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

August 20, 2001

MEMORANDUM TO: ACRS Members

FROM: Maggalean W. Weston, Senior Staff Engineer
ACRS/ACNW

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY AND ON PLANT OPERATIONS, JULY 10, 2001, ROCKVILLE, MD

The minutes of the meeting on Circumferential Cracking of PWR Vessel Head Penetrations, issued August 20, 2001, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc via Email: J. Larkins
S. Bahadur
H. Larson

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CHAIRMAN

August 17, 2001

MEMORANDUM TO: Maggalean W. Weston, Senior Staff Engineer
ACRS

FROM: F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE MEETING OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY AND ON PLANT OPERATIONS, JULY 10, 2001, ROCKVILLE, MD

I hereby certify that, to the best of my knowledge and belief, the minutes of the meeting on Circumferential Cracking of PWR Vessel Head Penetrations issued August 17, 2001, are an accurate record of the proceedings for that meeting.

F. Peter Ford, Chairman — August 17th, 2001
Date

August 9, 2001

MEMORANDUM TO: F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

FROM: Maggalean W. Weston, Senior Staff Engineer
ACRS

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS SUBCOMMITTEES ON MATERIALS AND METALLURGY AND ON PLANT OPERATIONS, JULY 10, 2001, ROCKVILLE, MD

A working copy of the minutes for the subject meeting is attached for your review. Please review and comment at your earliest convenience.

Attachment:
As Stated

Certified by: F. Peter Ford
August 17, 2001

CERTIFIED

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MATERIALS AND METALLURGY AND PLANT OPERATIONS SUBCOMMITTEES
CONTROL ROD DRIVE MECHANISM CRACKING
ROOM T-2B3, 11545 ROCKVILLE PIKE
ROCKVILLE, MARYLAND
JULY 10, 2001

The ACRS subcommittees on Materials and Metallurgy and on Plant Operations held a meeting on July 10, 2001, with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The purpose of this meeting was to discuss the staff's proposed bulletin to all holders of operating licenses for pressurized-water nuclear power reactors regarding circumferential cracking of reactor pressure vessel head penetration nozzles. The meeting was open to the public. Mrs. Maggalean W. Weston was the cognizant ACRS staff engineer and designated federal official (DFO) for this meeting. There were no written comments provided by the public. The meeting was convened by the Materials and Metallurgy Subcommittee Chairman, Dr. Peter Ford, at 8:30 a.m. on July 10, 2001, and adjourned at 2:47 p.m. that day.

Attendees

Attendees at the meeting included ACRS members and staff, NRC staff, representatives of NEI, EPRI, MRP, and members of the public as follows.

ACRS Members/Staff

P. Ford, Chairman,
J. Sieber, Co-Chairman
G. Apostolakis, Member
M. W. Weston, DFO
M. Bonaca, Member
T. Kress, Member
G. Leitch, Member
G. Wallis, Member
R. Uhrig, Member
S. Rosen, Member

NRC Staff

J. Strosnider, NRR
K. Wichman, NRR
J. Collins, NRR
J. Zimmerman, NRR
M. Kirk, RES
R. Caldwell, NRR
A. Buslik, RES
J. Chung, NRR
A. Hiser, NRR
S. Malik, NRR
D. Jackson, RES
M. McConnell, NRR
T. Colburn, NRR
L. Abramson, RES
E. Chow, RES
E. Hackett, RES
F. Eltawila, NRR
J. Medoff, NRR
L. Marsh, NRR
M. Reinhart, NRR
W. Bateman, NRR
D. Kalinousky, RES
B. Jasinski, OPA
R. Assa, OPA

NEI/EPRI- MRP

L. Mathews, EPRI (SNOC)
A. Marion, NEI
F. Ammirato, EPRI
K. Cozens, NEI
C. Welty, EPRI
J. Bailey, EPRI

There were 25 members of the public in attendance at this meeting. A list of those attendees who registered is attached to the Office Copy of these minutes.

Presentations and Discussion

The presentations to the subcommittees and the related discussions are summarized below. The presentation slides and handouts used during the meeting are attached to the Office Copy of the Minutes.

Chairman's Comments

Dr. P. Ford, Subcommittee Chairman, convened the meeting. He stated that the purpose of the meeting was to discuss the issue of cracking in the control rod drive mechanisms (CRDM). He also noted that this was the first meeting on this issue before the ACRS. The staff requested that the ACRS provide a written opinion following the next full committee meeting on the appropriateness of issues in the bulletin and its licensees on this topic.

Industry Presentation

EPRI-MRP:

The industry's position was presented by Mr. Larry Mathews of Southern Nuclear Operating Company (SNOC) who serves as Chairman of the Alloy 600 Issues Task Group of the EPRI Materials Reliability Project (MRP). Mr. Mathews discussed the following topics:

- Industry Goals
  - Near-Term: Assure Structural Integrity of the Plants
  - Long-Term: Develop Program to Manage PWSCC
- Background of the Head Penetration Issue
- Status of the MRP Program
- MRP Recommendations for the Industry

Mr. Mathews discussed the cracks found at the Bugey-3 plant in France in 1991. He also noted that leaks in Alloy 600 pressurizer instrument nozzles at both domestic and foreign reactors were observed as early as 1986. In 1994, D.C. Cook became the first U.S. reactor to detect and repair axial cracking vessel head penetration. In 1997, the NRC issued Generic Letter (GL) 97-01 to request information to verify compliance with 10 CFR 50.55a and GDC 14 as defined in Appendix A to 10 CFR Part 50, and to determine whether an augmented inspection program was required. The industry responded generically with information that was based on models normalized to the D.C. Cook 2 plant. The recent experience with Oconee Nuclear Station (ONS), Units 1, 2, and 4 and Arkansas Nuclear (ANO), Unit 1 is prompted the bulletin that the NRC staff is proposing.

The industry has drawn the following conclusions:

- Axial cracks alone in CRDM nozzles do not impact plant safety and they are bounded by previously submitted safety assessments (1993/1994). However, throughwall axial cracks can be precursors to circumferential cracking.

- There is reasonable assurance that PWRs do not have circumferential cracking that would exceed structural margins. This is because of Oconee-1 and ANO-1 being in the

highest grouping based on an effective time-at-temperature for their heads. The leaks at these plants were discovered by careful visual examination of their heads. Volumetric examination of other nozzles has revealed only some minor craze cracking. Moreover, the leaks were discovered while plenty of structural margin remained, and several other plants in the highest groupings have examined their heads and found no evidence of leakage.

Mr. Mathews went on to discuss the following ongoing MRP Activities:

- Risk Assessments
- Probabilistic Fracture Mechanics
- NDE Demonstration
- Information and Training Package for Visual Examination
- Flaw Evaluation Guidelines
- Review of Repair and Mitigation Strategies

Mr. Mathews stated that while ensuring the integrity of the plants, the industry wants to work toward developing a program that will enable the utilities to effectively manage PWSCC in their units.

