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

Docket Number: 491st Meeting

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/ Rebecca Davis

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

March 18, 2002

SCHEDULE AND OUTLINE FOR DISCUSSION
491st ACRS MEETING
APRIL 11-13, 2002

THURSDAY, APRIL 11, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open)
- 1.1) Opening statement (GEA/JTL/SD)
- 1.2) Items of current interest (GEA/SD)
- 1.3) Priorities for preparation of ACRS reports (GEA/JTL/SD)

2) 8:35 - 10:30 A.M. Final Review of the Turkey Point License Renewal Application (Open) (MVB/RE/SD)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the NRC staff and the Florida Power and Light Company regarding the license renewal application for Turkey Point Units 3 and 4, and the associated staff's Safety Evaluation Report (SER).

10:30 - 10:45 A.M. ***BREAK***

3) 10:45 - 12:30 P.M. Advanced Reactor Research Plan (Open) (TSK/MME)
- 3.1) Remarks by the Subcommittee Chairman
- 3.2) Briefing by and discussions with representatives of the Office of Nuclear Regulatory Research (RES) regarding RES' draft Advanced Reactor Research Plan.

12:30 - 1:30 P.M. ***LUNCH***

4) 1:30 - 3:30 P.M. CRDM Penetration Cracking and Reactor Pressure Vessel Head Degradation (Open) (FPF/MWW)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of the NRC staff and industry, including Davis-Besse regarding issues related to the investigation of circumferential cracks in PWR control rod drive mechanism (CRDM) penetration nozzles and weldments, and reactor pressure vessel head degradation at the Davis-Besse Nuclear Power Plant.

Other interested parties may provide their views, as appropriate.

3:30 - 3:50 P.M. ***BREAK***

5) 3:50 - 5:15 P.M. Westinghouse Owners Group (WOG) and Electric Power Research Institute (EPRI) Initiatives Related to Risk-Informed Inservice Inspection of Piping (Open) (WJS/FPF/TJK/SD)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the staff's draft safety evaluation reports on WOG and EPRI addendums to their topical reports (WCAP-14572 and EPRI TR-112657) for risk-informed inservice inspection of piping, including extension of risk-informed methods to the break exclusion region piping.

Representatives of WOG and EPRI may provide their views, as appropriate.

5:15 - 5:30 P.M. ***BREAK***

6) 5:30 - 7:00 P.M. Proposed ACRS Reports (Open)

6.1) Final Review of the Turkey Point License Renewal Application (MVB/RE/SD)

6.2) Advanced Reactor Research Plan (TSK/MME)

6.3) Circumferential Cracking of CRDM and PWR Vessel Head Degradation (FPF/MWW)

FRIDAY, APRIL 12, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (GEA/JTL/SD)

8) 8:35 - 10:30 A.M. General Electric (GE) Nuclear Energy Topical Report: "Constant Pressure Power Uprate" (Open/Closed) (JDS/PAB)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff and General Electric Nuclear Energy regarding GE Topical Report, "Constant Pressure Power Uprate," and the associated NRC staff's safety evaluation.

NOTE: A portion of this session may be closed to discuss General Electric proprietary information.

10:30 - 10:50 A.M. ***BREAK***

9) 10:50 - 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (GEA/JTL/SD)

9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, and organizational and personnel matters relating to the ACRS.

10)	11:45 - 12:00 Noon.	Reconciliation of ACRS Comments and Recommendations (Open) (GEA, et al./SD, et al.) Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
	12:00 - 1:00 P.M.	*****LUNCH*****
11)	1:00 - 7:00 P.M.	Proposed ACRS Reports (Open) Discussion of proposed ACRS reports on: 11.1) Final Review of the Turkey Point License Renewal Application (MVB/RE/SD) 11.2) Advanced Reactor Research Plan (TSK/MME) 11.3) Circumferential Cracking of CRDM and PWR Vessel Head Degradation (FPF/MWW) 11.4) GE Topical Report, "Constant Pressure Power Uprate" (JDS/PAB)

SATURDAY, APRIL 13, 2002, CONFERENCE ROOM 2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

12)	8:30 - 12:30 P.M.	Proposed ACRS Reports (Open) Continue discussion of proposed ACRS reports listed under Item 11.
13)	12:30 - 1:00 P.M.	Miscellaneous (Open) (GEA/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**
- **Thirty-Five (35) copies of the presentation materials should be provided to the ACRS.**

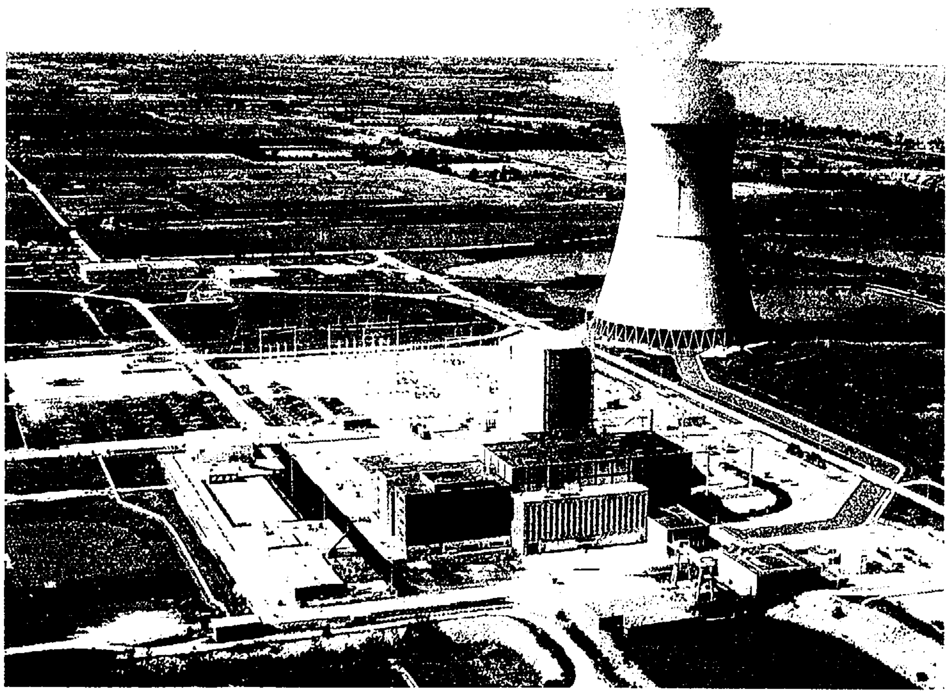
Reactor Pressure Vessel Head Degradation at the Davis-Besse Nuclear Power Station

April 11, 2002

Agenda

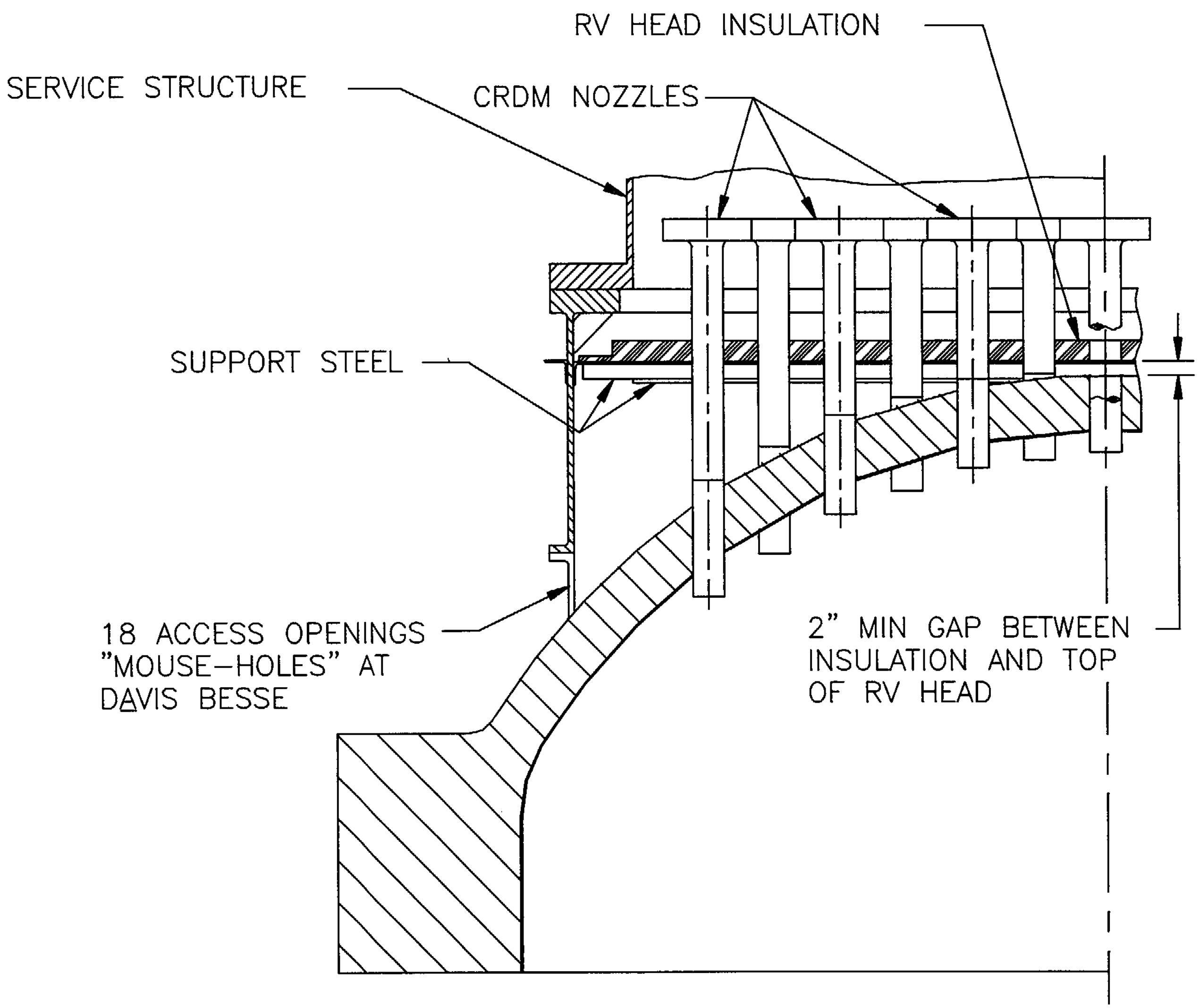
- Background / Discovery of Characteristics of Reactor Vessel Head Degradation
  - John Wood
- Safety Significance Assessment
  - Ken Byrd

