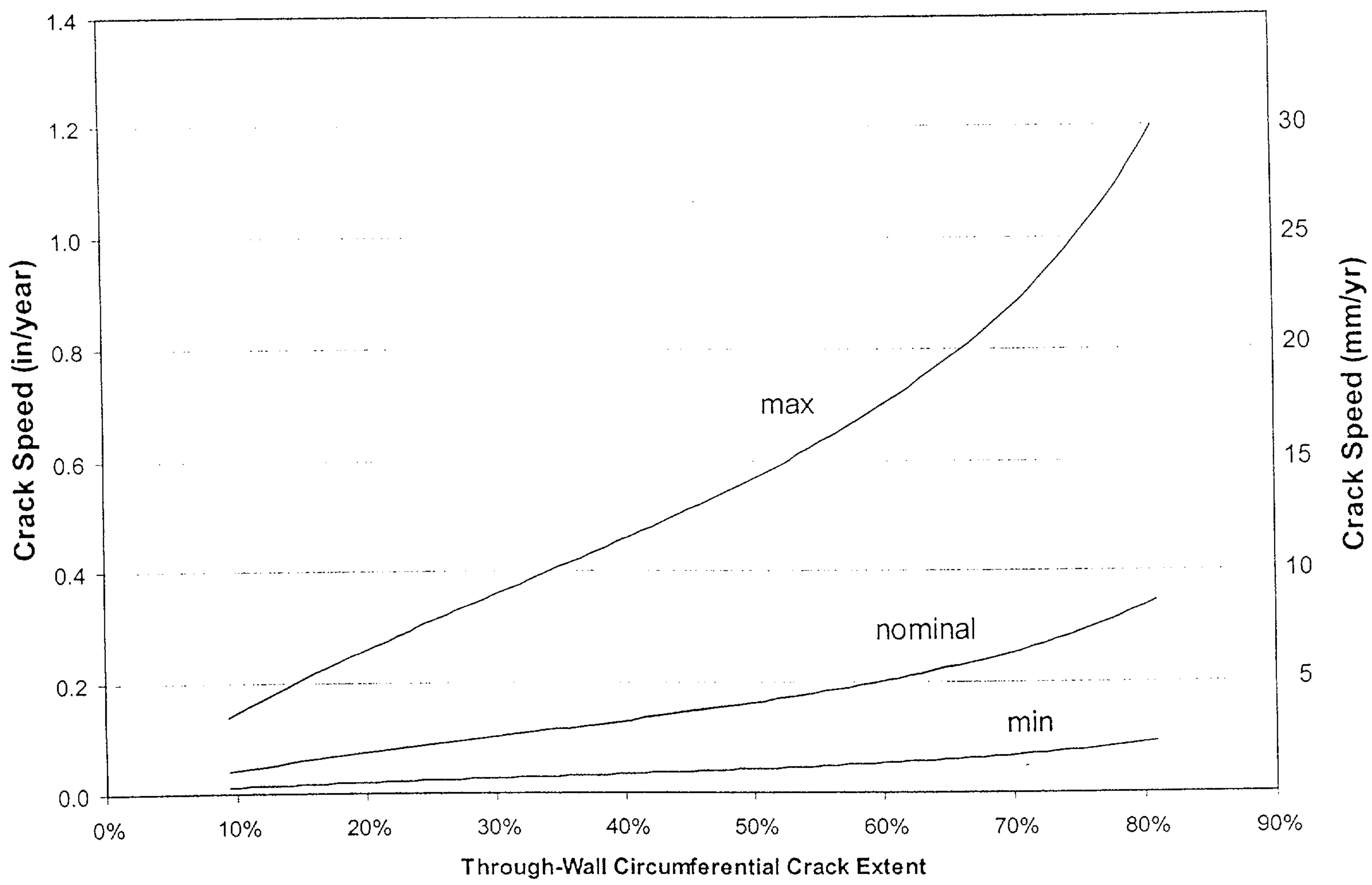
Risk Assessment (continued)