Subcommittee Comments:

During the above discussion, subcommittee members noted the following:

- Dr. Ford asked what had been done to utilize data from the foreign experiences. The industry responded that they had not benchmarked their efforts against foreign plants.

- Dr. Ford questioned the use of D.C. Cook as the only data point in developing the susceptibility models. Mr. Mathews responded that yes, that was the normalization for ranking the plants. The models are probably still pretty good for what we are set to do and they used time and temperature.

- Mr. Leitch stated that the leakage you get would not only be a factor of the cracking, but also the interference fit. If the fit is very tight, you might not get any leakage evidence. Mr. Mathews acknowledged that this was of concern to the NRC staff as well.

- Dr. Ford questioned the assumption that the environment in the annulus between the tube and pressure vessel head was essentially the primary water coolant. This assumption was crucial to the assessment of the circumferential crack growth rates, which relied on existing data on the crack growth rate of Inconel 600 in this environment. Dr. Ford made the suggestion that, given the fact that boron crystals were observed at the mouth of the annulus, boiling had taken place and that a considerably more alkaline environment may exist in this area than that assured in the crack disposition analysis.

<u>NRC Presentation</u>

The NRC presentation included input from both NRR and RES as summarized below.

NRR Staff:

Mr. Jack Strosnider, Division of Engineering, NRR, presented a general overview of the issue from NRC's perspective and discussed the staff's approach to the resolution of the issue. He also commented that the questions raised during the industry's presentation focused on some of the major issues of concern to the staff. In addition, Mr. Strosnider said that he wanted to state up front that the staff does not have all of the answers.

The staff's background information indicated that cracking of CRDM nozzles was first identified in France in 1989. These cracks were predominately axial with minor circumferential tips. Circumferential flaws were subsequently detected at Oconee Units 2 and 3 during the Spring 2001 outages. The staff also made the point that axial flaws will cause leaks, while circumferential cracks can cause rod ejections and loss-of-coolant accidents (LOCAs).

In preparing its integrated response to GL 97-01, the industry used susceptibility models to rank plants. The highest ranked plants conducted voluntary volumetric examinations and performed boric acid walkdowns to detect throughwall leakage.

To continue NRR's presentation, Mr. Allen Hiser, Materials and Chemical Engineering Branch, NRR, who serves as the Project Manager for this issue discussed what the proposed bulletin requests, the information the staff is seeking, and some thoughts behind the approach. This discussion included the following:

- Safety Perspective
- Overview of Staff Approach
- Industry Justification for Continued Operation
- Staff Concerns
- Applicable Regulatory Requirements
- Staff Assessment of Susceptibility
- EPRI/MRP Relative Susceptibility Rankings
- Qualification of Examination Methods
- Proposed Information Request
- Proposed Required Response

Subcommittee Comments:

During the above discussions, subcommittee members noted the following:

- Dr. Wallis summarized that the real issue is whether you can detect enough boric acid in time before there is a circumferential crack which is growing sufficiently to cause problems for all possible situations of interference. He noted also that we have not seen any quantitative analysis of this. The NRC staff agreed with Dr. Wallis and stated that they expect some sort of demonstration. If it cannot be demonstrated then a visual examination is not appropriate. The boric acid must be reliably detectable, and also the source of the leakage must be identifiable.

- Dr. Bonaca raised the concern that significant cleaning of the head is required before you can be really certain you see a leak, and this would exclude many of the plants.

NRC staff responded that this is one of many technical questions to which industry is expected to respond during this information collection phase.

- Mr. Leitch stated that he expected that the other CDRMs at Oconee are perhaps the highest susceptibility areas that we have, because we do not understand why some of them have cracked while others have not. He questioned whether we are simply allowing them to operate on a normal refueling cycle. The NRC staff indicated that the Agency is in the information collection stage and would be asking the plants for their plan and proceed on the basis of the information provided by the plants.

Mr. Mark Reinhart, Probabilistic Safety Assessment Branch, NRR talked about the risk perspectives that have been developed to date. He noted that a rod ejection or a LOCA are the two scenarios of concern from a risk perspective. The discussion covered the current status of the risk effort and the risk estimates.

Subcommittee Comments:

During the above discussions, subcommittee members noted the following:

- Dr. Wallis asked how small does the initiating event frequency have to be in order for the proposed action to take to be commensurate with the risk? This question generated a lot of discussion with no clear answers.

- Dr. Bonaca stated that the conditional core damage probability (CCDP) here in this context is wrong. All that has been done is that the staff has taken the IPEs and looked at the medium LOCA, which is 10^2, 10^3, ignoring the potential consequences to core damage of the rod injection. This should not be ignored. The NRC staff responded that they have noted the comment, and will look at the neutronics and make the calculation to see what it does to that rod ejection.

RES Staff:

Mr. Ed Hackett, Materials Engineering Branch, RES summarized of the ongoing work in RES, including that of an Expert Panel that has been contracted to provide technical advice. He discussed the following topics:

- Status of RES Initiatives on Reactor Vessel Head Penetrations

- Independent Group of Experts
  - Charter
  - Preliminary Conclusions and Recommendations
    - Susceptibility Evaluation
    - Environmentally Assisted Cracking
    - Detection/Characterization of Deposits from Annulus Subject to Uncertainties
    - Reliability and Effectiveness of Volumetric Examinations
    - Potential for Online Monitoring of Leakage and Cracking
    - Structural Margin

Subcommittee Comments:

During the above discussions, Dr. Wallis inquired as to whether the subcommittee would see a presentation on the integrated assessment of everything (the chemistry, and the flow, etc.). The NRC staff stated that it will probably come before the Committee in early 2002.

NRR Staff:

The final portion of the staff's presentation was made by Mr. Tad March, Operational Experience and Non-Power Reactors Branch, NRR, who talked about the generic communication process and the major milestones associated with the proposed bulletin.

During the wrap up period, the subcommittee members came to the following general conclusions.

- The issuance of the bulletin was appropriate and timely.

- There were numerous issues requiring attention, the most significant being:

  - There needed to be a more rigorous treatment of the risk assessment with an expansion to cover, for instance, rod ejection with coincident small break lose of coolant and potential damage to adjacent control rods.
  - A reexamination of the inspection prioritization algorithm. The fact that the incidence of circumferential cracking is relatively rare so far is an indicator that there are other systems governing factors than just operating temperature and time.