Background

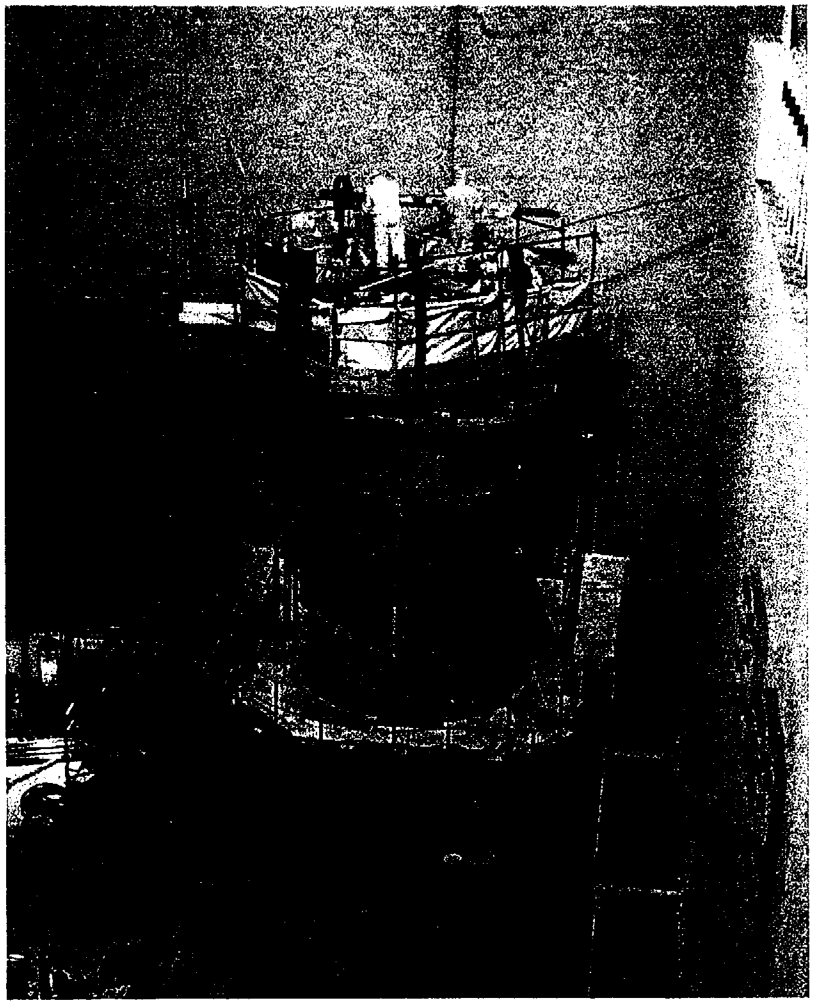


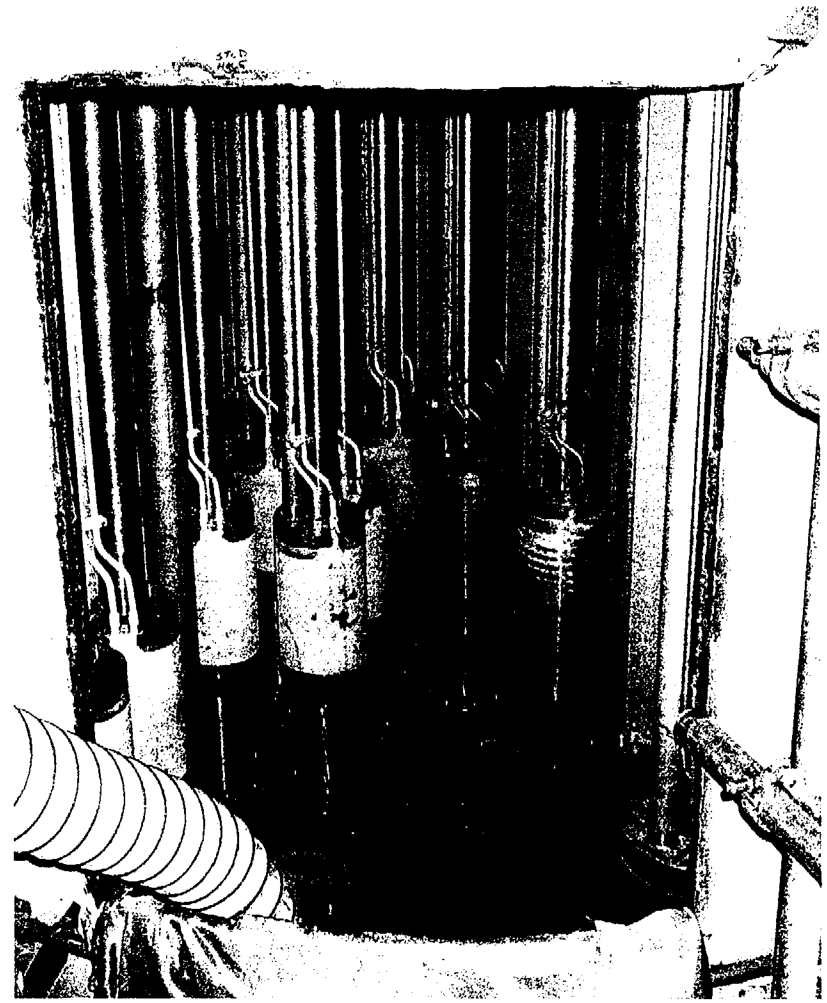
- Davis-Besse is located near Oak Harbor, Ohio
- Commercial operation began August 1977
- B&W Pressurized Water Reactor design (raised loop, 177 fuel assemblies)
- 2772 MWth
- 15.8 Effective Full Power Years
- Reactor Coolant System Pressure = 2155 psig
- Average Coolant Temperature (T_{ave}) =582°F
- Hot Leg Temperature (T_{hot}) = 605°F
- 69 nozzles in Reactor Pressure Vessel (RPV) Head (61 nozzles used for control rod drives)

RPV Head Configuration



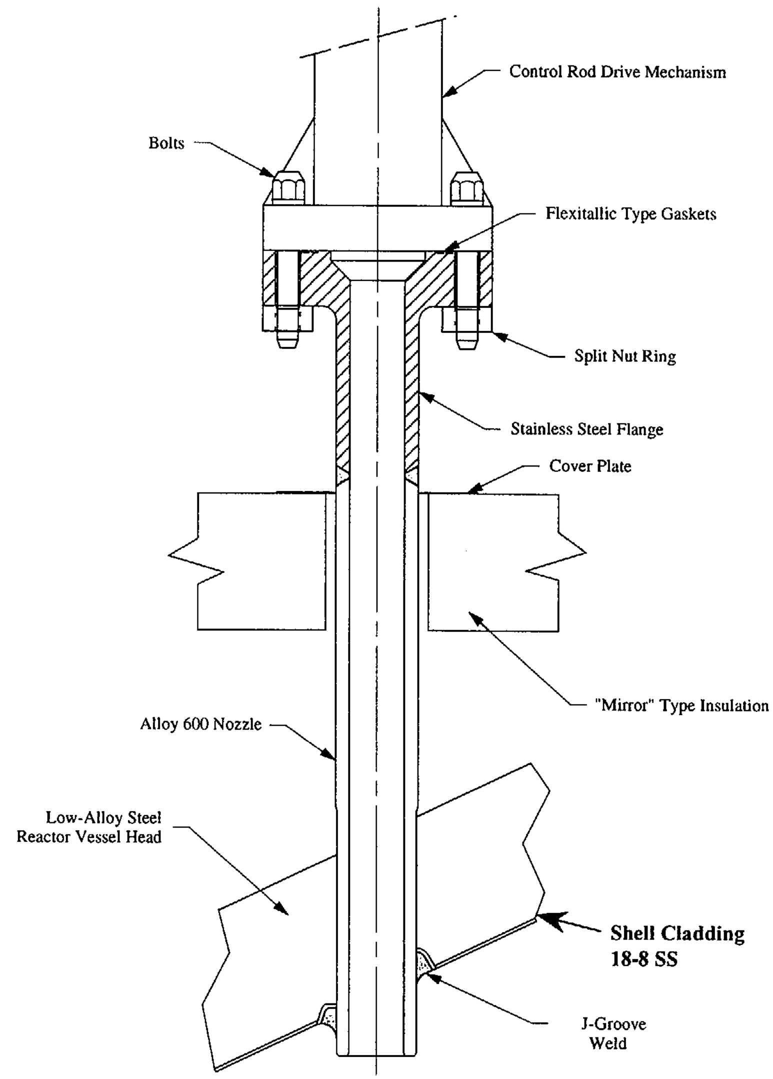
RPV Head Configuration





Inspection Results

Typical B&W Control Rod Drive Nozzle



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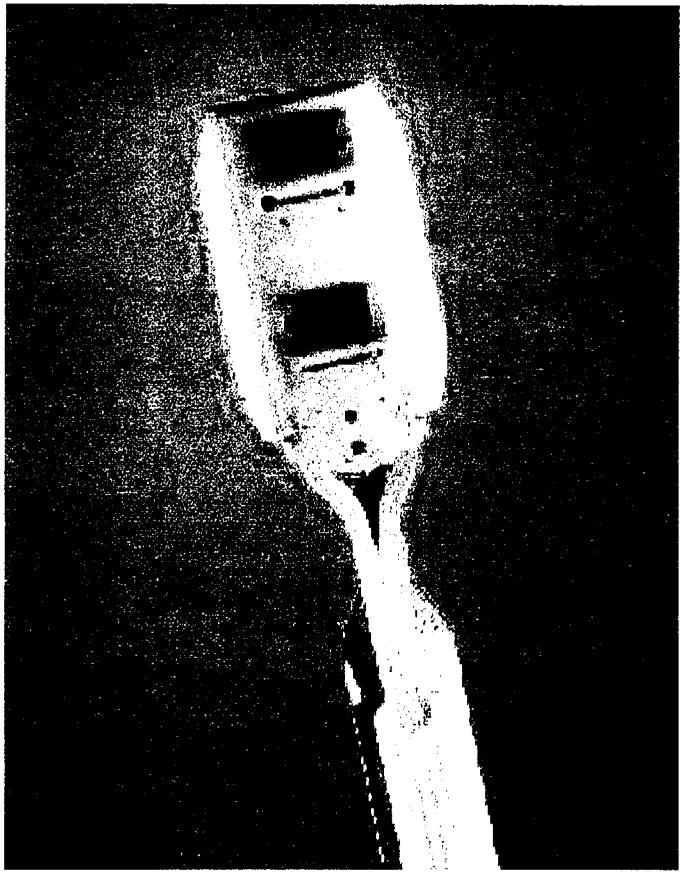
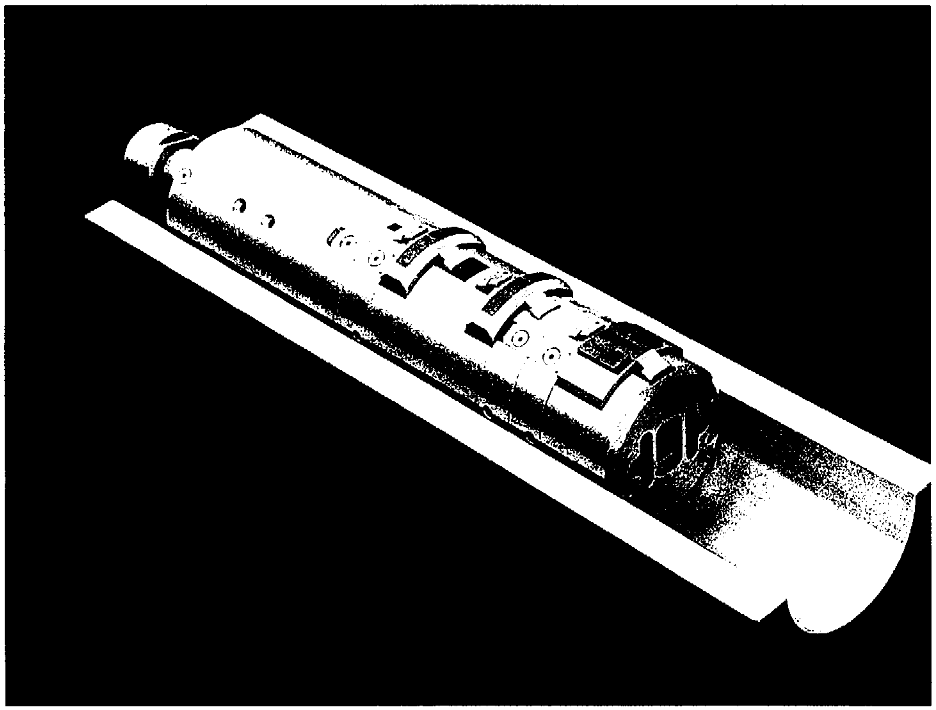
Under Head UT Examination