- Initial Flaw Size
  - NDE data for observed above-the-weld OD circumferential cracks indicates an extent of about 36, 66, 80, 165 degrees
  - Could have grown from single or multiple initiation sites
  - Multiple sites are approximated by a single very-long flaw
  - Assume uniform distribution from 0 to 180 degrees for initial flaw extent
  - Addresses uncertainty with respect to flaw size and multiple initiation sites

Risk Assessment (continued)

- Crack Growth Rate Parameters:
  - » EPRI data for PWSCC
  - » Peter Scott Model
    - Function of stress intensity, temperature, etc.
  - » Worst-case results from stress analysis
  - » Growth rate parameters distributed to address uncertainty

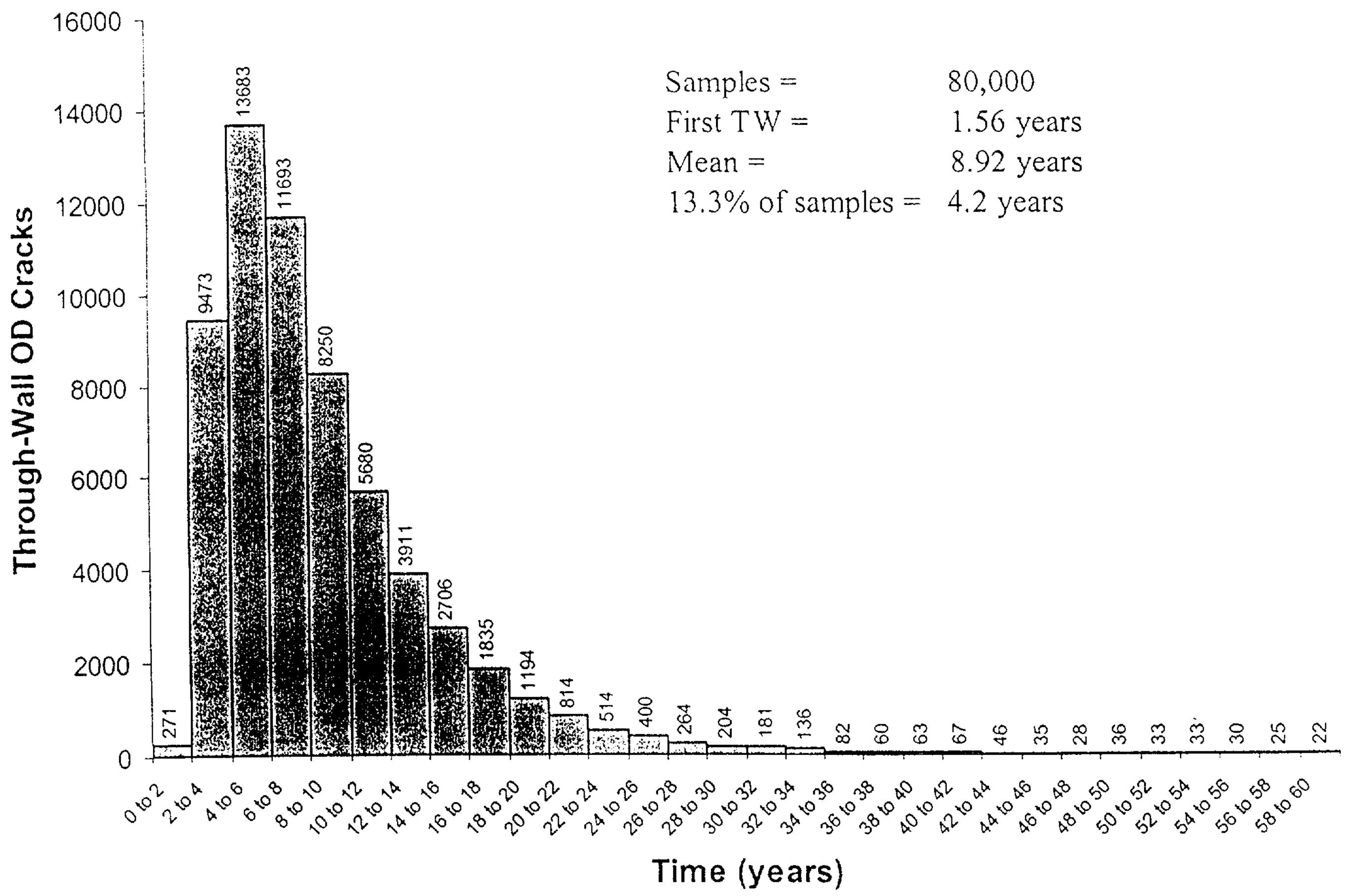
Crack Growth Rate Assumed in Monte Carlo Simulation