  - A reexamination of inspection methods in line with the regulatory requirement of GDC 32. Detection of boron-rich crystals at the mouth of the tube/pressure vessel head does not give an indication of the depth or orientation of the tube cracks, nor does the absence of observable crystals necessarily relate to an absence of circumferential cracks

  - A reexamination of the relevance of crack growth rates obtained in PWR primary water to the determination of ISI intervals for circumferential cracks. It is not at all clear that the environment in the tube/pressure vessel annulus is PWR primary water.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MATERIALS & METALLURGY AND PLANT OPERATIONS SUBCOMMITTEES
CONTROL ROD DRIVE MECHANISM CRACKING
ROOM T-2B3, 11545 ROCKVILLE PIKE
ROCKVILLE, MARYLAND
JULY 10, 2001

-PROPOSED AGENDA-

	SUBJECT	PRESENTER	TIME
I.	Introductory Remarks Subcommittee Chair	F. P. Ford, ACRS	8:30-8:35 a.m.
II.	Industry Perspectives [1]	Larry Mathews, et al, MRP/NEI	8:35-10:15 a.m.
		****BREAK****	10:15-10:30 a.m.
III.	NRC Staff Presentation		10:30-12:30 p.m.
	Introduction	-Jack Strosnider, NRR	
	Technical Discussion & Actions	-Allen Hiser, NRR	
	Risk Perspective	-Mark Reinhart, NRR	
	Staff Perspective Including Input From "Independent Group Of Experts"	-Ed Hackett, RES	
	Regulatory Process	-Tad Marsh, NRR	
	Summary	-Jack Strosnider, NRR	
		****LUNCH****	12:30-1:15 p.m.
IV.	General Discussion and Adjournment		1:15-2:30 p.m.

****

[1]A portion of this meeting may be closed pursuant to 5 U.S.C. 552b(c)(4) to discuss proprietary information.

Note: Presentation time should not exceed 50% of the total time allocated for a specific item. Number of copies of presentation materials to be provided to the ACRS - 35.

ACRS CONTACT: Ms Maggalean W. Weston, mww@nrc.gov or (301) 415-3151.

Thomas Smith at 301–415–7204,or toll free 1–800–368–5642 or e-mail aug@nrc.gov. Further instructions will be sent to you by e-mail or telephone.

Dated in Rockville, Maryland, this 19th day of June 2001.

For the Nuclear Regulatory Commission.

Lynn B. Scattolini,

Director, Information, Records and Document Management Division, Office of the Chief Information Officer.

[FR Doc. 01–16098 Filed 6–26–01; 8:45 am]

BILLING CODE7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Joint Meeting of the ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena, and Reliability and Probabilistic Risk Assessment; Notice of Meeting

The ACRS Subcommittees on Materials and Metallurgy, Thermal-Hydraulic Phenomena and Reliability and Probabilistic Risk Assessment will hold a joint meeting on July 9, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Monday, July 9, 2001—1:30 p.m. Until The Conclusion of Business

The Subcommittees will discuss the proposed risk-informed revisions to 10 CFR 50.46 for emergency core cooling systems. The Subcommittee will also discuss revisions to the framework for risk-informing the technical requirements of 10 CFR Part 50. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittees along with any of their consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittees will then hear presentations by and hold discussions with representatives of the NRC staff and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Michael T. Markley (telephone 301/415–6885) between 7:30 a.m. and 4:15 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: June 21, 2001.

James E. Lyons,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–16093 Filed 6–26–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting of the ACRS Subcommittees on Materials and Metallurgy and Plant Operations July 10, 2001, Notice of Meeting

The ACRS Subcommittees on Materials and Metallurgy and Plant Operations will hold a meeting on July 10, 2001, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, July 10, 2001—8:30 a.m. until 2:30 p.m.

The Subcommittees will discuss the control rod drive mechanism cracking issues. A portion of this meeting may be closed pursuant to 5 U.S.C. 552b(c)(4) to discuss proprietary information. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman and written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore, can be obtained by contacting the cognizant ACRS staff engineer, Ms. Maggalean W. Weston (telephone: 301/415–3151) between 8:00 a.m. and 5:30 p.m. (EDT). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: June 21, 2001.

James E. Lyons,

Associate Director for Technical Support.

[FR Doc. 01–16094 Filed 6–26–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on July 10, 2001, Room T–2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON MATERIALS METALLURGY/PLANT OPERATIONS
CONTROL ROD DRIVE MECHANISM CRACKING

JULY 10, 2001
Today's Date

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
Larry K. Mathews	Southern Nuclear Operating Co.
Jim Bennetch	Dominion Generation
Peter Ford	Westinghouse
FRANK AMMIRATO	EPRI
Chuck Wolfe	EPRI
Steve Hunt	DEI
Gary Shenwood	NMC LLC
KURT COZENS	NEI
Mike Siewertsen	CCNPP I - CN
Steven Doctor	PNNL
Robert Hermann	SIA
D Bryan Miller	Entergy
Roger Huston	Licensing Support Services
Pete Yarsky	Rensselaer Polytech Ins.
Paul Gunter	NIRS
Ramin Assa	NRC/OPA
Michael Robinson	Duke Power
Alex Marion	NEI
William Gray	Framatome ANP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON MATERIALS METALLURGY/PLANT OPERATIONS
CONTROL ROD DRIVE MECHANISM CRACKING

JULY 10, 2001
Today's Date

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
Patricia L. Campbell	Winston & Strawn
B. Richard Bass	ORNL
GERY WILKOWSKI	Engineering Mech. Corp. of Columbus
JACK A. BAILEY	EPRI/MRP Chairman - TVA
M. A KRUPA	ENTERGY.
Stephen Fyfitch	Framatome ANP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON MATERIALS METALLURGY/PLANT OPERATIONS
CONTROL ROD DRIVE MECHANISM CRACKING

JULY 10, 2001
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
Allen Hiser	NRR/DE/EMCB
FAROUK ELTAWILA	NRR/DE
KEITH WICHMAN	NRR/DE/EMCB
SHAH MALIK	RES/DET/MEB
JAMES MEDOFF	NRR/DE/EMCB
JAK COLLINS	NRR/DE/EMCB
JACK STROSNIDER	NRR/DE
Debbie Jackson	RES/DET/MEB
L. B. (Tad) Marsh	NRR/DRIP/REXB
Jacob Zimmerman	NRR/DLPM/PDI-2
Matthew McConnell	NRR/DE/EEIB
Marll Reinhart	NRR/DSSA/SPSB
MARK KIRK	NRC/RES/DET/MEB
TIMOTHY G. COLBURN	NRR/DLPM/PDI
Bill Bateman	NRR/DE/EMCB
RK CALDWELL	NRR/DRIP/REXB
Lee Abramson	RES/DRAA/RES
Doug Kalinowski	RES/DET/MEB
ARTHUR BUSLIK	RES/DRAA/RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON MATERIALS METALLURGY/PLANT OPERATIONS
CONTROL ROD DRIVE MECHANISM CRACKING

JULY 10, 2001
Today's Date

NRC STAFF SIGN IN FOR ACRS MEETING

PLEASE PRINT

NAME	NRC ORGANIZATION
ED CHOW	RES / DRAA / PRAB
Bob Jasinski	OPA
JIN CHUNG	NRR / DSSA / SPSB
ED HACKETT	RES / DET / MEB
~~Stephen Fyfe~~	

R

MRP - Alloy 600 ITG RPV Penetrations

Presentation To ACRS Subcommittees
July 10, 2001

Purpose

- Industry Goals:
  - Near Term: Assure Structural Integrity
  - Longer Term: Develop Program to Manage PWSCC
- Explain Background of Head Penetration Issue
- Present Status of MRP Program
- MRP Recommendations for Industry