Circumferential Blade UT Probe

- Blade UT probe used from beneath the head
- Inserted in gap between guide tube and nozzle
- Optimized for circumferential flaw detection and characterization
- EPRI-demonstrated capability for axial and circumferential flaw detection
- Capability to identify leak path
- ARAMIS robotic delivery system and automated data acquisition system used

Top-down UT Examinations



Top-down UT Tool

- Used Top-down UT tool
- Detection and characterization of axial cracks
- Detection and characterization of both axial and circumferential cracks
- Capable of leak path identification

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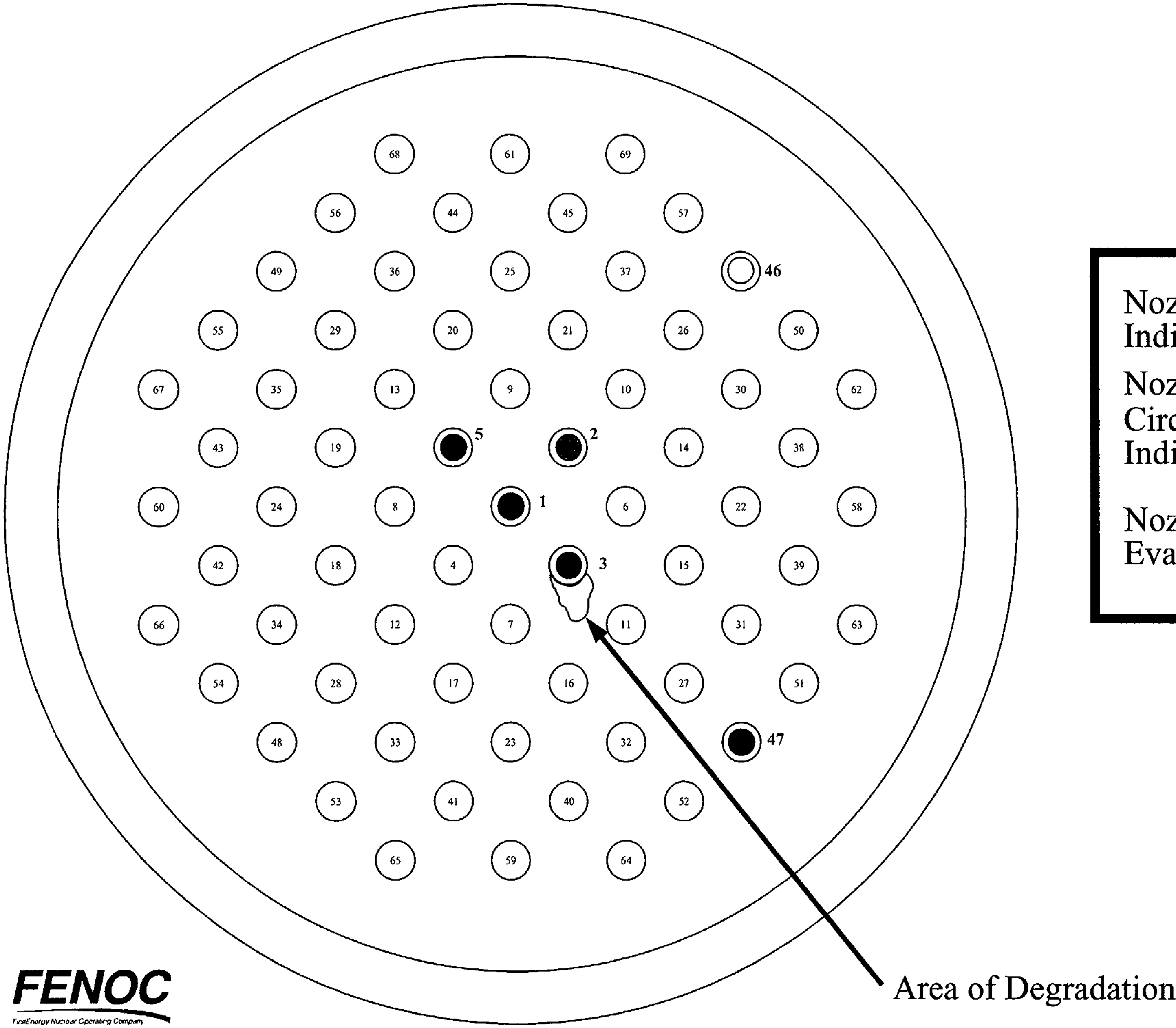
UT Examination Results

Nozzle #	Summary of Results
1	9 Axial Flaws (2 through-wall with a leak path)
2	8 Axial Flaws, 1 Circumferential Flaw (6 through-wall with a leak path)
3	4 Axial Flaws (2 through-wall with a leak path)
5	1 Axial Flaw
47	1 Axial Flaw
46	No crack flaws. Further investigation ongoing

Inspection Results

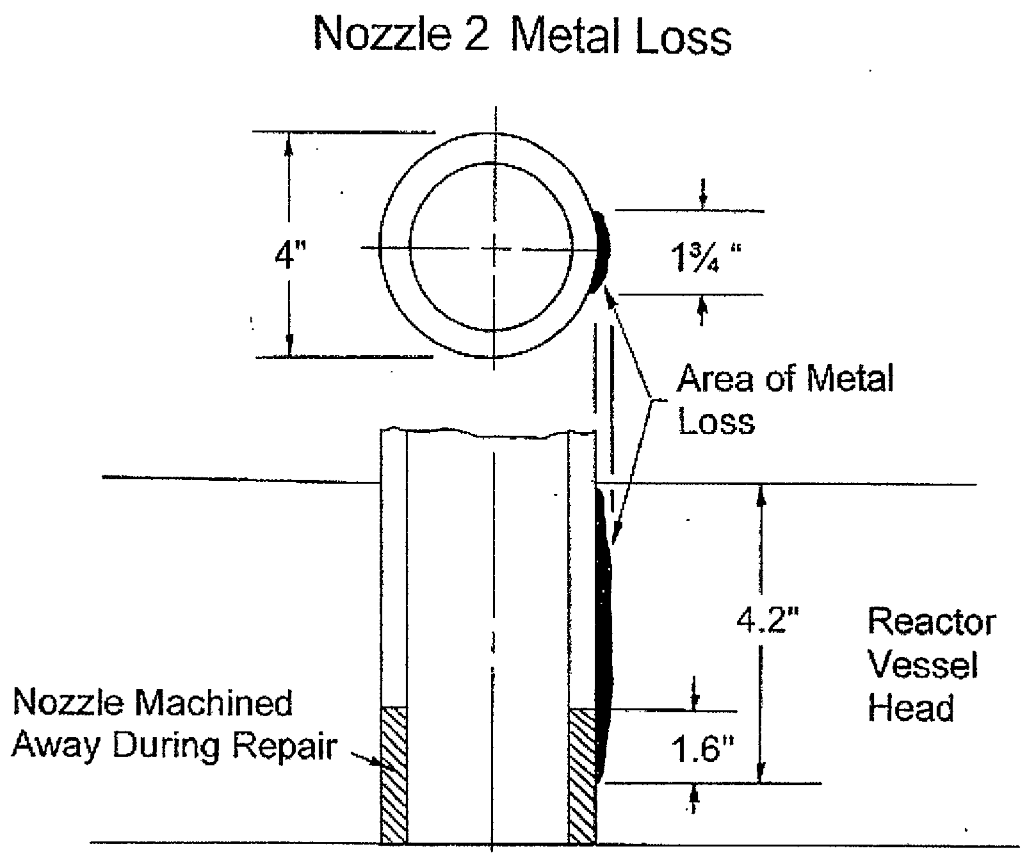


Nozzle with Axial Indication - ●

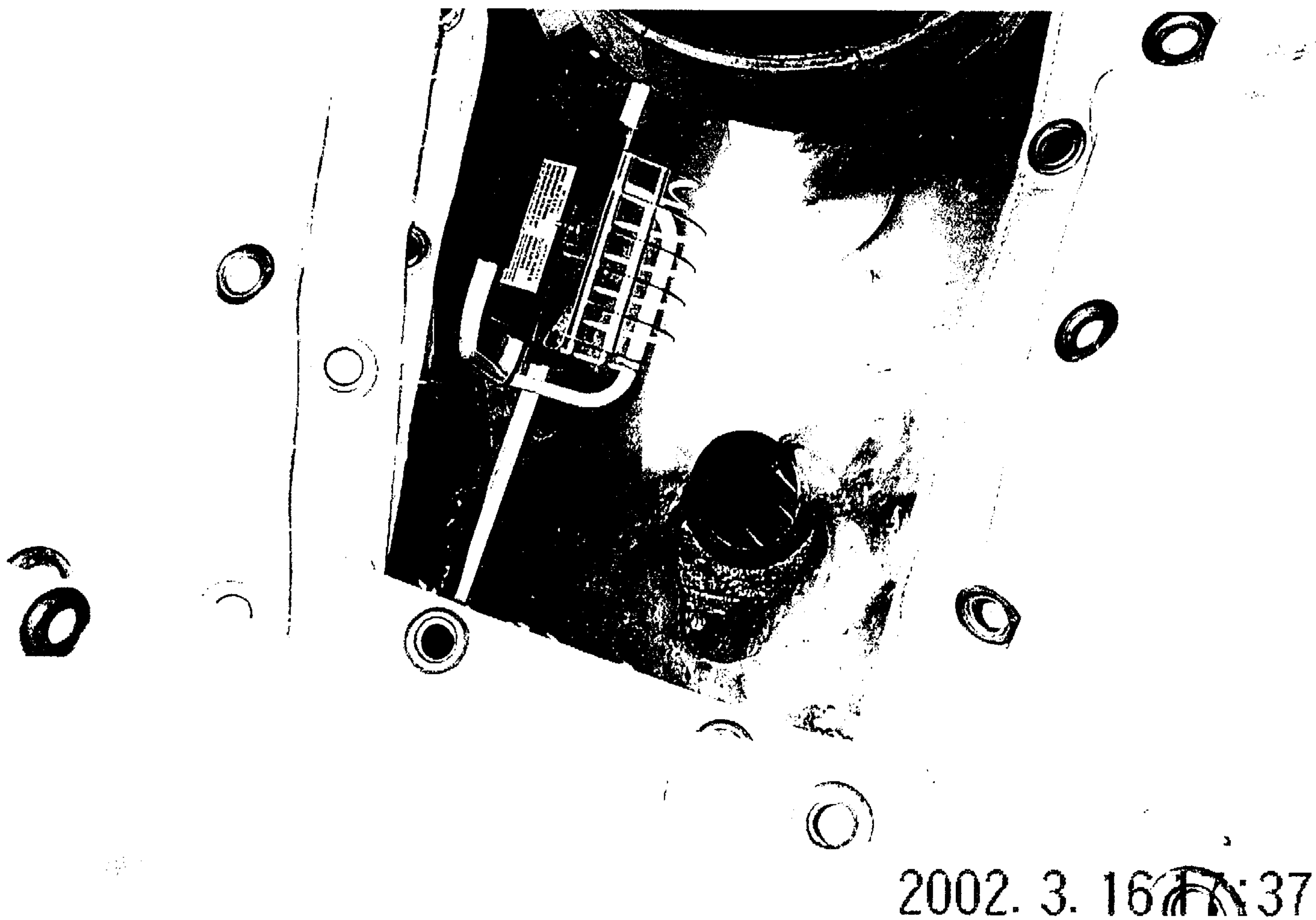
Nozzle with Axial and Circumferential Indication - ●