Risk Assessment (continued)

- Benchmark Simulation Against Oconee Observations
  - » Observation:
    - 2 of the 15 leaking CRDM nozzles (13.3%) had through wall (TW) or almost TW OD circumferential cracks
  - » Time-to-TW simulation:
    - Cumulative impact of PFM data and assumptions
    - 13.3 % of trials reached through wall in 4.2 years
  - » Crack growth model agrees or is conservative with respect to the observations

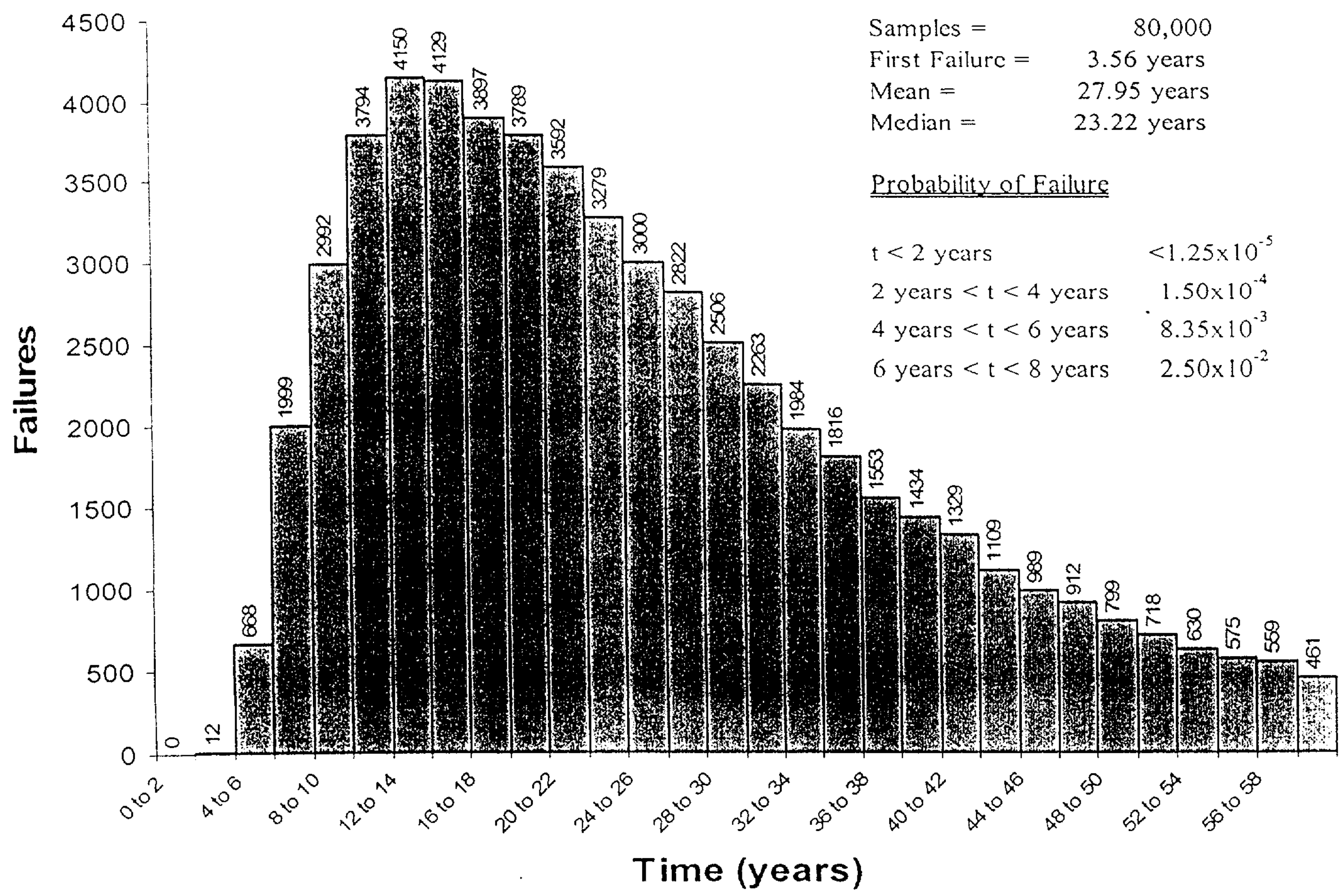
Time-to-TW Simulation for OD PWSCC



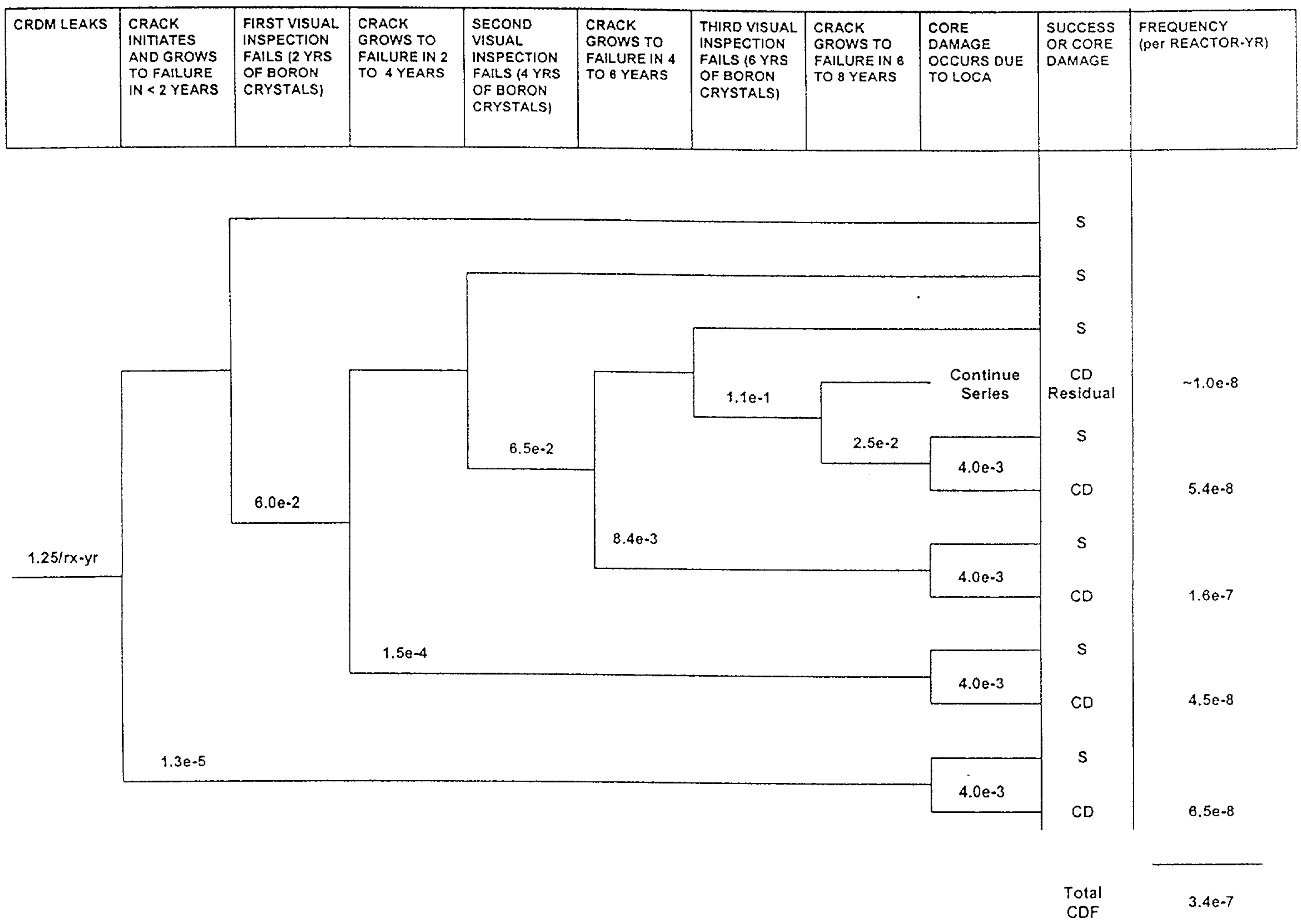
Risk Assessment (continued)