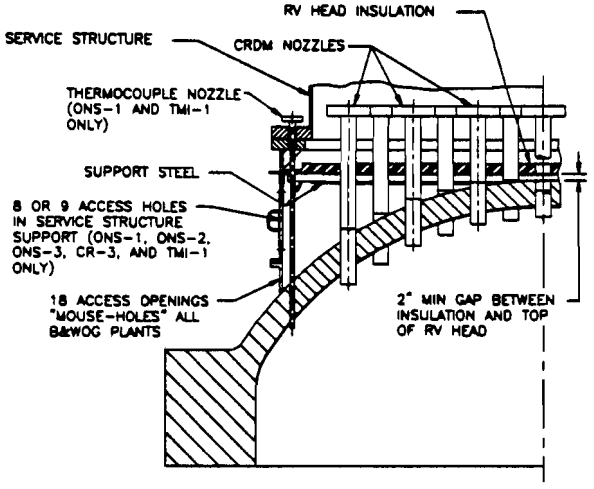
RPV Penetration Summary

- Near Term Conclusions:
  - Axial PWSCC in CRDM nozzles does not impact plant safety
    - Bounded by previously submitted Safety Assessments (1993/94)
  - Reasonable assurance that other PWRs do not have circumferential cracking that would exceed structural margin
    - Oconee and ANO-1 in highest grouping based on effective time-at-temperature
    - Leaks discovered by careful visual inspection of top head surface
    - Volumetric examination of other nozzles found only minor craze cracks
    - Leaks discovered with significant structural margin remaining
    - Several other plants in highest groupings have no evidence of leakage

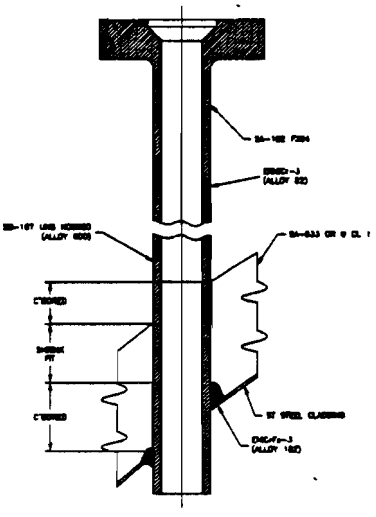
Other Ongoing MRP Activities

- Risk Assessments
- Probabilistic Fracture Mechanics
- Assessment of Crack Growth Data and Needs
- NDE Demonstration
  - Block Design and Fabrication
  - Technique Development and Demonstration
- Information and Training Package for Visual Examination
- Flaw Evaluation Guidelines
- Review of Repair and Mitigation Strategies

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



Schematic View of B&W-Design CRDM Nozzle Area



Issue Background

- Bugey-3 cracking in 1991 characterized as:
  - ID-initiated, through-wall axial flaws
  - Through-wall axial flaw initiated OD circumferential flaw in RV head penetration crevice
- Lack of fusion detected in attachment welds at Ringhals-2 (1992)
- Industry safety assessments prepared (early 90's) for these types of cracking
- Additional European PWRs Discovered Axial Penetration Cracks and Initiated Head Replacements
- DC Cook 2 Found and Repaired a Single Cracked Penetration (1994)
- Owners Groups Programs to Manage for Their Units

Background: GL 97-01

- GL 97-01 Issued April 1, 1997
- Owners Groups Prepared Generic Responses
- Responses Coordinated Between Owners Groups by NEI Task Force
  - Histogram Ranked Plants, Normalizing Both Industry Models to DC Cook 2
- Individual Utilities Supplied Information for Their Plants
- Lead Plants Scheduled for Inspections Based on Histogram
  - ET for Detection
  - UT for Sizing of ID Flaws

Background: GL 97-01 Histogram

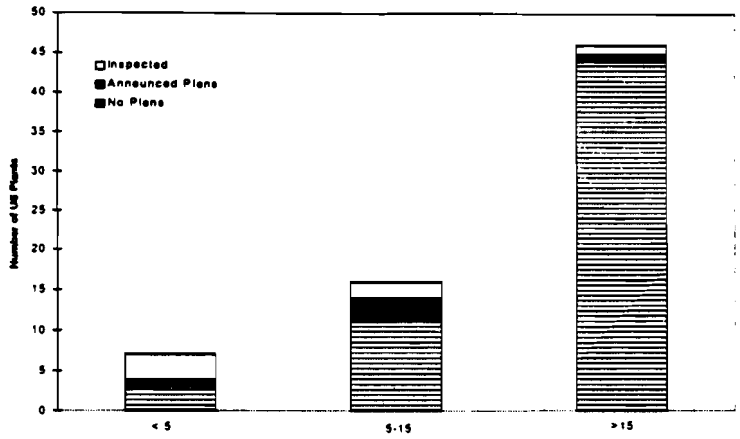
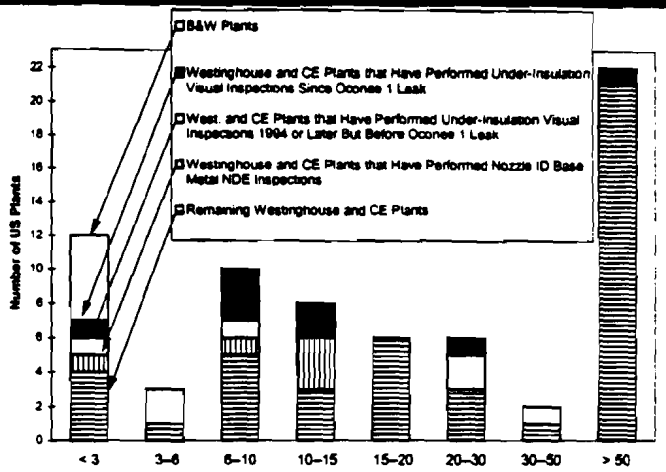


Figure 1: Effective Full Power Years, Measured from January 1997

Recent Experience

- Recent **J-groove Weld and OD-initiated** Cracking Observed at B&W-Design Plants
  - ONS-1 (November 2000)
  - ONS-3 (February 2001)
  - ANO-1 (March 2001)
  - ONS-2 (April 2001)

Time-Temperature Histogram Chart in MRP-44 Part 2 Interim Safety Assessment



Recent Experience

Oconee Experience

- Visual inspection of Unit 1 RV head identified small amounts of boron accumulation at the base of CRDM nozzle 21 and several T/C nozzles.
- Visual inspection of Unit 3 reactor vessel head identified small amounts of boron accumulation at the base of several CRDM nozzles. The suspect nozzles were #'s 3, 7, 11, 23, 28, 34, 50, 56, 63.
- Visual inspection of Unit 2 reactor vessel head identified boron accumulation at the base of CRDM nozzle #'s 4,6,18, and 30

Oconee Background Information

- Modifications to cut access ports (9 each - 12 in diameter) into the Oconee service structure were completed during outages in Spring 1994, Spring 1993, and Fall 1994 for Units 1, 2, and 3 respectively.
- Modifications to service structure allowed access to domed portion of head for bare metal inspections and wash down of the head to remove old boron deposits.

