NRC PROPOSED BULLETIN TO ADDRESS:

CIRCUMFERENTIAL CRACKING OF REACTOR PRESSURE VESSEL HEAD PENETRATION NOZZLES

Allen Hiser

US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Engineering
Materials and Chemical Engineering Branch

Meeting with
Advisory Committee on Reactor Safeguards
Materials & Metallurgy and Plant Operations Subcommittees

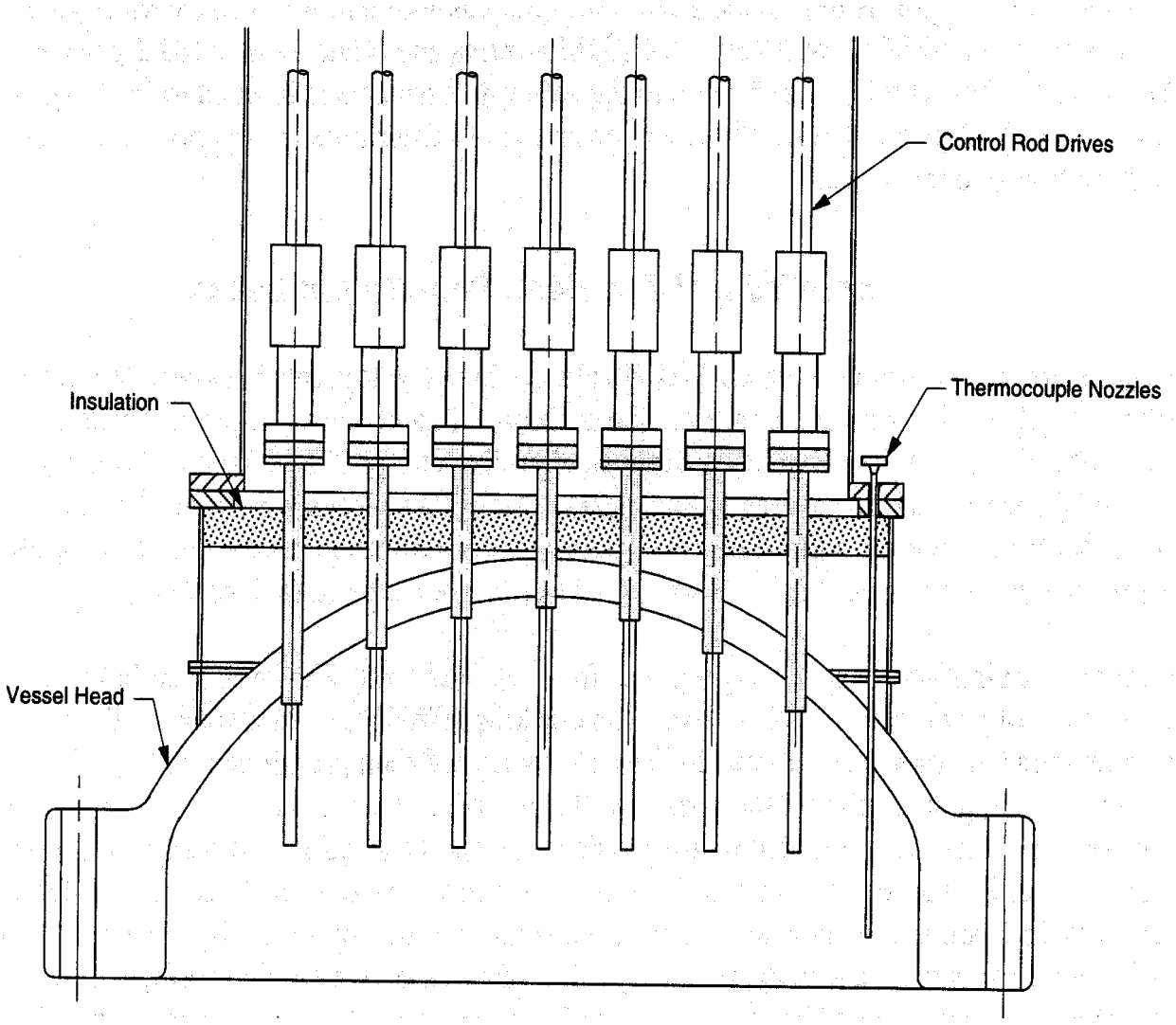
July 10, 2001

BACKGROUND ON CRDM CRACKING HISTORY

- First cracking of CRDM nozzles identified in France in 1989
  - Predominantly axial cracks -- minor circumferential tips
  - Axial flaws will cause leaks, circumferential can cause rod ejection/LOCA