- Time-to-Failure Simulation
  - » Failure is defined as an insufficient ligament to meet ASME stress limits using safety factor of 3 (1.5 for emergency and faulted conditions)
  - » Corresponds to a circumferential extent of approximately 81%
  - » Conservative failure definition is used to bound uncertainty

Time-to-Failure Simulation for OD PWSCC



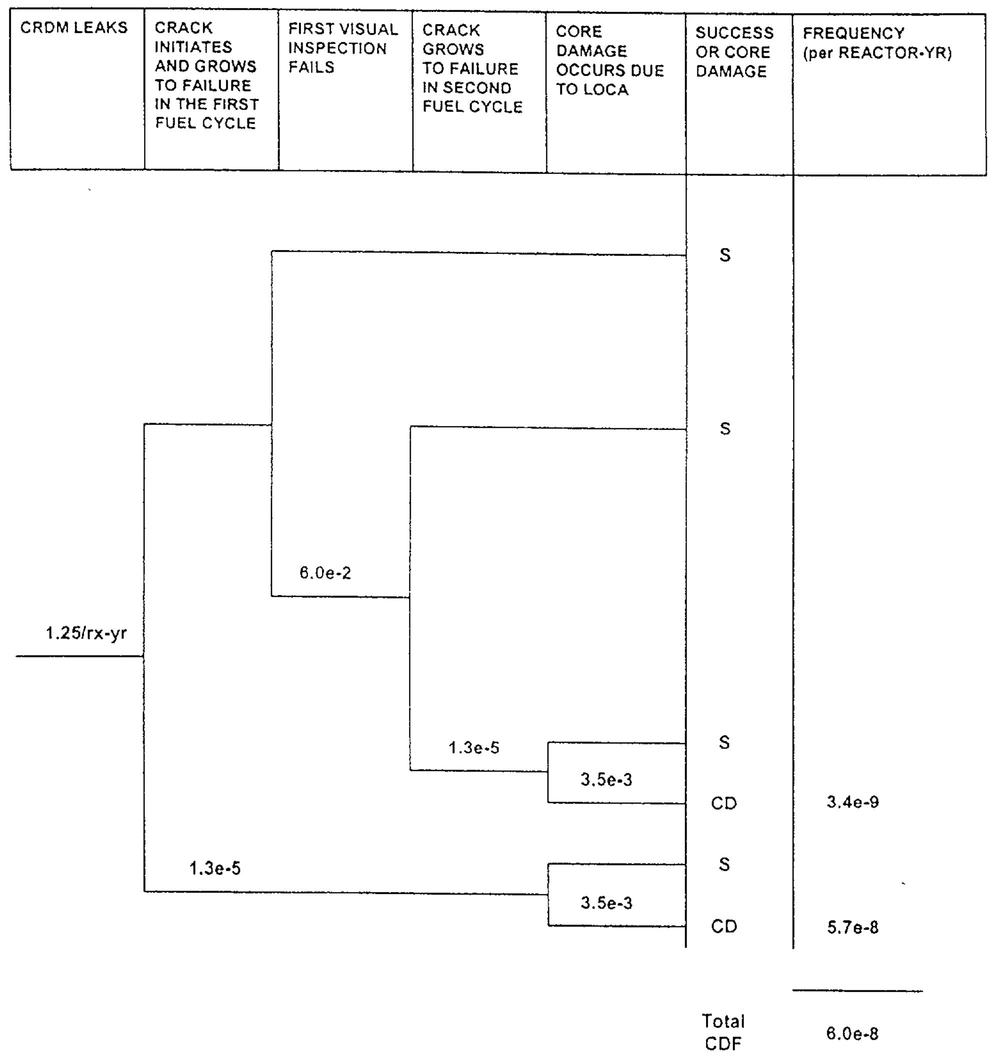
Generic B&WOG Event Tree for OD PWSCC



Risk Assessment (continued)

- Probability of Core Damage
  - » Medium break LOCA (MBLOCA) is evaluated assuming a complete severance
  - » CCDP for MBLOCA is:
    - Average B&WOG = 4.0E-3
    - Oconee PRA, Rev. 2 (Dec. 1996) = 3.5E-3
  - » CCDP for MBLOCA bounds small break LOCA (SBLOCA)
    - SBLOCA = 1.6E-3 (average B&WOG)
      = 2.9E-3 (Oconee PRA, Rev. 2, Dec. 1996)
  - » Conservative because Safety Analysis review indicates LOCA at top of vessel is easier to mitigate

Oconee-Specific Event Tree



Risk Assessment (continued)

- Results
  - » CDF due to OD PWSCC
    - Generic B&WOG = 3.4E-7/rx-yr
    - Oconee until head replacement = 6.0E-8/rx-yr
  - » Conservative assumptions assure conservative results
    - Treatment of probability of leak
    - Human error probability for visual inspection
    - OD crack initiation
    - Initial flaw size
    - Stresses
    - Failure definition