Nozzle under Evaluation - ○

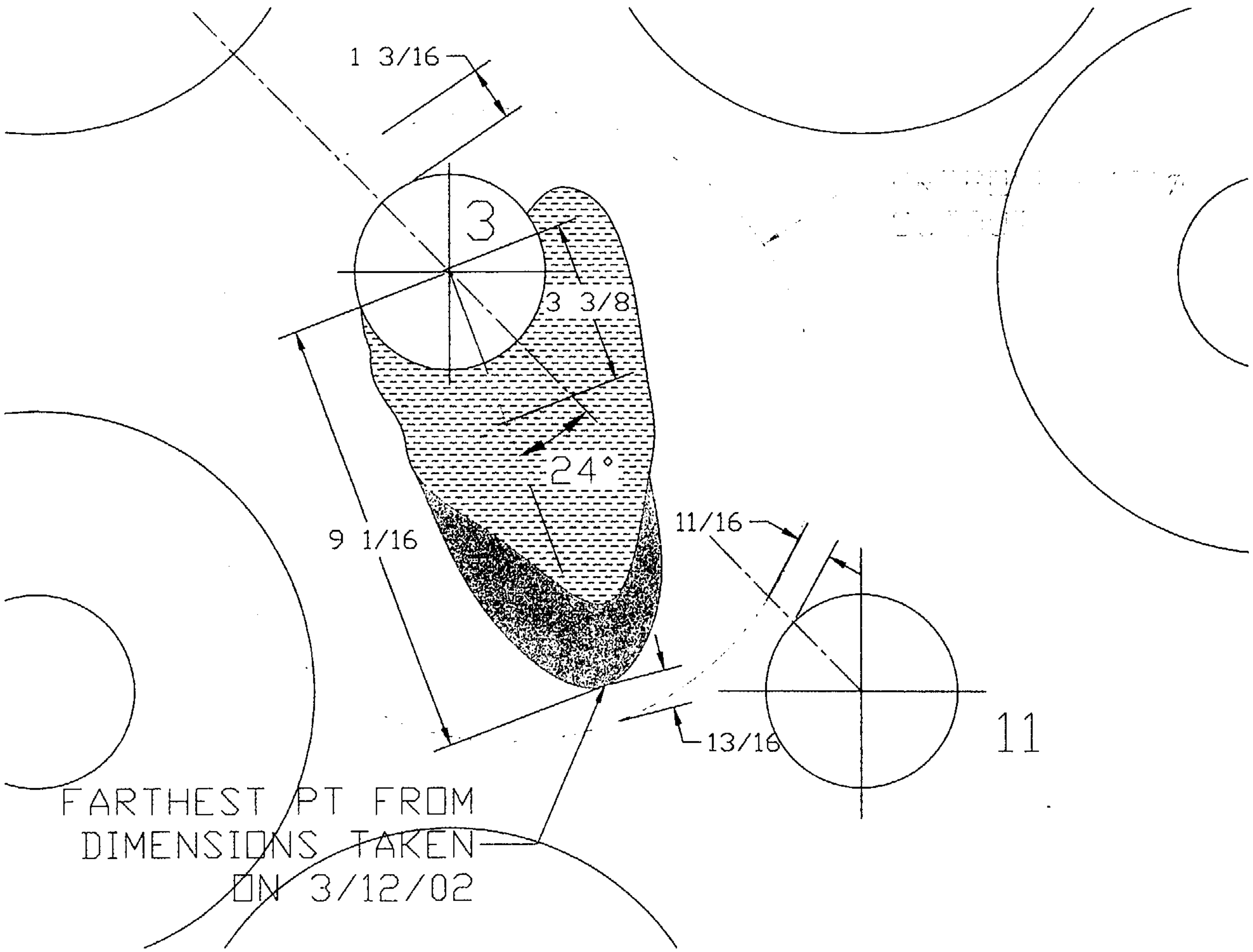
Nozzle 2 Corrosion Profile



Visual Inspection of Nozzle 3



Nozzle 3 Corrosion Profile



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Root Cause Initial Investigation Team

- Chartered to determine root and contributing causes for RPV head damage experienced at nozzles 2 & 3
- Team led by FENOC staff
- Team augmented by industry experts from Framatome, Dominion Engineering, and EPRI

Probable Timeline

- 1990 (+/- 3yrs) Nozzle 3 crack initiated
- 1994-1996 Nozzle 3 crack propagates through-wall
- 1998-2000 Nozzle leak not identified, boric acid accumulation not adequately removed
- 2002 Significant corrosion discovered at nozzle 3, minor damage to nozzle 2

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Probable Cause

PWSCC cracking in the CRDM nozzle interface at the J-groove weld due to material susceptibility in the presence of a suitable environment resulted in:

- CRDM nozzle crack initiation
- CRDM nozzle crack propagation to through-wall leak
- Boric acid corrosion of low-alloy steel head

Root Cause

Boric Acid Corrosion Control and Inservice Inspection programs and program implementation regarding the RPV head resulted in:

- Plant not identifying the through-wall crack/leak during prior outages
- Plant returning to power with boron on the RPV head after outages
- Plant not identifying degradation of RPV head base metal during 12RFO

Contributing Cause

Environmental conditions, cramped conditions due to design and high radiation at the RPV head resulted in:

- Plant not identifying the through-wall crack/leak during prior outages
- Plant returning to power with boron on the RPV head after outages
- Plant not identifying degradation of RPV head base metal during 12RFO

Contributing Cause (continued)

Equipment conditions due to uncorrected CRDM flange leakage resulted in:

- Plant not identifying the through-wall crack/leak during prior outages
- Plant not identifying degradation of RPV head base metal during 12RFO

Leaking Nozzle Material Heat

All through-wall leaks at Davis-Besse are from material heat that has a history of leaks at another plant:

- Heat M3935 accounts for greatest number of nozzle leaks in B&W plants
- Nozzles 1, 2, and 3 had through-wall flaws and are all from heat M3935

Crack Length versus Leakage

Through-wall cracks (above the weld) at Davis-Besse are longer than reported at other plants

- Less than 1 inch at all other plants
- 1.1 inches at nozzle 2
- 1.2 inches at nozzle 3

Leak rate increases significantly as crack lengthens

Safety Significance Assessment

Ken Byrd

FENOC Supervisor Nuclear Engineering

Safety Significance Assessment

Accident Analysis Assumptions

- Hole size 50% larger than exposed cladding area
- CRDM Number 3 assumed to eject

Accident Analysis Results

- Adequate core cooling could be established and maintained for the long term
- Reactor could be placed and maintained in safe shutdown condition
- Integrity of containment not compromised

Safety Significance Assessment

Stress Analysis of As-Found Head Condition

- Three-dimensional finite element stress analysis of wastage area and reactor pressure vessel head
- Failure criterion set as maximum strain of 11% through the thickness of the clad
- Results verified by an independent analysis
- Results are that the degraded cavity would maintain its integrity in excess of twice the transient loads

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Safety Significance Assessment

Risk of As-Found Condition

- Considers probability of failure at pressure less than calculated maximum
- Core damage frequency (CDF) estimated to be 1.0×10^{-5} /yr
- Large early release frequency (LERF) estimated to be 1.4×10^{-8}/yr
- Public health risk is approximately 0.56 Person-Rem / yr

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Concluding Remarks

491st ACRS Full Committee: Advanced Reactor Research Plan

April 11, 2002

John H. Flack
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Outline

- Introduction
- Advanced Reactor Research Plan
- Technical and Potential Safety Issues
- Summary and Next Step

NRC Advanced Reactor Research Plan

Overall Objective: To build an advanced reactor research infrastructure.

- Identify key research areas and activities.
- Identify associated technical and safety issues
- Identify necessary methods, tools and expertise
- Determine research outputs and outcomes.
- Schedule milestones and determine resource needs.
- Set research priorities.
- Communicate research objectives to stakeholders.