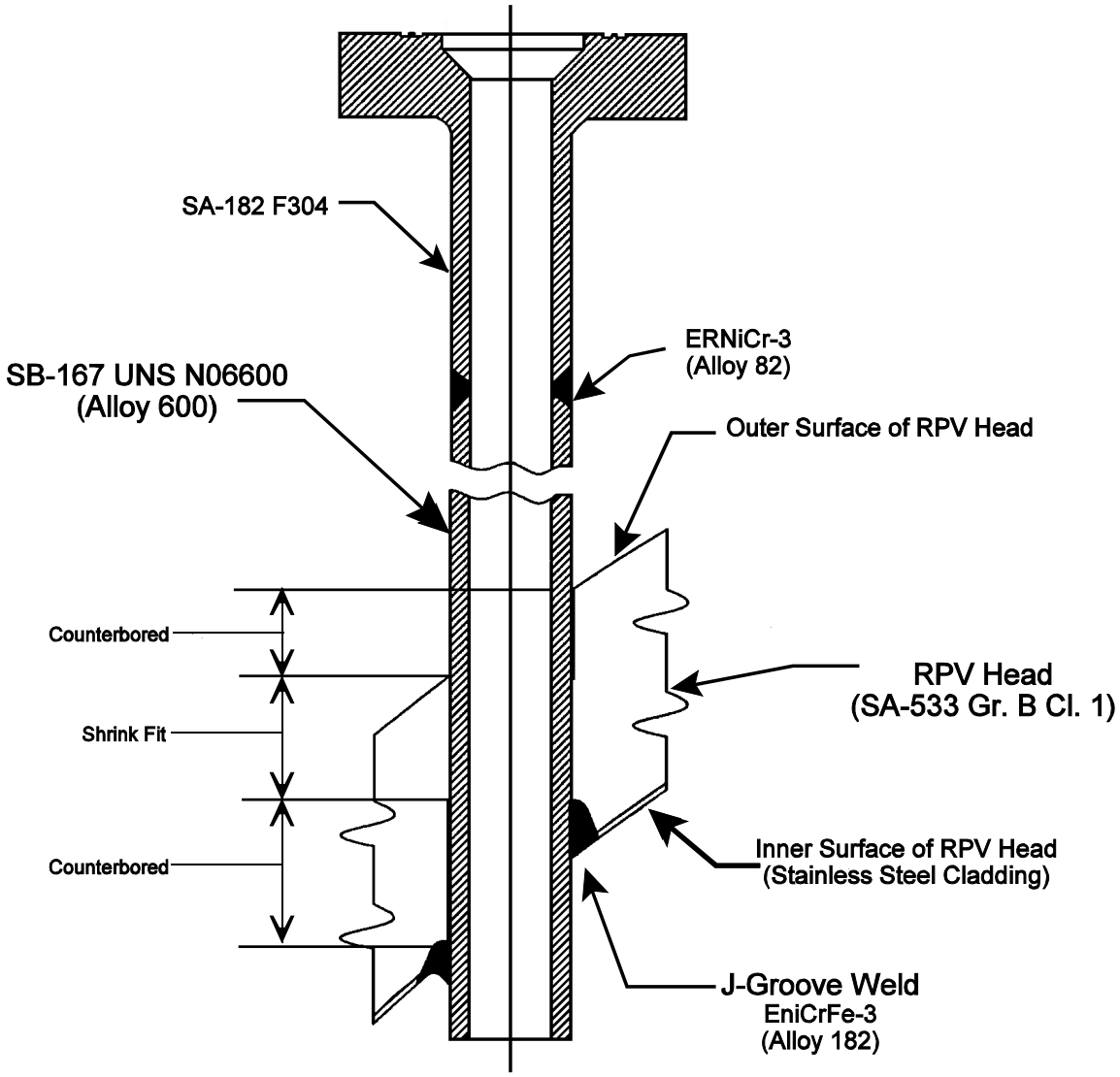
- NRC issued Generic Letter 97-01
  - Integrated industry resolution
  - Used susceptibility models to rank plants
  - Voluntary volumetric examinations at highest ranked plants
  - Boric acid walkdowns to detect throughwall leakage

- Spring 2001 Outages -- Circumferential flaws detected (boric acid deposits)
  - Oconee Unit 3
    - ➜ 2 nozzles, 165° around circumference (throughwall & pin-hole ID indications)
    - ➜ Circumferential flaws detected when repairing axial indications
  - Oconee Unit 2
    - ➜ 1 nozzle, 45° around circumference (0.1 inch in throughwall extent)
  - Chronology of circumferential cracks
    - ➜ Axial cracks in J-groove welds or HAZ allow leakage into annular region
    - ➜ Leakage to vessel head OD may be restricted by interference fit of nozzles
    - ➜ Circumferential cracks initiate on OD and grow in aggressive environment



Typical Reactor Vessel Head - Oconee Unit 1 (Babcock & Wilcox)

Schematic View of B&W Design CRDM Nozzle Area



SAFETY PERSPECTIVE

- Failure of a CRDM nozzle constitutes a LOCA and control rod ejection which are analyzed events
- Existing PRAs indicate a level of risk requiring increased attention
- Worst case crack found at a high susceptibility plant had a remaining ligament safety margin of ≈6 to failure
- No reason to conclude that cracking won't affect additional units
- Timely, effective inspections should provide additional information on extent of the problem and provide confidence that safety is maintained and regulatory requirements are satisfied