    - LOCA response

Risk Assessment (continued)

- Containment Failure Risk
  - Large Early Release Frequency is Insignificant
    - CRDM nozzle cracking has no impact on containment safeguards systems
    - Containment isolation and containment heat removal should both be available for the most likely core damage sequences
    - Oconee has a large dry containment with a mean failure pressure of 143 psi, so it is NOT susceptible to early over pressurization failure modes

Risk Assessment (continued)

- Public Health Risk
  - » Oconee PRA Rev. 2 was used to calculate Public Health Risk
  - » Postulating a MBLOCA core damage accident, the conditional person-rem is 1.1E4
  - » This is the weighted average of all the potential containment failure modes and expected emergency response scenarios
  - » The most likely sequence is intact containment with reactor building cooling units operating, so releases are low

Risk Assessment (continued)

- Public Health Results
  - » Incremental CDF
    - Generic B&WOG = 3.4E-7/rx-yr
    - Oconee until head replacement = 6.0E-8/rx-yr
  - » LERF not directly affected because containment safeguards not affected
  - » Public health risk = 6.0E-8/rx-yr x 1.1E4 person-rem
    = 6.6E-4 person-rem/rx-yr
  - » 6.6E-4 person-rem/yr (Oconee) is well below person-rem for volumetric inspection