NRC Advanced Reactor Research Plan

Phase 1 Identify Infrastructure Needs:

- Workshops on Adv Reactor Safety and Research Issues
- Pre-Application Reviews (e.g., MHTGR, PBMR, GT-MHR)
- Domestic and International Exchange of Research and Experience
- Training and Seminars
- Stakeholder Interactions

Phase 2 Identify Significant Technical Issues and User Needs

- Prioritize and Streamline Research Needs
- Utilize PIRT Process
- Interact with Stakeholders
- Establish Working Groups

Phase 3 Integrate, Implement and Maintain

Advanced Reactor Research Plan

Infrastructure needs basically includes:

- Staff Expertise and Skills
- Analytic Tools and Analysis Methods
- Facilities and Contractor Support

Scope: Pebble Bed Modular Reactor (PBMR)
Gas Turbine-Modular Helium Reactor (GT-MHR)
International Reactor Innovative and Secure (IRIS)
Westinghouse AP-1000

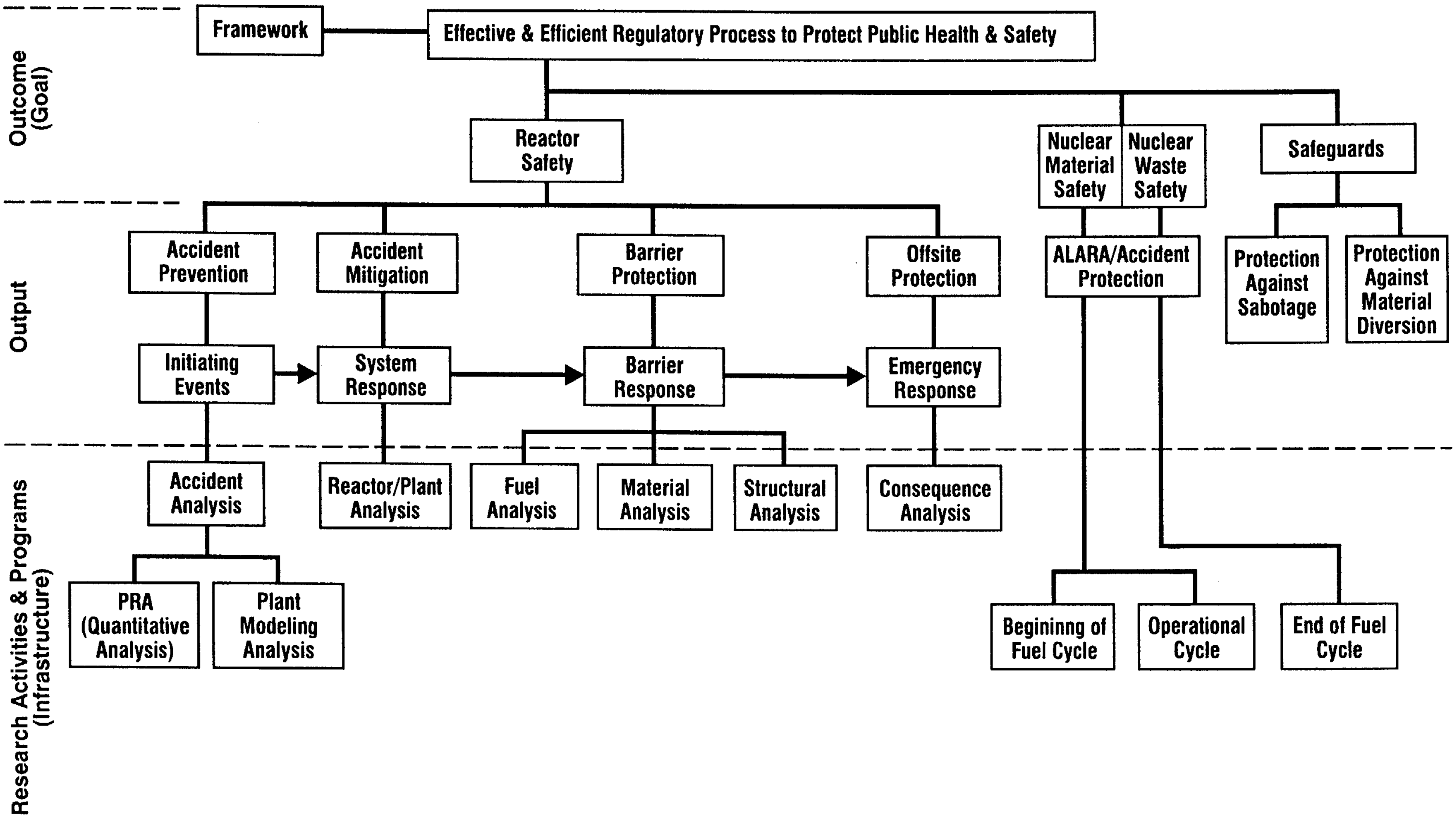
Structure: Built around key research activities and associated technical areas

Advanced Reactor Research Plan

Format of Technical Areas:

- Background
- Why? - Objectives
- What? - Planned activities
- How? - Application of Research Results

Advanced Reactor Research Infrastructure
Key Research Areas and Areas for Examination



March 8, 2002

Technical Areas

Framework

Reactor Safety:

- Probabilistic Risk Assessment
- Human Factors
- Instrumentation & Control
- Thermal-Hydraulic Analysis

Technical Areas
(continued)

<u>Reactor Safety (continued)</u>:

- Nuclear Analysis
- Fission Product Transport (Source Term Analysis)
- Fuels Analysis
- Materials Analysis
- Structural Analysis
- Consequence Analysis

Technical Areas
(continued)

Material Safety and Waste Safety:

- Nuclear Analysis
- Fuel Fabrication
- Transportation and Storage
- Waste Disposal
- Personnel Exposure Control

Safeguards:

- Material Diversion Safeguards

Technical Issues

- PRA Quality (Models and Data)
- HTGR Fuel Performance (Models, Testing, Data)
- HTGR Fuel Fabrication (Manufacturing Specifications)
- Material Performance (High Temperature, Graphite, Concrete)
- Passive System Reliability (DHR, Shutdown)

Technical Issues
(continued)

- Thermo-Fluid Dynamics
- Fission Product Transport (air/moisture intrusion)
- Reliability and Performance of I&C
- Role of the Operator
- Nuclear Analysis (reactivity insertion and shutdown capability)

Potential Safety Issues

- Accident Selection (Licensing Basis Events)*
- General Design Criteria (Safety Classification of SSCs)*
- Containment vs. Confinement (Source Term)*
- Emergency Planning (Evacuation Infrastructure)*
- Fuel Fabrication (Quality Control)*

(*potential policy issue)

Potential Safety Issues
(continued)

- Codes and Standards (Material & Structural)
- Testing and Validation (Fuel and Systems)*
- System Interactions (Multi-module)
- Reactivity Control and Shutdown
- Worker Risk (e.g., emission of ^{110m}Ag)

(*potential policy issue)

Research Products

- Technical basis or criteria for issue resolution.

- Independent confirmation of licensee calculations.

- Technical report that supports a safety evaluation.

- Identification of safety issues or other regulatory action technical report that would describe follow on actions.

- Regulatory guidance.

Summary and Conclusions

- Plan identified potential Infrastructure needs across technical areas.
- Plan focus is on the role of research:
-

  - to effectively and efficiently support the licensing process
  - to establish a technical basis for regulatory decision-making
  - to improve NRC's knowledge and understanding of new phenomena, safety margins and failure points.

- Next phase will focus user needs and priorities.
- Expect additional ACRS Interactions
- Transmit plan to Commission in Fall 2002

United States Nuclear Regulatory Commission

Advanced Reactor Research Plan

Presented By

Farouk Eltawila, Director

Division of Systems Analysis

and Regulatory Effectiveness

Office of Nuclear Regulatory Research

April 11, 2002

Overview

- The Advanced Reactor Research Plan Is Intended to Identify Data and Tools Needed to Support Regulatory Decisions
  - Is Not Intended to Identify Research to Be Done by NRC
- Applicants Would Be Responsible for Providing the Data to Support Regulatory Decisions on Safety Cases of a Particular Reactor Designs
- Decision on Significant Safety Issues Will Be Technically Sound and Utilizes a Blend of Deterministic, Risk-informed, Performance-based Analyses; Defense-in-depth; Safety Margins; Uncertainties Analysis; and Policy
- Establish Additional Risk Metrics to Supplement CDF & LERF
  - Build on Option 3 Framework, Develop Guidelines and Criteria That Will Be Used to Formulate the Regulations and Associated Regulatory Guides for Advanced Reactors
    - → The Regulations Will Be Technology-Neutral
    - → The Regulatory Guides Will Be Reactor-Specific

Objectives

- Objectives of the Advanced Reactor Research Plan
  - Determining the Critical Information Needed to Establish the Safety Standards New Reactor Designs Must Meet
  - Explore Issues of Large Uncertainties
  - Identifying Need for Independent Tools and Data
  - Identifying and Prioritizing Key HTGR Safety and Licensing Issues
    - Understanding the Status of International HTGR-Related Research and Experience
    - Identifying Research Topics for Future Cooperative Efforts to Ensure Optimum Mutual Benefits and to Off-set Costs
    - Collaborate Agreements With European Community and DOE
- Issues Identified in Dr. Powers Trip Report Are Included in the Request for Additional Information We Sent to Exelon, and Were Taken Into Consideration During the Development of the Advanced Reactor Research Plan

Key Elements

- Risks Associated With Plant Design Features and Activities That Are Not Examined in Typical LWR PRAs
  - Online Fuel Handling and Storage Operation
- Fuel Fabrication Quality and Performance Under Normal Plant Operation and Accident Conditions Is Critical for PBMR System
- Assess Temperature and Irradiation, Helium Impurity and Graphite Dust Effects on Materials Used in PBMR and GT-MHR
- Address Waste-related Issue in the Early Part of the Program to Ensure Adequate Data and Tools Are Available
- There Are Several Complex Technical and Policy Issues, Which We Will Seek Commission Guidance
  - Policy Issues Related to Fuels, Source Term, and Containment
  - Policy Issues Related to Risk Acceptance Criteria for New Reactor Designs and Multiple Modules- Site

MRP Update to ACRS April 11, 2002

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

EPRI MRP

Outline

- Alloy 600/82/182 Strategic Plan
- Crack Growth Rate Update
- Risk Assessment/Probabilistic Fracture Mechanics
- Response to Davis-Besse Issue

EPRI MRP

MRP Strategic Plan to Manage Alloy 600/82/182 Corrosion

Alloy 600/82/182 Strategic Plan

- Problem Statement
- Goal/Mission
- Approach to Address Stated Problem
- Roles of Industry Groups/Stake Holders
- Strategy and Approach for Each of 5 Areas
  - Area 1: Alloy 600/82/182 Primary System Butt Weld
  - Area 2: RPV Head Penetrations, Near-Term
  - Area 3: RPV Head Penetrations, Long Term
  - Area 4: All other Alloy 600/82/182 Locations (except SG tubes)
  - Area 5: Alloy 600 Management Guideline for all locations
- Inspection Plan for RPV Head Penetrations

Crack Growth Rate for Alloy 600 Nozzle Material

MRP Crack Growth Rate Approach

- Goal is to establish appropriate CGR guidelines for generic application in nozzle base material
- Involvement of MRP 'Expert Panel' (includes ANL/NRC) is ongoing in refining approach
- Crack growth database has been consolidated, but some key data is still under review
- A revised MRP Crack Growth Rate Curve was proposed and presented to NRC in February 02
- Implications of Davis Besse findings are being assessed as root-cause details become available

Ongoing Work

- CGR curve can be used for deterministic evaluation of real (ID) or hypothetical (OD) flaws in a RVH nozzle on a plant-specific basis (factor of 2X for above weld circ flaws)
- CGR data for base material feeds directly into the probabilistic risk assessment
- Expert Panel working to screen data and develop a recommended approach for the weld metals (Alloy 182/82)
- Research is being initiated by EPRI (DOE/NEPO program)
- MRP will continue to update NRC/ACRS on further CGR developments

Overview of Generic Industry Risk Assessment

Risk Assessment Methodology

- Approach
  - Predict probability of leakage using industry leakage experience and the Weibull model
  - Compute probability of nozzle ejection considering
    - Initiation and growth rate of circumferential flaws above J-groove weld
    - Probability of leak detection (qualified visual or NDE)
    - Growth to critical flaw size
  - Compute probability of core damage considering
    - Probability of nozzle ejection
    - Conditional Core Damage Probability (CCDP) for SBLOCA or MBLOCA
  - Assess potential effect of collateral damage on CCDP