Oconee
Background Information

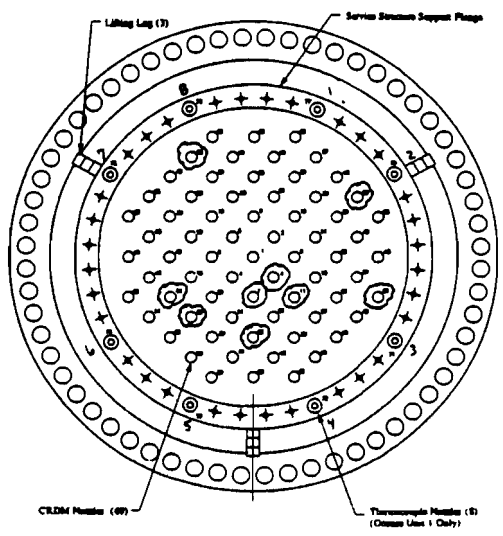
- T/C nozzles installed in Unit 1(only) for instrumentation purposes, but were never put into service.
- Located outboard of the CRDMs and fabricated from 0.75" Schedule 160 Alloy 600 pipe
- Material Specification is SB-167 and procured from Huntington Alloys as cold drawn, ground, and annealed pipe
- Procured to 1965 ASME B&PV Code

ONS-1 RV Head Showing Boric Acid At Thermocouple Nozzle

Oconee
Background Information

- CRDM (69) nozzles are constructed of Alloy 600 and procured in accordance with requirements of SB-167, Section II to 1965 Edition including addenda through Summer 1967 of ASME B&PV Code.
- CRDM nozzle material was hot rolled and annealed by B&W Tubular Products Division.
- CRDM nozzles were shrink fit into reactor vessel head penetration and welded with a J-groove weld with Alloy 600 filler

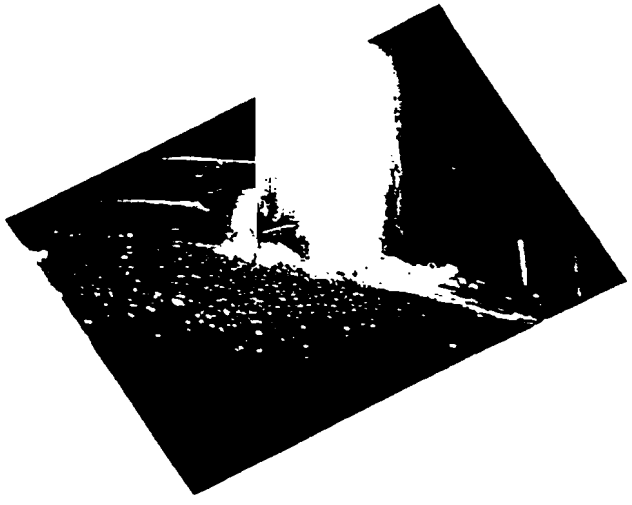
CRDM Nozzle Layout



Summary of Recent Cracking Incidents

- ONS-1:
  - All eight thermocouple nozzles contained flaws predominantly axial in orientation
    - Five nozzles identified as leaking
    - ID cracking observed on all eight nozzles
    - Cracking penetrated into all eight nozzle welds
  - CRDM nozzle 21 did not contain ID flaws
    - Flaws in weld material, predominantly axial/radial in orientation, identified as leak source
    - Flaw propagated through the weld and nozzle base material

ONS-1 RV Head Showing Boric Acid At CRDM Nozzle 21



Summary of Recent Cracking Incidents (Cont.)

- ONS-3:
  - Nine CRDM nozzles found leaking
    - Numerous axially oriented flaws identified
    - OD-initiated circumferential flaws (relatively deep and below the weld) identified on four nozzles
    - OD-initiated circumferential flaws (above the weld and up to through-wall) identified on two nozzles
    - Some weld cracking also identified

CRDM Nozzle #56

CRDM Nozzle #50

Summary of Recent Cracking Incidents (Cont.)

- ANO 1 CRDM nozzle 56 found leaking
  - No ID axially oriented flaws identified
  - One OD-initiated circumferential flaw below the weld that turned axial identified
  - Flaw propagated through the weld area along the nozzle OD

Visual Inspection
ANO 1

Investigations Performed
ONS 1 & 3

- Non-Destructive Examinations
- Metallurgical Examinations
- Analytical Evaluations

Non-Destructive Examinations

- Pre-Repair Inspections Performed
  - Visual inspections of all 69 CRDM nozzles
  - Dye Penetrant (PT)
  - Eddy Current Testing (ECT)
  - Ultrasonic Examination-Axial
  - Ultrasonic Examination-Circumferential

Visual Inspections

- Bare head inspections are performed through the modified openings in the head service structure
- Visual inspections are performed as part of each refueling outage for our response to GL 88-05 and 97-01
  - The same experienced system engineer performs these inspections
- Heads essentially clear of old boron deposits
- Amount of leakage from each leaking nozzle has been very small, which suggests, low leak rates
- No evidence of boric acid corrosion on top of head

Non-Destructive Inspections

- Dye Penetrant (PT) Inspection
  - Surface examination that looks at the weld surface area and the top 1 inch of the nozzle that projects down into the plenum of the head
  - Performed on suspected leaking CRDM nozzles
- Eddy Current (ECT) Inspection
  - Surface examination (plus 2 to 3 mm into the material) from the nozzle ID
  - Performed on suspected leaking nozzles
  - Checks a band 6 inches above the weld down to free end of nozzle
  - Later performed on additional nozzles, to address extent of condition
    - 8 Unit 1 CRDM nozzles
    - 9 Unit 3 CRDM nozzles

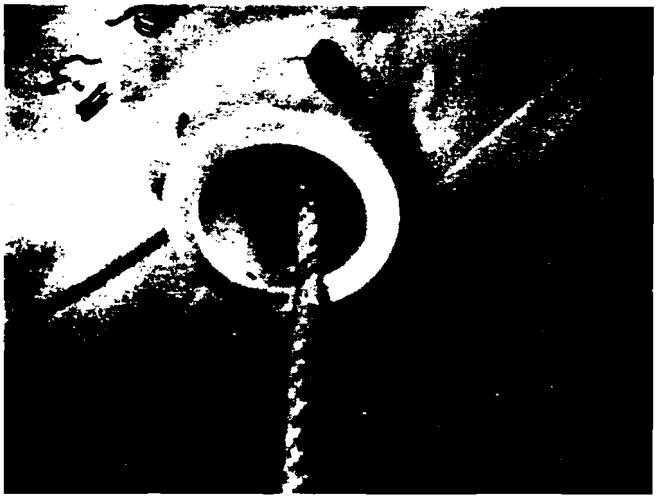
Non-Destructive Inspections

- Ultrasonic Examinations (UT) Axial
  - Volumetric examination to locate and depth size axial indications on both the nozzle inside diameter and the nozzle outside diameter
  - Performed on the suspected leaking nozzles and on additional nozzles to address extent of condition
    - 18 nozzles on Unit 3 inspected
- Ultrasonic Examinations (UT) Circumferential
  - Volumetric examination to detect the presence of circumferential cracking or indications and lack of bond
  - Performed on the suspected leaking nozzles and on additional nozzles to address extent of condition
    - 18 nozzles on Unit 1 (lack of bond)
    - 18 nozzles on Unit 3 (circumferential)