OVERVIEW OF STAFF APPROACH

- Public meeting with industry -- April 12, 2001

- Industry report (MRP-44, Part 2) -- May 18, 2001

  - Staff review highlighted technical issues - questions to MRP (June 22, 2001)
  - Public meeting on June 7, 2001

- Proposed Generic Communication

  - Assess compliance with regulations and licensee actions
  - Determine prevalence and severity of PWSCC
  - Formulate future actions

INDUSTRY JUSTIFICATION FOR CONTINUED OPERATION (MRP-44, Part 2)

- Staff requested industry submittal (received May 18)

- Uses susceptibility ranking to assess entire industry (effective time at temperature)
  - 14 plants within 4 EFPY of Oconee Unit 3
  - 25 plants within 10 EFPY of Oconee Unit 3
  - 33 plants within 15 EFPY of Oconee Unit 3
  - 24 plants greater than 30 EFPY of Oconee Unit 3

- Uses Oconee Unit 3 as the benchmark case (cracking and leakage detection)

- Finds that nozzle leaks are detectable in all vessel heads

- Critical remaining ligament is 87° of the circumference (using ASME Code margins)

- Recommendations in industry report
  - Continue inspections for boric acid deposits
  - For plants within 10 EFPY of Oconee Unit 3 and having Fall 2001 outages, perform visual inspection of top head capable of detecting small amounts of leakage

STAFF CONCERNS

- Susceptibility model only provides plant ranking relative to Oconee Unit 3 (not predictive capability) - large uncertainties

- 10 EFPY threshold is not supported by operating experience
  - ANO-1 with axial cracks was > 11 EFPY “behind” according to GL 97-01 modeling
  - 33 out of 69 PWRs are within 15 EFPY of Oconee Unit 3

- Questions regarding adequacy of visual examinations for detection of boron
  - Small quantities of boric acid deposits (< 1 $in.^3$ at Oconee Unit 3)
    - ➜ Variability in interference fits
    - ➜ Tightness of PWSCC cracks
  - Difficulty in identifying leakage from CRDM nozzle cracking
    - ➜ Leakage from Conoseals®, etc. - has head been cleaned ?
    - ➜ Insulation on head -- cannot readily inspect bare metal of RPV head

- Remaining ligament margins do not incorporate time margin and crack growth rate

- Potential for reaching critical crack size before detecting leakage
  - Periodic examination -- no continuous monitoring
  - Inspection under insulation is not adequately addressed

- Postulated accident analysis/risk insights

- Compliance with regulatory requirements

APPLICABLE REGULATORY REQUIREMENTS

- 10 CFR 50.55a
  - References Section XI of ASME B&PV Code
  - Does not permit through-wall cracking
- Plant Technical Specifications
  - Do not permit through-wall leakage
- GDC 14 - Reactor Coolant Pressure Boundary (Appendix A to 10 CFR Part 50)
  - RCPB shall have extremely low probability of abnormal leakage, or rapidly propagating failure and of gross rupture
- GDC 31 - Fracture Prevention of Reactor Coolant Pressure Boundary (Appendix A)
  - RCPB must minimize the probability of rapidly propagating fracture
- GDC 32 - Inspection of Reactor Coolant Pressure Boundary (Appendix A)
  - RCPB shall be designed to permit periodic inspection and testing to assess their structural and leaktight integrity

APPLICABLE REGULATORY REQUIREMENTS

- Criterion IX - Control of Special Processes (Appendix B to 10 CFR Part 50)
  - Special processes such as non-destructive testing shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with codes/standards/specifications/criteria & other special requirements

- Criterion V - Instructions, Procedures, and Drawings (Appendix B to 10 CFR Part 50)
  - Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, including appropriate acceptance criteria