PFM Analysis

- Key elements
  - Probability of Leakage
    - Simulated in Monte Carlo Model
  - Fracture mechanics modeling for Stress Intensity Factors
    - Through-Wall Cracks
    - Part Through Wall Cracks
    - Multiple Flaw Initiation
  - Stress Corrosion Crack Growth Statistics
  - Effect of Inspections
    - Inspection Interval
  - Inspection Reliability

Preliminary Results (Delta CDF)

- The increase in CDF per year for a high temperature plant is the product of the
  - Probability of nozzle ejection after first inspection < 1 E-3
  - CCDP in the event of SBLOCA or MBLOCA < 5 E-3
- The bounding delta CDF is (<1E-3)(<5 E-3) < 5 E-6
- Collateral damage is not expected to be a significant contributor to CDF
- Most plants are expected to be well below 5 E-6/year

Impact of Davis-Besse

- Current MRP Risk and Safety Analysis Work
  - Will be updated to reflect Davis-Besse situation
  - Preliminary assessment
    - PFM evaluation model and results not significantly affected
      - Gap elements represent vessel wall constraint opposite crack opening
        - Gaps could be adjusted to address affect of vessel wastage if applicable
      - No back-wall constraint assumed in part-through-wall crack model, therefore vessel wastage not a factor for this part of calculation

Response to Davis-Besse Issue

MRP Actions to Address DB Corrosion

- Conference calls with staff – 3/7 & 8
  - Questions on generic implications
- EPRI staff person and contractor to Davis-Besse 3/7
  - Participate on Root Cause Team
- Distribute MRP survey questions to industry prior to bulletin – 3/8
- Conference call with staff – 3/13
  - Status of actions
- Meeting with staff – 3/19
  - Provide preliminary assessment of survey
- Provide assessment results to staff – 3/28
  - Specific responses, summary table

Industry Survey

- Based on Initial Information from Davis-Besse
  - 3 potential root causes
    - Leakage from sources above head (flanges, etc.)
    - Leakage of reactor coolant through flaws in penetration nozzles
    - Previous stable accumulation of boric acid deposits wetted by leakage through flaws in penetration nozzles
- On-going Davis-Besse Root Cause Work

Industry Survey (cont'd)

- 4 Questions
  - At most recent inspection
    - sufficient visual examination over 100% of the head
      - detect external surface corrosion or accumulation of boric acid crystals?
    - If visual inspection < 100%/some way hampered
      - Confident no external head corrosion?
    - If UT/other non-visual approach was used
      - examination capable of detecting corrosion of the low alloy steel head material ?
      - was examination full-length of nozzles to the top of the head?
  - For plants with spring 02 outages
    - plans to show no significant boric acid corrosion?
- Responses Received from all US PWRs

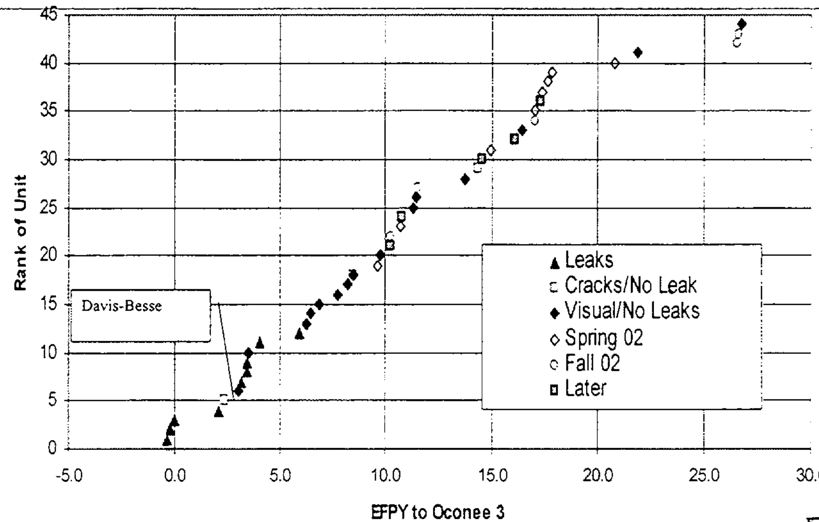
Assessment

- Acceptance Criteria
  - Category 1
    - At most recent outage 100% bare-metal VT of RPV head and region above head
      - No boric acid on head and none above head
  - Category 2
    - During Category 1 examination,
      - Boric acid accumulation detected
        - Boric acid deposits removed, head inspected, source determined and corrected
  - Category 3
    - Bare-metal inspection limited/not able to be performed
      - Plant history and above head inspections show no evidence of leakage

Assessment (cont'd)

- Acceptance Criteria (cont'd)
  - Category 4
    - Bare-metal inspection limited/not able to be performed
      - Above-head inspections indicate boric acid leakage, but leakage managed
        - None reached outer surface of head, OR
        - Affected area(s) cleaned and inspected
  - "Other" Category
    - Bare-metal inspection limited/not able to be performed
      - Above-head inspections indicate boric acid leakage, and leakage may have accumulated on outer surface of head
    - Or, plant situation does not specifically fit first 4 categories

MRP Nozzle Cracking Susceptibility Ranking



EPRI MRP

Survey Results

- Evaluation approach in light of Davis-Besse event:
  - Plants assigned to Categories based on reviews of survey responses
  - Combined with plant-specific responses to NRC Bulletin 2001-01
  - Plant categorizations need to be considered in light of:
    - Past inspection experience and current inspection plans
    - Additional factors, e.g., insulation configurations
    - Root Cause is in progress

EPRI MRP

Survey Summary

Boric Acid Deposit Status		MRP-48 Ranking for Head Nozzle PWSCC: < 5 EFPYs (Highest Susceptibility)	5 – 10 EFPYs	10 – 15 EFPYs	15 – 20 EFPYs	20 – 30 EFPYs	> 30 EFPYs (Lowest Susceptibility)
Other		Davis-Besse		Beaver Valley 1	Palo Verde 2		Cook 1 Wolf Creek Watts Bar 1
No Deposits on Head Per Review of Above-Head Leak Sources and Maintenance	Category 4		Point Beach 2	Calvert Cliffs 2 Indian Point 3 Point Beach 1 St. Lucie 1 Millstone 2 (R)	Diablo Can. 2 Palo Verde 1 Palo Verde 3	Indian Point 2 Diablo Canyon 1	Byron 2 Callaway McGuire 1 Millstone 3 STP 2 Vogtle 2 Summer Byron 1 Sequoyah 1 STP 1 Harris
No Deposits on Head Per Review of Above-Head Leak Sources and Maintenance	Category 3			San Onofre 3 San Onofre 2 Ginna	Fort Calhoun ANO 2 Salem 2	Prairie Island 1 Palisades	Coman. Peak 1 Seabrook Catawba 1 Braidwood 2 Coman. Peak 2 Sequoyah 2 Braidwood 1 Catawba 2
100% Bare-Metal Insp. – Returned to Service with Clean Head	Category 2	North Anna 2 (R) Oconee 3 (R) TMI 1 (R) North Anna 1 Surry 1 (R) Oconee 2 (R) Robinson 2 ANO 1 (R) Oconee 1 (R)	Crystal River 3 (R)				
100% Bare-Metal Insp. – Returned to Service with Clean Head	Category 1	Surry 2	Turkey Point 4 Waterford 3 Calvert Cliffs 1 Cook 2 Farley 1 Turkey Point 3 Farley 2	St. Lucie 2 Salem 1	Beaver Valley 2	Prairie Island 2 Kewaunee	Vogtle 1 McGuire 2

Summary

- Plants at <10 EFPY will have inspected by end of Spring 2002 (highest ranked 20 units)
  - Reasonable assurance of no significant corrosion of the head or CRDM leakage
- 34 out of 45 plants <30 EFPY will have performed inspections by Spring 2002
  - 5 Fall 2002
  - 6 Spring 2003

Recent Experience

- Leakage Detection and NDE Results
  - All 34 leaking penetrations were detected by visible evidence of boric acid during visual exams of the top of the head
  - A total of 203 nozzles have been inspected by NDE at the nine plants where leaks have been found
    - NDE confirmed the through-wall defects in all 34 penetrations which leaked
    - NDE did not detect through-wall defects in any of the additional 169 penetrations which were inspected to assess "extent of condition" or to disposition "masked conditions" at the top of the head
  - Four plants without evidence of leakage have inspected nozzles by NDE. No through-wall defects were found.

Recent Experience (cont'd)

- No significant wastage adjacent to leaks
  - In the 31 penetrations that have been shown to have leaked (other than Davis-Besse), no evidence of significant corrosion or wastage of the top of the head adjacent to leaking nozzles has been observed
  - No evidence of significant wastage adjacent to through-wall defects (i.e., in the head annulus) has been observed during nozzle repairs at plants which were repaired with the temper-bead weld method
    - Temper-bead repair method allows portion of area adjacent to through-wall flaw to be observed prior to repairs
    - Weld prep area is penetrant tested as part of repair procedure

MRP Response Plan

- Ongoing Activities
  - Review of Davis-Besse 'root-cause' analysis when available for generic implications, using November 2001 Boric Acid Corrosion Guidebook
    - Initial letter report
      - Need to assure that boric acid build-up doesn't occur
  - Review early 90s Owners Group work relative to RPV head corrosion/wastage – determine applicability in light of above
  - Incorporation of Davis-Besse lessons learned into future MRP inspection guidance

Conclusions for RPV Head

- Bulletin 2001-01 inspections can identify leakage prior to significant head wastage
  - Plants that detect through-wall flaws need to ensure that wastage has not occurred
- All plants have current programs for managing leak sources
  - Response to GL 88-05
  - Bulletin 2002-01 (on-going)
- MRP is actively addressing the issues

Risk-Informed Inservice Inspection of Break Exclusion Region Piping

Stephen Dinsmore	NRR/DSSA/SPSB
Andrea Keim	NRR/DE/EMCB
Yueh-Li Li	NRR/DE/EMEB
Syed Ali	RES/DET/ERAB

ACRS Presentation
April 11, 2002

BACKGROUND

The PRA Implementation Plan Included the following Guidance for Risk-Informed Decisionmaking:

- General RG and SRP
  - RG 1.174 and SRP Chapter 19
- Four Application Specific RGs and SRPs
  - Technical Specifications (RG 1.177 and SRP 16.1)
  - Inservice Testing (RG 1.175 and SRP 3.9.7)
  - Graded Quality Assurance (RG 1.176)
  - Inservice Inspection (RG 1.178 and SRP 3.9.8)

REGULATORY APPROACH RI-ISI

Issued Regulatory Guidance (for Trial Use) and SERs on Industry Topical Reports (TR)

- RI-ISI Regulatory Guide 1.178, "An Approach for Plant-Specific Risk Informed Decisionmaking Inservice Inspection of Piping," September 1998
- Standard Review Plan Section 3.9.8, "Standard Review Plan for the Trial Use for the Review of Risk-Informed Inservice Inspection of Piping," September 1998
- SER for WOG TR issued December 1998
- SER for EPRI TR issued October 1999