CRDM Nozzle #11

CRDM Nozzle #23

CRDM Nozzle #56



ONS 3: Summary Nozzle Indications and Characterization

- Total of 48 indications in the nine leaking CRDMs
  - 39 are axial and located beneath the weld at the uphill and downhill
  - 16 indications thru wall (39%), all are axial, and occur on 6 of 9 nozzles
- Confirmed two (2) above the weld circumferential cracks
  - Nozzle 56 crack was thru wall
  - Nozzle 50 except for pin hole indications on ID was not thru wall
  - Inspection and metallurgical results indicate the circumferential cracks were O.D. initiated.
- Unit 3 CRDMs extent of condition inspections (9 additional nozzles):
  - Cluster indications above and/or below the J groove weld.

Circumferential Cracks Above Weld

- Discovered during post weld repair NDE of Nozzles 50 & 56
- Circumferential cracks followed the weld profile contour and were O.D. initiated.
- Both ECT and UT inspections identified indications in these areas but were dispositioned as crazed cracks with unusual characteristics
- The original NDE characterization for nozzles 50 and 56 subsequently changed.
- This change in interpretation of the NDE signals is related to the flaw orientation with respect to the sound beam of the UT search units.
- Actions taken as a result of this discovery were:
  - All Unit 1 and 3 ECT and UT data re-reviewed applying the LLs
  - EPRI NDEC led an independent review of ONS 1 & 3 data to confirm results and findings

Metallurgical Examinations

- T/C nozzle specimen (2) from Unit 1
- CRDM #21 182 weld filler material boat sample from Unit 1
- CRDM nozzle end pieces (7) from Unit 3
- CRDM nozzle 56 circumferential crack boat sample, Unit 3

Unit 1: Summary Results of Metallurgical Examinations

- T/C Nozzles:
  - Cracks are intergranular and branched
  - Cracks are axial and radial in orientation
  - Material appears to be typical of mill annealed Alloy 600 with some evidence of cold working on both the OD and ID surfaces
  - Microstructure mixed with both intra and intergranular carbides
  - Microstructure characterized by small clusters of small grain with some large grains; Grain size ASTM 7-8
  - No indication of aggressive chemical species on the crack face
  - PWSCC was the primary mechanism for crack propagation

Unit 1: Summary Results of Metallurgical Examinations

- CRDM Nozzle 21:
  - Crack in weld was completely interdendritic
  - No conclusive evidence of manufacturing defects in the original weld
  - Crack in weld was connected to a branched intergranular crack in the nozzle wall
  - Qualitative comparison of boat sample to a 182 weld pad confirmed alloy type material, as expected
  - PWSCC was the primary mechanism for crack propagation in the CRDM weld and housing

Unit 3: Summary Results of Metallurgical Examinations

- CRDM Housing Material Specimen:
  - Microstructure of all nozzle materials very similar and typical for mill annealed Alloy 600. Grain size is ASTM 4.
  - Grain boundaries contain a semi-continuous carbide decoration
  - No ghost grain boundaries or segregated carbide clusters
  - All cracks in the samples were intergranular with slight branching
  - Micro-hardness survey across the thickness shows a range from about Rb 80 at the ID to Rb 95 at the OD
  - Several nozzles exhibited cracks originating at free end of nozzle
  - All cracks are stress corrosion cracks with PWSCC as the primary mechanism for crack propagation

Unit 3: Summary Results of Metallurgical Examinations

- CRDM 56 Boat Sample (Circ Crack):
  - Boat sample in the area of circ crack that was found above the weld after the weld repairs were completed
  - Boat sample contained a face of the circ crack along with 3 small axial cracks that intersect the circ crack
  - Section through the axial crack confirms crack is totally intergranular with small intergranular branches
  - Scanning electron microscopy of the circ crack face revealed only intergranular morphology.
  - There are no tears or other indications of the origin of the circ crack
  - Circ crack is indicative of PWSCC

Correlation of Observed Crack Locations with FE Stress Analysis

- Cracks are:
  - predominantly axial and located on the uphill and downhill sides of the nozzle
  - most initiate on the OD of the nozzle
  - circumferential cracks found below and above the weld, at the weld toe on the uphill and downhill sides of the nozzle

Correlation of Observed Crack Locations with FE Stress Analysis

- Stress analysis (residual + operation) preliminary results:
  - Hoop stresses exceed axial stresses at most locations which suggests axial cracking would be expected. This is consistent with observed field conditions
  - Axial stresses are higher on the uphill side of the nozzle relative to downhill side of nozzle. Field observed locations of the above the weld circumferential cracks align with this analysis prediction.
  - Microhardness measurements suggest the material yield strength is significantly higher on outside of nozzle than on the inside. The high outside yield strength may explain the preferred OD cracking

Oconee Repairs

- Repairs performed in accordance with 1992 Section XI of ASME Code, applicable Code Cases, and NRC approved alternatives, as required
- Removed flaws from both weld material and nozzle base material for Units 1 & 3
  - Automated weld process to apply protective layer over J groove weld
- Automated repair method used for Unit 2 removed cracked nozzle material and established new pressure boundary location. Cracks left in remaining J-groove weld

CRDM Nozzle #50

ANO 1 Repair

- Embedded Flaw Repair
  - OD axial flaw removed down to the butter
  - Weld repaired, isolating remaining flaw above the weld from the environment
  - Peened repair area
- Post-repair UT to confirm remaining flaw did not grow during repair process

Industry Response

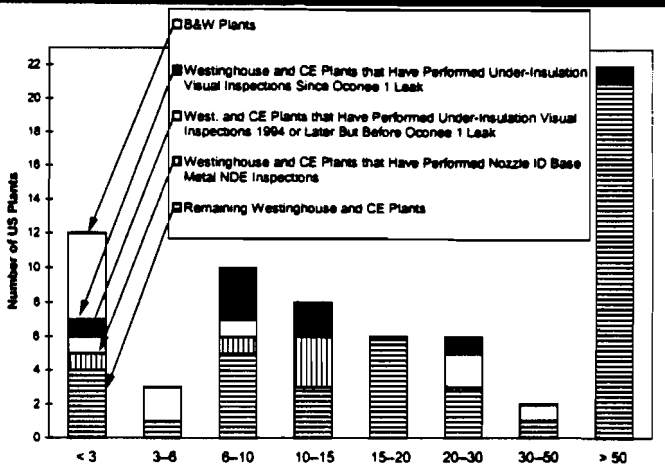
Industry Response Organization

- Integrated effort is being coordinated through
  - EPRI Materials Reliability Project - Alloy 600 ITG
    - NEI - Regulatory Interface
    - Committees Under Alloy 600 ITG
      - Assessment
      - Inspection
      - Repair/Mitigation
    - Owners Groups
- Work is being performed by
  - Utilities
  - NSSS Vendors
  - Contractors