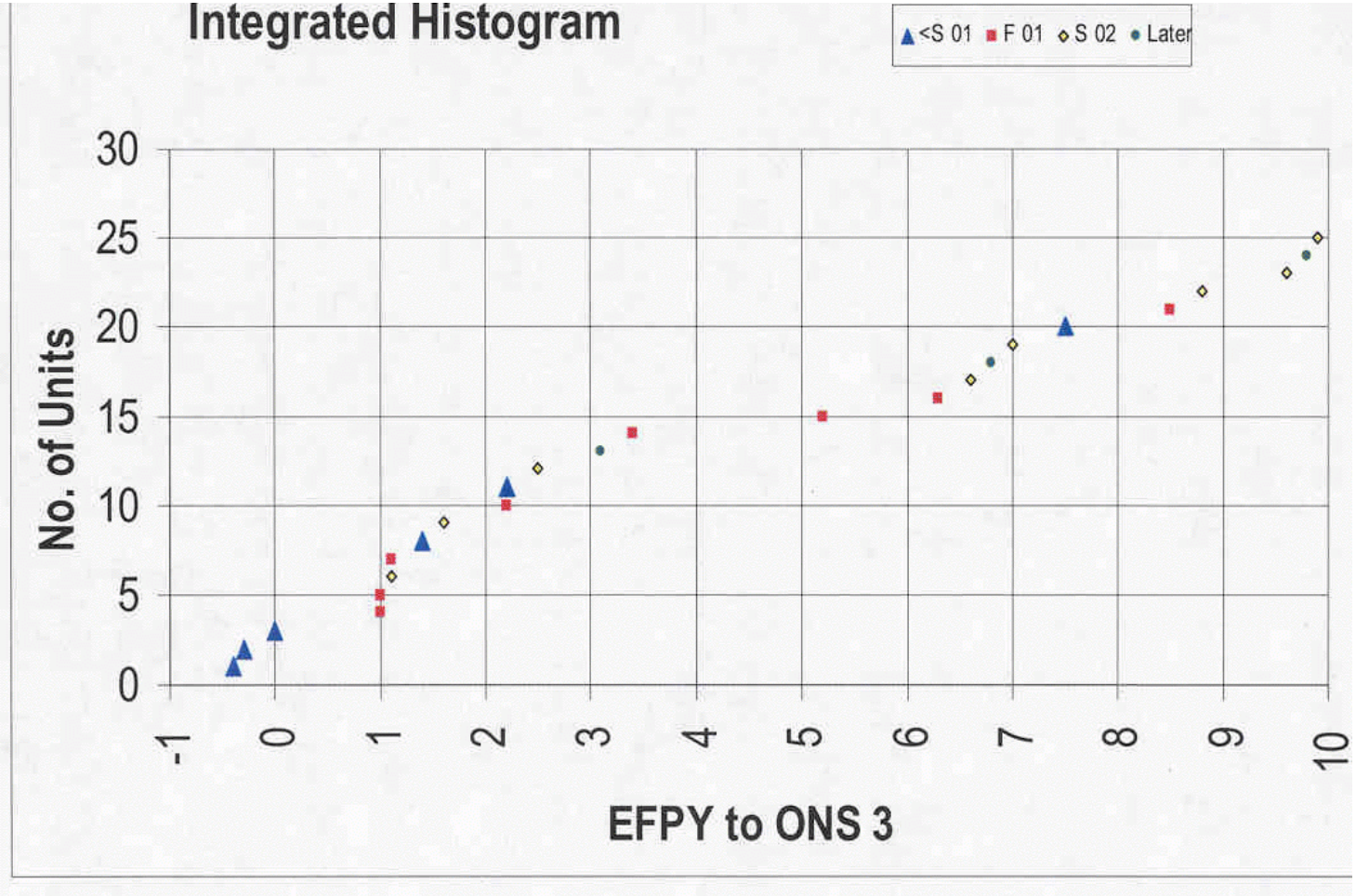
- Criterion XVI - Corrective Action (Appendix B to 10 CFR Part 50)
  - Conditions adverse to quality are promptly identified and corrected
  - Determine cause of condition and corrective action to preclude repetition