Current Status of RI-ISI Reviews

- Risk-informed inservice inspection (RI-ISI) has been one of the most successful risk-informed initiatives
- Number of plants expected to implement RI-ISI programs: 99
  - Submittals through 2001: 46
  - Anticipated submittals for 2002: 42
  - Anticipated submittals after 2002: 11
- Number of plants that have submitted RI-ISI programs: 50
  - Based on EPRI methodology: 37
  - Based on WOG methodology: 13
- Number of Plants approved by the NRC: 46
- Number of Plants currently under review: 4

CURRENT ACTIVITIES

- Extension of EPRI and WOG RI-ISI methodologies to BER piping
  - Staff review of EPRI submittal almost complete
  - Staff review of WOG submittal to be reviewed later this year

OBJECTIVE OF ISI

- The objective of the ISI program is to identify degraded conditions that are precursors to pipe failures
- Regulatory Requirements for ISI are specified in 10 CFR 50.55a(g)
- 10 CFR 50.55a(g) references ASME Code, Section XI, for ISI requirements

BREAK EXCLUSION REGION

- General Design Criterion 4 (GDC 4)
- SRP 3.6.2 (Branch Technical Position MEB 3-1) - postulating pipe break location, dynamic effects, protection criteria
  - Pipe breaks not postulated in BER if criteria satisfied including augmented ISI of piping welds
  - BER generally in containment penetration area between the containment isolation valves
- MEB 3-1 requires 100% volumetric examination of all BER pipe welds (references ASME Code, Section XI)

EPRI-ISI-TR Methodology Applied to BER Overview

EPRI-ISI-TR as modified by EPRI-BER-TR

- Scope definition (expands the scope to include BER program welds)
- Consequence evaluation (EPRI defined criteria in EPRI-BER-TR)
- Degradation mechanism evaluation (no change)
- Piping segment definition (no change)
- Risk categorization (no change)
- Selection of welds for examination (no change)
- Risk impact assessment (risk criteria applied to BER and ISI + BER)
- Monitoring and feedback (no change)
- Implementation (implementation utilizing 50.59 for BER programs referenced in FSAR)

CONSEQUENCE EVALUATION

Consequence of failure of the BER piping has not been evaluated and protected against in the same way as non-BER piping

- Non-BER piping
  - Pipe failure postulated in design and evaluated using SRP guidelines
  - Mitigative hardware (pipe whip restraints and jet impingement shields) constructed as needed
- BER piping
  - Pipe failures were not postulated in design and potential consequences not evaluated
  - Mitigative devices were not constructed
- Consequence evaluation performed relying on the guidance of SRP 3.6.2

RISK ASSESSMENT

- Quality of PRA
  - Acceptable quality same as for RI-ISI
- Change in Risk Guidelines
  - Current guidelines maintained at system and plant levels AND applied to BER scope

Implementation Process for RI-ISI to BER Piping

- 10 CFR 50.59 determines when license amendment or relief request is required
  - One question is whether there is a change in method of evaluation described in FSAR
  - If method of evaluation has been approved by the NRC for the intended application, amendment not required
- BER SE approves a method of evaluation to replace 100 percent weld inspections with risk-informed percentage
- If BER includes ASME, Section XI, req'd volumetric exams, 10 CFR 50.55a applies

CONCLUSIONS

- Methodology conforms to the guidance provided in RGs 1.174 and 1.178
- Methodology consistent with EPRI Topical Report for RI-ISI
- Changes to BER program, as described in FSAR, may be made under 10 CFR 50.59
- 10 CFR 50.55a still applies, if BER includes ASME Code, Class 1 and 2 piping

Concrete Containment Structures

Effect of Voids

ACRS Meeting

April 11, 2002

Goutam Bagchi
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Concrete Containment Structures

Effect of Voids

- Concrete carries compression as required
  - Maintains reinforcement bars in designed locations
- Reinforcement bars carry all tensile loads
- Post tensioning tendons keep concrete in compression
- Highly ductile (0.2% ultimate elongation) liner plate provides leak-tight barrier

Concrete Containment Structures

Effect of Voids

- Design basis load is internal pressure due to postulated accident load
- Containment structure goes into tension
  - Concrete cracks due to tension
  - Reinforcement bars take all tension loads
  - Liner plate maintains leak tight integrity
- At the shell-mat and shell-dome junctions bending moment puts concrete in compression
- By Code requirement concrete is under reinforced

Concrete Containment Structures

Effect of Voids

- Crushing failure of concrete is prevented by Code
- Redistribution of load around any void provides necessary strength
- Structural Integrity Test (SIT) would reveal locations of unacceptable voids by bulging, spalling or local failure
- Every reinforced concrete containment structure passed SIT satisfactorily the first time

Concrete Containment Structures

Effect of Voids

- Post-tensioning puts the highest load during construction
- Any weakness in concrete shows up at this time
  - Delamination of dome
- A containment weakened by pervasive voids will not pass the SIT

Concrete Containment Structures

Effect of Voids

- Conclusion:
- Design (use of liner plate), high quality of construction and testing before and during service ensure functionality of concrete containment structures

ACRS FULL COMMITTEE MEETING
IN REVIEW OF THE FINAL SAFETY ANALYSIS REPORT FOR LICENSE RENEWAL OF THE TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4

April 11, 2002

RAJ AULUCK
PROJECT MANAGER
NRR

OVERVIEW
Background Information

- Application Submitted by Letter Dated September 8, 2000

- First Westinghouse Pressurized Water Reactor

- 3-Loop Westinghouse PWR, 2 Unit Site, Each Designed for 2300 MWT

- Plant Located in Dade County, Florida City, Florida, Approximately 25 Miles South of Miami

- Unit 3: License Expires July 19, 2012. Requests Renewal Through July 19, 2032.

- Unit 4: License Expires April 10, 2013. Requests Renewal Through April 10, 2033.

OVERVIEW (CONT.)

Turkey Point License Renewal Application Review Schedule

MILESTONE	TARGET DATE	ACTUAL DATE
Receive License Renewal Application	09/11/00	09/11/00
Notice Application Tendered	09/26/00	09/26/00
Complete Acceptance Review of Application	10/10/00	10/04/00
Publish Acceptance Review & Docket Application/Notice of Opportunity for Hearing	10/17/00	10/12/00
Notice of Intent/Notice of Environmental Scoping Meeting	10/20/00	10/24/00
Deadline for Filing Hearing Requests and Petitions for Intervention	11/17/00	11/13/00
Environmental Scoping Meeting	12/06/00	12/06/00
EIS Scoping Period Ends	12/22/00	12/22/00
Staff Complete Environmental RAIs	01/31/01	01/31/01
Staff Complete Safety RAIs	02/05/01	02/02/01
Applicant Responds to Environmental RAIs	03/30/01	03/30/01
Applicant Responds to Safety RAIs	04/23/01	04/19/01
Complete Scoping Inspection	06/22/01	05/25/01
Staff Issue Draft EIS for Comment	07/17/01	06/12/01
Staff Issue SER & Identify Open Items	08/17/01	08/17/01
Public Meeting to Discuss DEIS	07/17/01	07/17/01
ACRS Subcommittee Meeting	09/25/01	09/25/01
ACRS Full Committee Meeting	10/4-5/01	10/05/01
Complete Aging Management Review Inspection	09/14/01	09/14/01
End of DEIS Comment Period	09/06/01	09/06/01
Applicant Completes Responses to SER Open Items	11/16/01	11/07/01
Staff Issue Final EIS	01/29/02	01/11/02
Complete Optional Final Inspection	01/15/02	01/15/02
Staff Issue SSER	02/28/02	02/27/02
ACRS Subcommittee Meeting	03/13/02	03/13/02
Regional Administrator's Letter	03/08/02	02/01/02
ACRS Full Committee Meeting	04/11/02	
ACRS Letter	04/19/02	
Issue SER as NUREG	04/30/02	
Commission Paper W/Staff Recommendations	05/17/02	
Commission Decision	07/17/02	

OVERVIEW (CONT.)

- The applicant has met the requirements for license renewal, as required by 10 CFR 54.29

  - Actions have been identified and have been or will be taken such that there is reasonable assurance that activities will continue to be conducted during the Turkey Point renewal terms in accordance with the current licensing bases for the units.

  - The applicable requirements of 10 CFR Part 51 have been satisfied.

  - Matters raised under 10 CFR 2.758 have been addressed.

WESTINGHOUSE ELECTRIC LICENSE RENEWAL REPORTS

- Five Westinghouse Generic Reports (WCAPs) submitted for NRC Staff Review:
  - WCAP-14574, "License Renewal Evaluation: Aging Management Evaluation for Pressurizers." Final NRC safety evaluation issued on Oct. 26, 2000.
  - WCAP-14575, "License Renewal Evaluation: Aging Management Evaluation of Class 1 Piping and Associated Pressure Boundary Components." Final NRC safety evaluation issued on Nov. 8, 2000.
  - WCAP-14577, "License Renewal Evaluation: Aging Management Evaluation Reactor Internals." Final NRC safety evaluation issued on Feb. 10, 2001.
  - WCAP-14422, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports." Final NRC safety evaluation issued on Nov. 17, 2000.
  - WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants (MUHP-6110)." Final NRC safety evaluation issued on Oct. 15, 2001.
- Application addressed applicability of WCAPs to Turkey Point.

LIST OF OPEN ITEMS

- Scoping for Seismic II over I (II/I) piping systems
- Acceptance criteria for Field Erected Tanks Internal Inspection Aging Management Program
- Scope of Reactor Vessel Head Alloy 600 Penetration Inspection Program
- Reactor Pressure Vessel Underclad Cracking
- Station Blackout (SBO) Issue - New Emerging Issue

OPEN ITEMS

Scoping of Seismic II/I Piping Systems

- Structures, systems, and components (SSC) identified as Seismic II/I should be included in the scope of license renewal and subject to AMRs, as consistent with the scoping requirements of 10 CFR 54.4(a)(2).

- Staff's position has always been that Seismic II/I piping systems whose failure could prevent safety-related systems and structures from accomplishing their intended functions should be within the scope of license renewal.

- In response to the staff's position on Seismic II/I, the applicant included additional components as being within the scope of license renewal and identified appropriate aging AMPs to manage the aging effects that correspond to these components.

OPEN ITEMS

Acceptance Criteria for Field Erected Tanks Internal Inspection

- The applicant uses the Field Erected Tanks Internal Inspection Program to manage aging effects associated with field erected tanks in the auxiliary feedwater and condensate storage system, feedwater and blowdown system, and safety injection system.

- This program is a new aging management program - acceptance criteria and other program attributes were not developed at the time of the draft SER.

- In response, the applicant provided the information related to the acceptance criteria, provisions for additional examinations, and justification for one-time inspections to resolve the issue.