MRP Interim Safety Assessment

- Interim Safety Assessment Submitted May 18, 2001
- Developed a Histogram of Time for Each Unit to Reach the Equivalent Time at Temperature as ONS 3 (normalized to 600F)
  - Sorted plants into bins, <3 EFPY, 3-6 EFPY, 6-10 EFPY, etc.
- Recommended Plants <10 EFPY from ONS 3 with Fall Outages perform visual inspections
  - Capable of detecting small amounts of Boron similar to ONS & ANO

Time-Temperature Histogram Chart in MRP-44 Part 2 Interim Safety Assessment



MRP Interim Safety Assessment

- Bases for No Significant Near-term Impact on Plant Safety:
  - The Three Oconee Units and ANO-1 Are Among the Lead Units in the US Based on Time at Temperature
  - Leaks Were Found by Careful Visual Inspections
  - Structural Integrity Evaluations Showed the Nozzles and Welds Were Well Within Required Margins
  - Leakage Should Also Be Detectable in Other Plants
  - Several Other Lead Units With Long Operating Times and High Head Temperatures Had Already Performed Inspections From Above and Below the Head Without Any Significant Findings
  - A CRDM Nozzle Ejection Is an Analyzed Event in Plant FSARs
  - Existing Symptom Based EOPs and Operator Training Adequate

NRC Questions

- NRC identified several questions on May 25, 2001:
  - Leak detection
    - Effect of initial interference fit on leak detection
  - Time-temperature histogram
    - Effect of activation energy on predictions
    - Benchmarking against foreign plant inspections
    - Basis for ten year inspection criterion
  - Growth rate of circumferential cracks
    - Time until Oconee 3 would have reached allowable flaw size
    - Effect of crack growth rates on histogram
  - Loose parts
  - Risk assessment

NRC Questions

- NRC Documented Those and Asked Additional Questions on June 22, 2001:
  - Photos of visual inspections performed at other units
  - Inspection Capabilities
    - Ability to Perform Volumetric NDE
      - Nozzles for ID/OD Flaws
      - J-groove Welds
    - Estimate of Number, Time, Other Costs to Perform Volumetric and Visual Inspections by 1/1/2002
      - During Scheduled Outage
      - During Unscheduled Outage

Safety Assessment Status

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

Leakage Detection

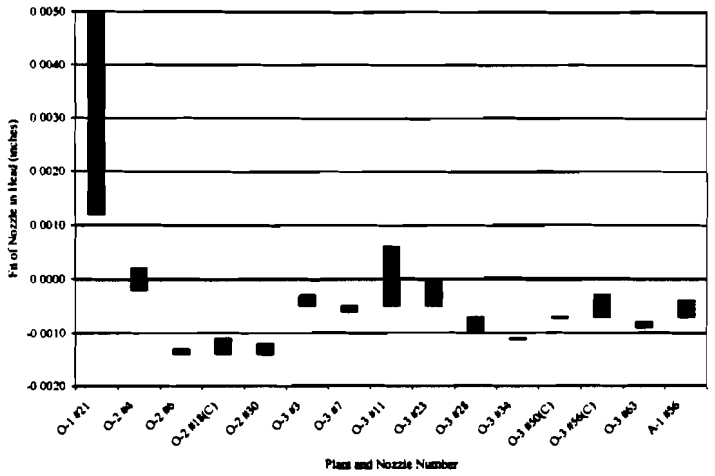
Leakage Detection

- Oconee and ANO-1 detected leakage, but
  - Some other plants have greater interference fits (see Table 3-2 of Interim Safety Assessment)
- Leakage should be detectable at most other penetrations given similar cracks
  - Only minor craze cracking was found in NDE examinations of 17 additional "non-leaking" Oconee 1 and 3 CRDM nozzles. This supports appropriateness of visual inspections for detection of through-wall cracks in CRDM nozzles
  - Interference fits at other plants are only slightly larger than Oconee and ANO-1
  - Further experience has shown that it is difficult to prevent leakage of 2,250 psi water without roll, hydraulic or explosive expansion or use of a sealant

Leakage Detection
Actual Fits at Oconee and ANO-1

- Fabrication records for Oconee 1, 2, and 3 and ANO-1 vessel heads have been reviewed
- The following measurements were taken
  - ID of the hole in the vessel head at the top and bottom of the interference fit region
  - OD of the nozzle
- Results for the 14 leaking CRDM nozzles at Oconee 1, 2, and 3 and ANO-1 are shown on next slide
  - One nozzle had a clearance fit (gap)
  - The remaining nozzles had at least one end within the specified diametral interference range of 0.0005 - 0.0015 inches. Three of the four leaking ONS 2 nozzles had interference fits of 0.0014 inches on one end and at least 0.0011 inches on the other.

Leakage Detection
Actual Fits at Oconee 1, 2, and 3 and ANO-1



Leakage Detection
Effect of Operating Conditions on Fit

- Differential thermal expansion has only a small effect, increasing the initial interference fit by <0.0014"
- The change in fit under operating conditions is primarily due to pressure dilation of the vessel head
- For the example, the change in diametral fit due to pressure dilation is approximately
  - ΔD = 0.00402" - 0.00048" = 0.0035"
  - The hole will open up further when the effect of reduced effective modulus due to the effect of multiple nozzles is considered
- Therefore annular gaps are expected for most CRDM nozzles under operating conditions

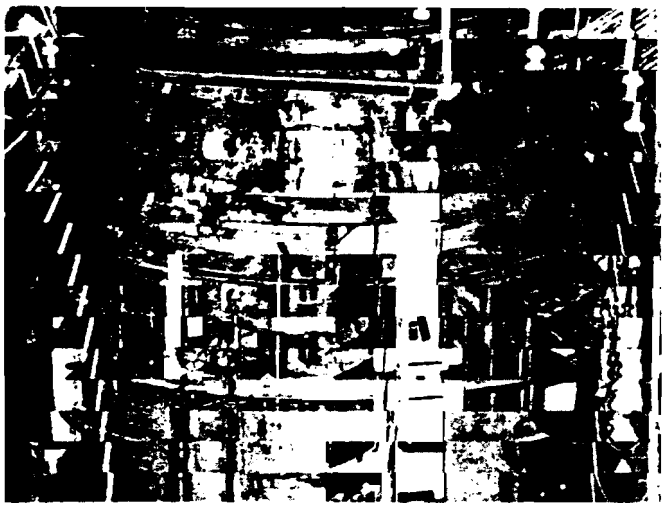
Leakage Detection
Other Effects on Fit

- Finite element analyses show that outer row CRDM nozzles displace laterally and become slightly ovalized in the vessel head as a clearance opens up under operating conditions
  - The displacement and ovalization reduce the leak path at some locations and increase the leak path at other locations
  - The net effect is to create a spiral flow path which has less resistance than a uniform annular gap
- Finite element analyses also show a minor (~20%) increase in ovality for peripheral CRDMs from flange tensioning and rotation