STAFF ASSESSMENT OF SUSCEPTIBILITY

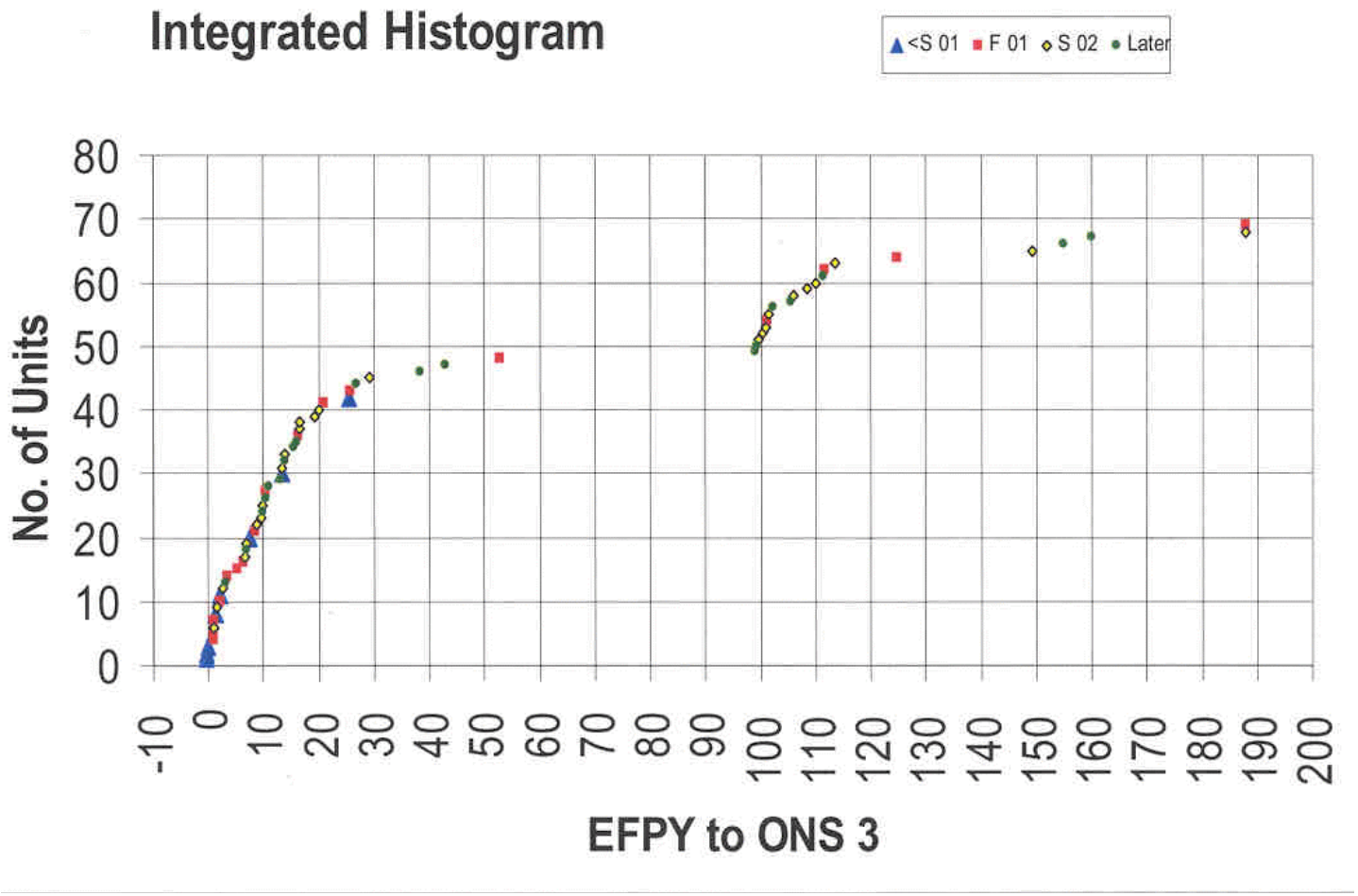
- Subpopulations of PWRs based on PWSCC susceptibility ranking
  - Plants that have identified cracking
    - ➜ PWSCC of nozzles is a documented occurrence
    - ➜ 4 plants total (Oconee 1,2,3 and ANO-1)
  - Plants with HIGH susceptibility to PWSCC (<4 EFPY from the ONS3 condition)
    - ➜ PWSCC of nozzles likely to occur in the near term
    - ➜ 10 plants total
  - Plants with MODERATE susceptibility to PWSCC (from 4 to 30 EFPY of ONS3)
    - ➜ PWSCC of nozzles not likely in short term, but could occur
    - ➜ 31 plants total
  - Plants with LOW susceptibility (balance of plants)
    - ➜ PWSCC of nozzles not likely throughout current license period
    - ➜ 24 plants total

- Verify compliance with regulatory requirements through QUALIFIED examinations
  - Graded approach depending on PWSCC likelihood
  - Examinations of 100% of all VHP nozzles
    - ➜ Based on statistics and no identified preferential cracking tendencies
    - ➜ All VHPs - similar materials, etc., only failure consequences vary

EPRI/MRP RELATIVE SUSCEPTIBILITY RANKINGS



EPRI/MRP RELATIVE SUSCEPTIBILITY RANKINGS



QUALIFICATION OF EXAMINATION METHODS

- VT-2 Visual Examination Qualification
  - Capable of detecting small amounts of boric acid deposits and discriminating deposits from VHP nozzle and other sources
  - Appropriate for Moderate Susceptibility Plants (31 total) - PWSCC of nozzles not likely in short term, but could occur

- Plant-Specific Visual Examination Qualification
  - Plant-specific demonstration that VHP nozzle cracks will lead to deposits on the RPV head (interference fit measurements, etc.)
  - Must be capable of reliable detection and source identification of leakage (insulation, pre-existing deposits, other impediments)
  - Appropriate for High Susceptibility Plants (10 total) - PWSCC of nozzles likely to occur in the near term

- Volumetric Examination Qualification
  - Demonstrated capability to reliably detect cracking on the OD of VHP nozzles
  - Appropriate for plants that have identified cracking (4 total) - PWSCC of nozzles is a documented occurrence
  - Default if Visual Examination cannot be Qualified
  - Applies for any plant finding leakage

PROPOSED INFORMATION REQUEST

Within 30 days of issue date:

! Provide plant-specific susceptibility ranking (data used to determine ranking) and description of VHP nozzles (number, type and materials of construction)

! For plants that have identified leakage or cracking in VHP nozzles

a. Describe the extent of VHP nozzle leakage and cracking (number, location, size, and nature of each crack detected)

b. Describe the inspections (type, scope, qualification requirements and acceptance criteria), repairs, and other corrective actions taken

c. Discuss plans and schedule for future inspections (type, scope, qualification requirements and acceptance criteria)

d. Discuss how the planned inspections will meet regulatory requirements

(1) If inspection plans do not include inspections before end of 2001, provide the basis for concluding that the regulatory requirements will continue to be met until the inspections are performed