Safety Margin

- The CDRM nozzle cracking does NOT significantly increase the LOCA probability assumed in the PRA nor does it increase the frequency above the level assumed for design basis accidents
- LOCA and rod ejection accidents are part of the design basis for ECCS and containment systems. A CRDM nozzle failure would be less severe than assumed in FSAR analysis
  - » Break location is less severe than a LOCA in the cold or hot legs
- Evaluated impact to nuclear fuel in unlikely event of a rod ejection accident using NRC approved methodology and Oconee specific core design inputs
  - » No fuel damage for rod ejection for the actual core designs

Defense In Depth

- Reactor coolant pressure boundary is maintained
  - Visual inspection is qualified by analysis to identify through wall cracked VHP nozzles
  - Even conservative crack growth rates DO NOT result in failure in two 18 month cycles
- Core damage mitigation is maintained
  - CRDM nozzle cracking has NO affect on LOCA mitigation capability
- Containment safeguards are maintained
  - CRDM nozzle cracking has NO affect on containment functions
- Emergency planning effectiveness is maintained
  - CRDM nozzle cracking has NO affect on emergency planning

Risk Assessment Summary

- Risk Assessment shows that the increase in risk to the public from CRDM nozzle cracking is insignificant
- Effects of degradation will be observed and corrected *before* there is significant risk of a CRDM detachment

Regulatory Compliance

- **10 CFR 50 App. B Criterion XVI** - Corrective Action - Conditions adverse to quality are to be promptly identified and corrected - significant conditions adverse to quality shall have the cause determined and corrective action taken to preclude repetition
- **Corrective actions:**
  - » Cause of condition for each Oconee leakage event was determined formally using established root cause processes.
  - » Extent of condition was determined through supplemental inspections (e.g., other Alloy 600 components and non-leaking CRDM nozzles)

Regulatory Compliance (continued)

- **Corrective actions: (continued)**
  - » Interim
    - Inspections and repairs identified and corrected conditions prior to returning units to service
    - Duration of the interim period is limited by reactor head replacement schedule
    - Monitoring and inspection will provide adequate condition monitoring until reactor head replacements complete
  - » Corrective actions identified to date are commensurate with scope of identified conditions and their safety significance

Conclusions

- Reactor Heads Have Been Characterized
  - » 100% Effective Visual Examination
    - Identified 14 through wall cracked CRDM nozzles
    - Identified thermocouple through wall cracks
- NDE Examination
  - » 39% (81) non-leaking CRDM nozzles ID characterized
  - » 19% CRDM nozzles ID and OD volumetrically characterized
    - 26 non-leaking nozzles
    - 14 suspect CRDM nozzles for repair

Conclusions (continued)

- Phenomena of VHP Nozzle Cracking is Understood
  - » Alloy 600 - Primary Water Stress Corrosion Cracking is driving mechanism
  - » Evidence of boric acid deposits is most reliable early indicator of degradation
    - Cracks do not propagate rapidly
    - Boric Acid deposits provide evidence of through wall cracking
    - Axial through wall cracks precede formation of circumferential cracks

Conclusions (continued)

- Extremely unlikely to grow a circumferential crack to critical flaw size in a single 18 month operating cycle
- No nuclear fuel failures from a postulated rod ejection accident
- Insignificant risk from OD PWSCC in time period prior to head replacement

Conclusions (continued)

- Corrective Actions are Reasonable and Appropriate
  - Repairs have restored reactor coolant pressure boundary integrity prior to returning unit to service
  - Qualified visual examination of CRDM nozzles allows monitoring of conditions that could contribute to future reactor coolant pressure boundary leakage
  - Aggressive replacement of Reactor Heads limits interim period

Conclusions (continued)

- Commitment to Nuclear Safety is Foremost
  - » Design Criteria remains effective
    - NO rapid propagation of cracks
    - NO gross leakage
  - » Manage Risk
    - Corrective Action plan ensures NO significant increase in risk