OPEN ITEMS

Reactor Vessel Head Alloy 600 Penetration Inspection Program (RVHPIP)

- LRA submitted prior to issuance of NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."
- Open Item to address whether the RVHPIP was current with Bulletin 2001-01 and the latest industry programs for monitoring for PWSCC in U.S. VHP nozzles.
- Applicant's response to Open Item:
  - referred to FPL response to Bulletin 2001-01 as current status of the program
  - Bulletin response indicates continued participation in the industry-wide program
  - Bulletin response provides the revised susceptibility rankings
  - Bulletin response proposes timely VT-2 examinations of bare metal surfaces
- Information in the applicant's responses to the Open Item and to Bulletin 2001-01 provide a sufficient basis that PWSCC will be managed effectively in the Turkey Point VHP nozzles during the extended periods of operation. Staff considers Open Item 3.9.12-1 to be resolved.

OPEN ITEMS

Reactor Pressure Vessel (RPV) Underclad Cracking

- WCAP-15338 provides Westinghouse Electric's generic evaluation for growth of potential RPV underclad cracks by thermal fatigue. FPL's evaluation of this type of aging effect is addressed appropriately as a time-limited aging analysis (TLAA) covered under the scope Section 4.3 of the application.

- NRC safety evaluation on WCAP-15338 indicated applicant's for renewal of Westinghouse 3-loop plants would need to address two Renewal Applicant Items:

  - applicants would need to indicate whether the number of design cycles and transients assumed in WCAP-15338 bounds the number of cycles for 60 years of operation of the applicant's RPVs under review. (RAI 4.3.2-1)
  - applicants referencing the report would need to ensure that the evaluation of the TLAA is summarily described in the FSAR supplement. (Open Item 4.3-1)

- Conformance with the first item resolved in response to RAI 4.3.2-1 (4/19/01)

- Conformance with the second item resolved in FPL response to Open Item 4.3-1 dated November 1, 2001.

STATION BLACKOUT (SBO) ISSUE

- License Renewal Requirements
  - 10 CFR 54.4(a)(3) requires SSCs that demonstrate compliance with station blackout be included within the scope of license renewal

- Station Blackout Requirements
  - 10 CFR 50.63(a)(1) requires licensees to be able to cope with a SBO
  - Rule requires SBO duration (coping) time be based on four factors
    - first - redundancy of onsite emergency ac power sources
    - second - reliability of onsite emergency ac power sources
    - third - expected frequency of loss of offsite power
    - fourth - probable time needed to restore offsite power

STATION BLACKOUT (SBO) ISSUE (Cont.)

- There is NRC and Industry guidance for determining coping time (as based on the four factors in the SBO rule)
  - NRC Regulatory Guide 1.155
  - NUMARC 87-00

- Conclusions
  - Restoration of offsite power required to demonstrate compliance with SBO rule
  - SSC associated with offsite power required to be within the scope of license renewal

- NRC Position
  - Offsite power circuits between switchyard and safety buses are within the scope of license renewal

COMMENTS FROM A PUBLIC CITIZEN

Discussion of Concerns

- Effects of voids on aging degradation rates and the structural integrity of concrete containment structures
- Effect of design basis hurricane wind speeds and storm waves on safe operation of Turkey Point Nuclear Plant
- Effect of terrorist air attacks on the safety and operability of the Turkey Point nuclear units
- Turkey Point spent fuel capacity

VOIDS IN CONCRETE CONTAINMENTS

- Voids may occur where vibrators cannot reach
- Voids found during construction of NPPs
- Quality assurance requirements would require corrective if voids are detected
- Impact of voids - local stiffness reduction
- Show up as cracks and spalls - ISIT, ILRT
- ISIs, SIT, ILRT of containment structures confirm intended function
- Effect on CPCF/LERF - not significant

CONCLUSIONS

- The staff has completed its review of the Turkey Point License Renewal Application

- All Open Items identified in the SER have been resolved

- Recommendation to the Commission for the renewed license will involve resolution of the SBO issue

- The applicant has met the requirements for license renewal, as required by 10 CFR 54.29.

FPL

LICENSE RENEWAL

TURKEY POINT PLANT

ACRS FULL COMMITTEE MEETING

April 11, 2002

Agenda

FPL

- License Renewal Application Overview
- Open Item Resolution
  - Non-safety/safety related interactions
  - Field-Erected Tanks

FPL

Application Overview

- Safety Review Requirements and Guidance
  - 10 CFR Part 54-License Renewal Rule
  - Draft Standard Review Plan for License Renewal
  - Draft GALL Report
  - Regulatory Guide DG-1047
  - NEI 95-10
- Lessons learned from previous applications, RAIs and RAI responses, and resolution to generic issues factored into application, as available

FPL

Application Overview

- Application follows NRC/NEI Standard Format
  - Chapter 1 - Administrative Information
  - Chapter 2 - SCs Subject to AMR (Scoping/Screening)
  - Chapter 3 - Aging Management Reviews (AMRs)
  - Chapter 4 - Time Limited Aging Analyses (TLAAs)

FPL

Application Overview

- Application follows NRC/NEI Standard Format (continued)
  - Appendix A - UFSAR Supplement
  - Appendix B - Aging Management Programs (AMPs)
  - Appendix C - AMR Process for Non Class 1 Components
  - Appendix D - Technical Specification Changes
  - Applicant's Environmental Report - Operating License Renewal Stage

FPL

Application Overview

- Scoping
  - Safety Related
  - Non Safety which can affect Safety Related
    - Provides functional support
    - Potential for interactions
  - Regulated Events
- Scoping Summary (Tables in Section 2.2)
  - 48 of 81 Systems in Scope
  - 20 of 50 Structures in Scope

Application Overview

FPL

- Screening
  - Purpose - To identify Structures and Components (SCs) which require an Aging Management Review
  - Screening Criteria - SCs which:
    - Support License Renewal System Intended Functions (component level scoping)
    - Perform the intended functions without moving parts or without a change in configuration or properties (passive)
    - Are not subject to replacement based on a qualified life or specified time period (long-lived)

FPL

Application Overview

- Screening (continued)
  - Results in 6 column tables in Chapter 3
    - Mechanical Sections
      - RCS and Connected Systems
      - ESF Systems
      - Auxiliary Systems
      - Steam and Power Conversion
    - Structures and Structural Components Sections
    - Electrical and I&C Section
  - License Renewal Boundary Drawings and UFSAR references provided

FPL

Application Overview

- Aging Management Reviews
  - Application Chapter 3 and Appendix B
  - Same groupings as screening
  - Results presented in 6 column tables
  - Technical criteria for non-class 1 components described in Appendix C

FPL

Application Overview

- Aging Management Review Operating Experience Reviews
  - Reviewed applicable INPO and NRC generic communications and FPL responses
  - Extensive review of plant specific history including:
    - Non-Conformance and Condition Reports
    - Event Response Team and Licensee Event Reports
    - FPL Metallurgical Laboratory Reports
    - Interviews with component/system engineers and plant walk-downs

Application Overview

FPL

- Time Limited Aging Analyses (TLAAs)
  - EQ
  - Class 1 and BOP Fatigue
  - Containment Tendon Relaxation
  - Reactor Vessel Irradiation Embrittlement
  - Wear/Erosion
  - Containment Liner Fatigue
  - Crane Fatigue
- No time bound license exemptions identified

FPL

Application Overview

- Appendix A - UFSAR Supplement
  - Submitted markup with application
  - New Chapter 16
    - AMP descriptions
    - TLAA descriptions
  - Program commitment dates identified

FPL

Application Overview

- Aging Management Programs (AMPs)
  - For each aging effect requiring management, AMPs are identified
  - Description and 10 attributes in Appendix B
  - 3 categories of AMPs
    - 8 Existing
    - 8 Modified Existing
    - 8 New

FPL

Application Overview

- Appendix C - Non-Class 1 Component AMR Process
  - Not required by regulation
  - Submitted to address various prior RAIs
- Appendix D - Technical Specification Changes

FPL

Application Overview

- Environmental Report
  - Turkey Point Closed Cooling Canal System
  - Refurbishment
  - Water Sources
  - Evaluation against alternatives
    - License Renewal option is lowest impact option

Open Item Resolution

FPL

- Open Item 2.1.2-1, Scoping of Seismic II over I Piping Systems
  - Items FPL included in LRA to address non-safety related/safety related interactions:
    - Non-safety related pipe segments and supports at safety-related/non-safety related functional boundaries
    - Piping/component supports for non-safety related mechanical systems with the potential of Seismic II over I interaction
    - Non-safety related conduit, cable trays, supports, and other structural components with the potential of Seismic II over I interaction
    - Design features required to accommodate the effects of flooding
    - Design features required to accommodate the effects of spray, jet impingement, and pipe whip

FPL

Open Item Resolution

- Open Item 2.1.2-1, Scoping of Seismic II over I Piping Systems (cont.)
  - NRC issue was that the effects of pipe whip, jet impingement, physical contact, and leakage due to credible non-safety related pipe failures (beyond current assigned break locations) need to be considered based on industry operating experience
    - May require additional non-safety related pipe segments to be included in scope of license renewal

FPL

Open Item Resolution

- Open Item 2.1.2-1, Scoping of Seismic II over I Piping Systems (cont.)
  - FPL performed an evaluation assuming credible (based on operating experience) non-safety related piping failures beyond the CLB
  - If there was interaction with safety related components as a result of the assumed failures, the non-safety related pipe segments were included in the scope of license renewal

FPL

Open Item Resolution

- Open Item 2.1.2-1, Scoping of Seismic II over I Piping Systems (cont.)
  - Conservative criteria was utilized to determine whether there was an interaction due to the non-safety related piping failure
    - Leakage-
      - Non-EQ safety related electrical/I&C components not qualified for outdoor service were assumed to be affected
    - Pipe whip, jet impingement, physical contact
      - Any safety related components in the proximity of the piping were assumed to be affected
      - Components affected were primarily safety related cable trays and conduit

FPL

Open Item Resolution

- Open Item 2.1.2-1, Scoping of Seismic II over I Piping Systems (cont.)
  - A limited number of pipe segments in five structures were added to the scope of license renewal
  - Aging effects requiring management were identified
  - Pipe segments requiring aging management were included in Chemistry Control Program, Systems and Structures Monitoring Program, and Flow Accelerated Corrosion Program, as applicable

Open Item Resolution

FPL

- Open Item 3.8.4-1, Field Erected Tanks Internal Inspection
  - NRC identified three items to be addressed
    - Specific acceptance criteria for the inspection
    - Provisions for additional examinations if the inspection reveals extensive loss of material
    - Justification for one-time inspection
  - Acceptance criteria and additional examinations
    - For acceptance criteria, any loss of material greater than the tank's corrosion allowance will require corrective action
    - Corrective actions will consider the use of additional volumetric or surface inspections, and followup inspections, if needed

FPL

Open Item Resolution

- Open Item 3.8.4-1, Field-Erected Tanks Internal Inspection (cont.)
  - One-time inspection is justified
    - No significant aging is expected
    - Plant operating experience revealed no incidences of internal degradation for CSTs, RWSTs, or DWST, other than inspections, repairs, and recoating activities for the CSTs attributed to operational practices and inadequate original coatings
    - The DWST was recently inspected as part of an internal tank modification, and the inspection did not identify any degraded coatings or tank corrosion
    - The RWSTs, CSTs, and DWST are externally inspected periodically as part of the Systems and Structures Monitoring Program