(2) If inspection plans do not include volumetric examination of all VHPs, provide basis for concluding that the regulatory requirements will be satisfied

PROPOSED INFORMATION REQUEST

! For plants with susceptibility rankings within 4 EFPY of Oconee Unit 3

a. Describe the VHP nozzle inspections (type, scope, qualification requirements and acceptance criteria) performed in the past 5 years

b. Discuss plans and schedule for future inspections (type, scope, qualification requirements and acceptance criteria)

c. Discuss how the planned inspections will meet regulatory requirements

(1) If inspection plans do not include inspections before end of 2001, provide the basis for concluding that the regulatory requirements will continue to be met until the inspections are performed

(2) If inspection plans include only visual inspections, discuss corrective actions, including alternative inspection methods (for example, volumetric examination), if leakage is detected

PROPOSED INFORMATION REQUEST

! For plants with susceptibility rankings within between 4 and 30 EFPY of Oconee 3

a. Discuss plans and schedule for future inspections (type, scope, qualification requirements and acceptance criteria)

b. Discuss how the planned inspections will meet regulatory requirements

(1) If inspection plans do not include a visual examination at the next scheduled refueling outage, provide the basis for concluding that the regulatory requirements will continue to be met until the inspections are performed

! For plants with refueling or scheduled maintenance outages, provide within 30 days after restart

a. Describe the extent of VHP nozzle leakage and cracking (number, location, size, and nature of each crack detected)

b. Describe the inspections (type, scope, qualification requirements and acceptance criteria), repairs, and other corrective actions taken

PROPOSED REQUIRED RESPONSE

Within 30 days of issue date, submit a written response indicating:

(1) whether the requested information will be submitted

(2) whether the requested information will be submitted within the requested time period

Addressees who choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action that is proposed to be taken, including the basis for the acceptability of the proposed alternative course of action.

Risk Perspective - Failure of Control Rod Drive Mechanism

Mark Reinhart

US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Systems Safety and Analysis
Probabilistic Safety Assessment Branch

Meeting with
Advisory Committee on Reactor Safeguards
Materials & Metallurgy and Plant Operations Subcommittees

July 10, 2001

RISK ESTIMATE

CRDM Rupture: Control Rod Ejection & Medium Break LOCA.

- **Analyzed Events.**
- **Significant Uncertainties.**
- **Potential Collateral Damage.**
- **Potential Operator Recovery Action.**

Assume MLOCA limiting:

- **CDF = IE(f) X CCDP.**
  - **IE(f): Need to understand mechanisms.**
    - **Flaws: welds, materials, chemistry, time, temperature, stresses, synergism.**
    - **Crack initiation.**
    - **Crack propagation.**
    - **Circumferential cracking leading to event.**
  - **For now, assume IE(f) = 1.**
  - **CCDP: <u>E-2 to E-3</u>, not considering collateral effects or additional recovery action.**

CURRENT STATUS

MLOCA Mitigation strategy understood.

Collateral damage and recovery action uncertainties remain.

No anticipated immediate threat to containment.

Estimated level of risk requires management attention.

STATUS OF RESEARCH INITIATIVES ON REACTOR VESSEL HEAD PENETRATION CRACKING

Ed Hackett

US Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Engineering Technology
Materials Engineering Branch

Meeting with
Advisory Committee on Reactor Safeguards
Materials & Metallurgy and Plant Operations Subcommittees

July 10, 2001

Status of RES Initiatives on Reactor Vessel Head Penetrations (VHPs)

" At the request of NRR (June 11, 2001), RES formed an independent group of experts to review technical aspects of the recent VHP cracking occurrences at Oconee and ANO. The expert group has completed their initial assessment as of June 29, 2001. Preliminary conclusions and recommendations will be summarized.

" RES staff and contractors have continued to provide technical support to NRR through on-going programs:

- S Environmentally Assisted Cracking
- S Non-destructive Evaluation
- S Structural Integrity/Fracture Mechanics
- S Probabilistic Risk Assessment

" RES is planning on support of NRR for any VHP inspection oversight activities for Fall 2001 outages

Status of RES Initiatives on
Reactor Vessel Head Penetrations
Independent Group of Experts

- Dr. William Shack (Argonne National Laboratory) - Environmentally Assisted Cracking

- Dr. Steven Doctor (Pacific Northwest National laboratory) - Non-destructuve Evaluation

- Dr. Gery Wilkowski (Engineering Mechanics Corporation) - Leakage Integrity

- Dr. Richard Bass (Oak Ridge National Laboratory) - Structural Integrity

- Mr. Mark Cunningham (Office of Nuclear Regulatory Research, Probabilistic Risk Assessment Branch) - Risk Assessment

Status of RES Initiatives on
Reactor Vessel Head Penetrations
Charter for Independent Group of Experts

- Evaluate technical/safety bases for continued operation

- Evaluate technical issues and provide conclusions/recommendations relevant to:

  - Contents of proposed generic communication
  - Guidance for inspection activities for Fall 2001 outages

- Provide written inputs to NRC by June 29, 2001

- Provide technical support for ACRS meetings (July 10&11, 2001)

Status of RES Initiatives on Reactor Vessel Head Penetrations Independent Group of Experts

Preliminary Conclusions/Recommendations

- Susceptibility Evaluation
  - Significant uncertainty
  - Industry model - time and temperature
  - Other factors (yield strength, fabrication, etc.) can significantly influence susceptibility
  - Best available information for now

Status of RES Initiatives on
Reactor Vessel Head Penetrations
Independent Group of Experts

Preliminary Conclusions/Recommendations (cont.)

" Environmentally Assisted Cracking

- S Annulus region between the head and VHP will likely be a site for concentration of aggressive chemical species
- S Initiation frequency and crack growth rates for this situation are not known, but would likely be more rapid than those observed for PWSCC
- S Initiation at multiple sites around the circumference is likely
- S Crack growth rates in excess of 1 inch/year are possible

Status of RES Initiatives on
Reactor Vessel Head Penetrations
Independent Group of Experts

Preliminary Conclusions/Recommendations (cont.)

" Detection and Characterization of Boric Acid Deposits from Annulus Leakage is Subject to Significant Uncertainties:

- S Interference fits
- S Occlusion of annulus by deposits
- S Quantity and differentiation of deposits
- S Configuration of head insulation
- S Need for plant-specific qualification

Status of RES Initiatives on
Reactor Vessel Head Penetrations
Independent Group of Experts

Preliminary Conclusions/Recommendations (cont.)

" Need for and Reliability and Effectiveness of Volumetric Examinations:

- S Volumetric examinations are indicated for plants with known cracking and would be the preferred inspection method for high susceptibility plants
- S Vendors have current equipment capabilities but not qualified inspection methods
- S Inspections can be effective if adequate pre-qualifications can be performed
- S There will be limitations on the number of qualified industry methods and teams that could be fielded by Fall 2001 outages

Status of RES Initiatives on Reactor Vessel Head Penetrations Independent Group of Experts

Preliminary Conclusions/Recommendations (cont.)

" Potential for On-line Monitoring for Leakage or Cracking:

- S On-line monitoring for leakage or cracking is technically feasible
- S On-line leakage monitoring for certain French plants is accomplished through N-13 monitoring
- S Acoustic emission monitoring has demonstrated potential for identifying cracking in nuclear plant applications
- S Application of either on-line leakage or crack monitoring for VHPs in U.S. PWRs will require development efforts that would be longer-term (beyond Fall 2001 outages)

Status of RES Initiatives on
Reactor Vessel Head Penetrations
Independent Group of Experts

Preliminary Conclusions/Recommendations (cont.)

" Structural Margin:

- S Expert group has verified structural margin calculations by the industry
- S Alloy 600 is capable of tolerating very large through-wall circumferential cracks while still maintaining adequate structural integrity
- S Margin calculations do not consider crack growth

Status of RES Initiatives on Reactor Vessel Head Penetrations

" RES and NRR are developing an integrated technical perspective on the issue through consideration of the expert group reports, industry and staff analyses, and other applicable analyses and data

" The integrated perspective will be documented in an memorandum - expected completion - July, 2001

" The memorandum will be made available to the public upon completion

" Perspectives and recommendations from the expert group have been factored into development of the draft generic communication

" It is not anticipated that continued technical evaluation will have a significant effect on the generic communication, but could influence development of long term programs for dealing with the issue

REGULATORY PROCESS

Tad Marsh

US Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Regulatory Improvement Programs
Operational Experience and Non-Power Reactors Branch

Meeting with
Advisory Committee on Reactor Safeguards
Materials & Metallurgy and Plant Operations Subcommittees

July 10, 2001

GENERIC COMMUNICATION PROCESS

- SECY 99-143 - more rigorous process put in place; work with industry before initiating generic communication

- utilize RIS - endorsement of industry program which satisfies safety concern

- when “compliance exception” cited, staff will do a limited value-impact study to justify request for action (either a bulletin or generic letter)

BULLETIN

- used to address significant issues that also have great urgency
- requests action and/or requests information
- requires written response (10 CFR 50.54(f)) - staff will justify burden relative to safety significance of the issue)
- near term action and one-time response
- interact with industry as time permits; may bypass CRGR if sufficiently urgent
- Commission information paper

GENERIC LETTER

- used to address “routine” technical issues
- requests action and/or requests information
- 10 CFR 50.54(f) will not be invoked unless staff has been unable to obtain information through other means (staff will justify the burden relative to the safety significance of the issue)
- long term action and information gathering
- publish in Federal Register for public comment
- issued only after prior staff interaction with industry and public
- Commission information paper

BULLETIN (BL) OR GENERIC LETTER (GL)

REQUEST ACTION

! assess backfit implications under 10 CFR 50.109

- compliance backfit
  10 CFR 50.109(a)(4)(i)

- adequate protection backfit
  10 CFR 50.109(a)(4)(ii)

- define/redefine what level of protection is adequate
  10 CFR 50.109(a)(4)(iii)

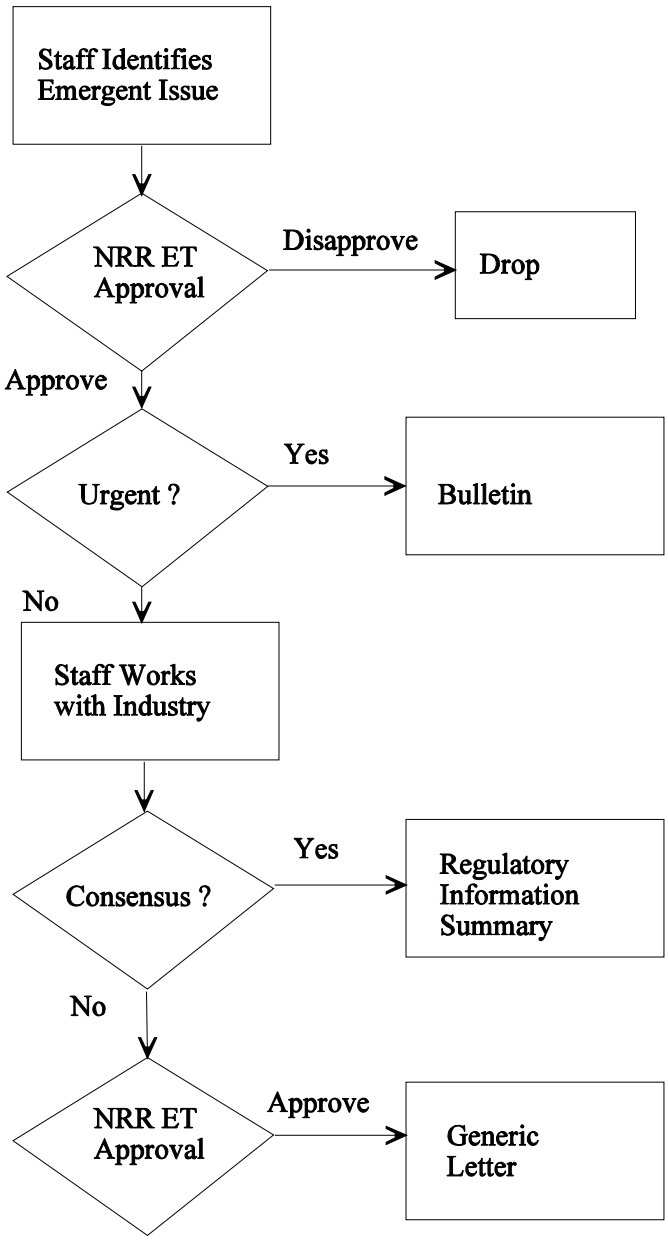
REQUEST INFORMATION

! assess application of 10 CFR 50.54(f)

- verify compliance with licensing basis of vacility; provide statement of need and plans for use

- EDO approval if information not for verifying compliance with licensing basis of facility: justify burden imposed in view of safety significance of issue

Generic Communications Process



MAJOR MILESTONES

- November 2000 — axial cracking discovered at ONS1
- February 2001 — axial cracking discovered at ANO1 circumferential cracking discovered at ONS3
- April 2001 — circumferential cracking discovered at ONS2
- April 12, 2001 — public meeting with EPRI Materials Reliability Program (MRP) to discuss circumferential cracking issue
- April 30, 2001 — IN 2001-05, “Through-Wall Circumferential Cracking of Reactor Pressure Vessel Head Control Rod Drive Mechanism Penetration Nozzles at Oconee Nuclear Station, Unit 3"
- May 18, 2001 — MRP-44/Part 2 report - interim safety assessment of PWSCC of Alloy 600 VHP nozzles and Alloy 182 J-groove welds at PWRs

MAJOR MILESTONES (cont.)

- June 5, 2001 — Commissioner's Technical Assistants briefed
- June 7, 2001 — public meeting with EPRI/MRP to MRP-44/Part 2 report
- June 11, 2001 — RES expert group to review VHP cracking at ONS and ANO
- July 2, 2001 — CRGR pre-brief/ Commissioner's Technical Assistants briefed
- July 3, 2001 — public meeting with NEI and EPRI/MRP representatives to brief them on status of proposed bulletin

BULLETIN PROCESS - REMAINING MILESTONES

- July 10, 2001 — ACRS Materials and Metallurgy Subcommittee briefing
- July 11, 2001 — ACRS Full Committee briefing
- July 12, 2001 — CRGR final briefing
- July 16, 2001 — ACRS letter; CRGR endorsement of draft BL
- July 18, 2001 — Issue Commission information paper
- July 27, 2001 — Issue guidance memorandum to NRR Project Managers
- August 1, 2001 — Issue bulletin