Visual Inspections Spring 2001

- Several other plants performed visual inspections during Spring outages
  - Robinson 2
  - Salem 1
  - Farley 2
  - Prairie Island 1
  - McGuire 1 (partial)
  - SONGS 3 (partial)
- Heads reasonably free of masking boric acid deposits
- No evidence of leakage found

Visual Inspection
Salem 1



Time Temperature Histogram

Time-Temperature Histogram Background

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

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

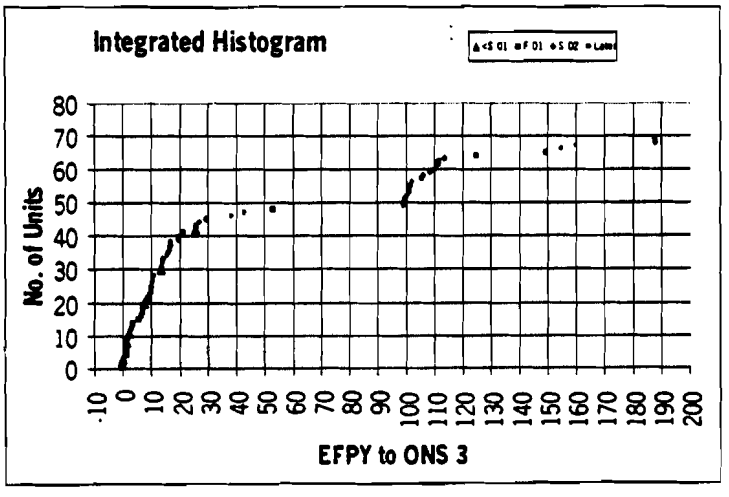
- A sensitivity study for the results of the plant assessments was performed
- The effect is small, as shown below:

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

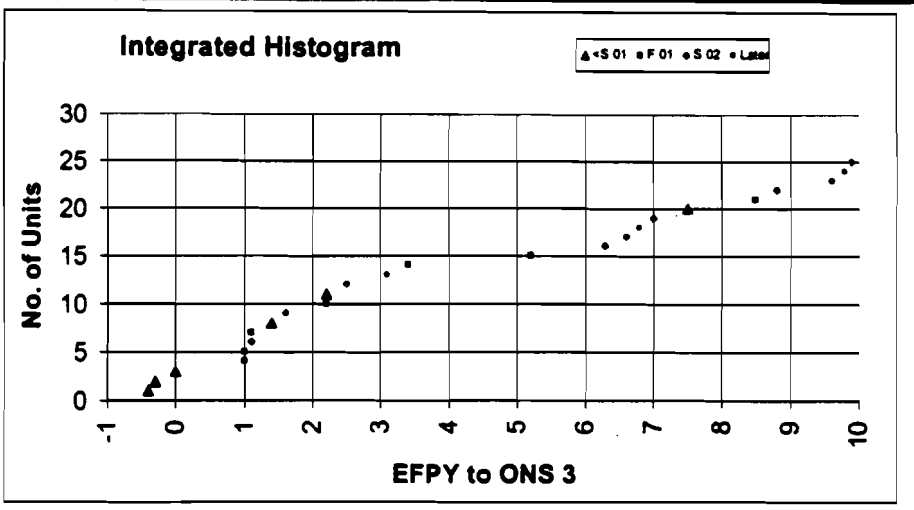
Time-Temperature Histogram Ten-Year Period

- 10 Year Period for Near-Term Inspection
  - The ten year period for recommending visual inspections of the top of the vessel head for small amounts of leakage similar to that observed at Oconee and ANO-1 was selected to provide some margin for uncertainties
  - Encompasses 25 units
  - All but two will have outages by Spring '02
  - The ten year period will be re-assessed based on results of upcoming outages

Time-Temperature Histogram



Time-Temperature Histogram



Circumferential Crack Growth Growth Rate in Annulus Environment

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

Crack Growth

Circumferential Crack Growth Growth Rate in Annulus Environment

- Since the fluid will contain lithium hydroxide and boric acid, it will likely be similar to a controlled PWR environment
- Comparison of BWR and PWR crack growth rates for Alloy 600 and 182 shows that, at a given temperature, the growth rates are comparable
- Temperature is a stronger variable than environment for these materials
- MRP has scheduled an international expert panel to assess crack growth rates
  - Initial meeting in August

Circumferential Crack Growth Margin for Oconee 3 Cracks

- Two Oconee 3 nozzles were cracked approximately 165°
- Stress analyses show that cracks initiated in a high stress region and propagated into a lower stress region
- The remaining time for Oconee 3 circ cracks to reach ASME Code allowable ligament (safety factor of 3) was estimated to be 4-5 years, based on the modified Peter Scott model and also by assuming the maximum crack growth measured in lab
- Efforts are underway to refine the stress intensity calculations in the nozzle in the intact and cracked conditions

Loose Parts & Risk Assessment

Loose Parts

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

Risk Assessment

- Risk calculations are in process now
- The effort includes interaction with all PWR vendors and others to ensure applicability to all plants
  - Consistent with past approaches
- Staff has conservatively estimated CCDP about 10-3, assuming rod ejection, but probability of ejection event likely to be a few orders of magnitude less than 1 for all plants

Summary & Ongoing Activities

Summary

- Near Term Conclusion:
  - Axial cracks alone in CRDM nozzles do not impact plant safety
    - Bounded by previously submitted Safety Assessments (1993/94)
    - But through wall axial cracks can be a precursor to circumferential cracking
  - There is reasonable assurance that PWRs do not have circumferential cracking that would exceed structural margin
    - Oconee and ANO-1 in highest grouping based on effective time-at-temperature
    - Leaks discovered by careful visual inspection of top head surface
    - Volumetric examination of other nozzles found only minor craze cracks
    - Leaks discovered with significant structural margin remaining
    - Several other plants in highest groupings have no evidence of leakage

Schedule

- Revised Inspection Recommendations - July-August
- Expert Panel on Crack Growth - First Meeting 8/01
- Inspections during Fall 2001 outages
- Final RPV Penetration Safety Assessment - 12/01
- Reassessment of Inspection Recommendations - 2/02

Other Ongoing MRP Activities

- Risk Assessments
- Probabilistic Fracture Mechanics
- NDE Demonstration
  - Block Design and Fabrication
  - Technique Development and Demonstration
- Information and Training Package for Visual Examination
- Flaw Evaluation Guidelines
- Review of Repair and Mitigation Strategies

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

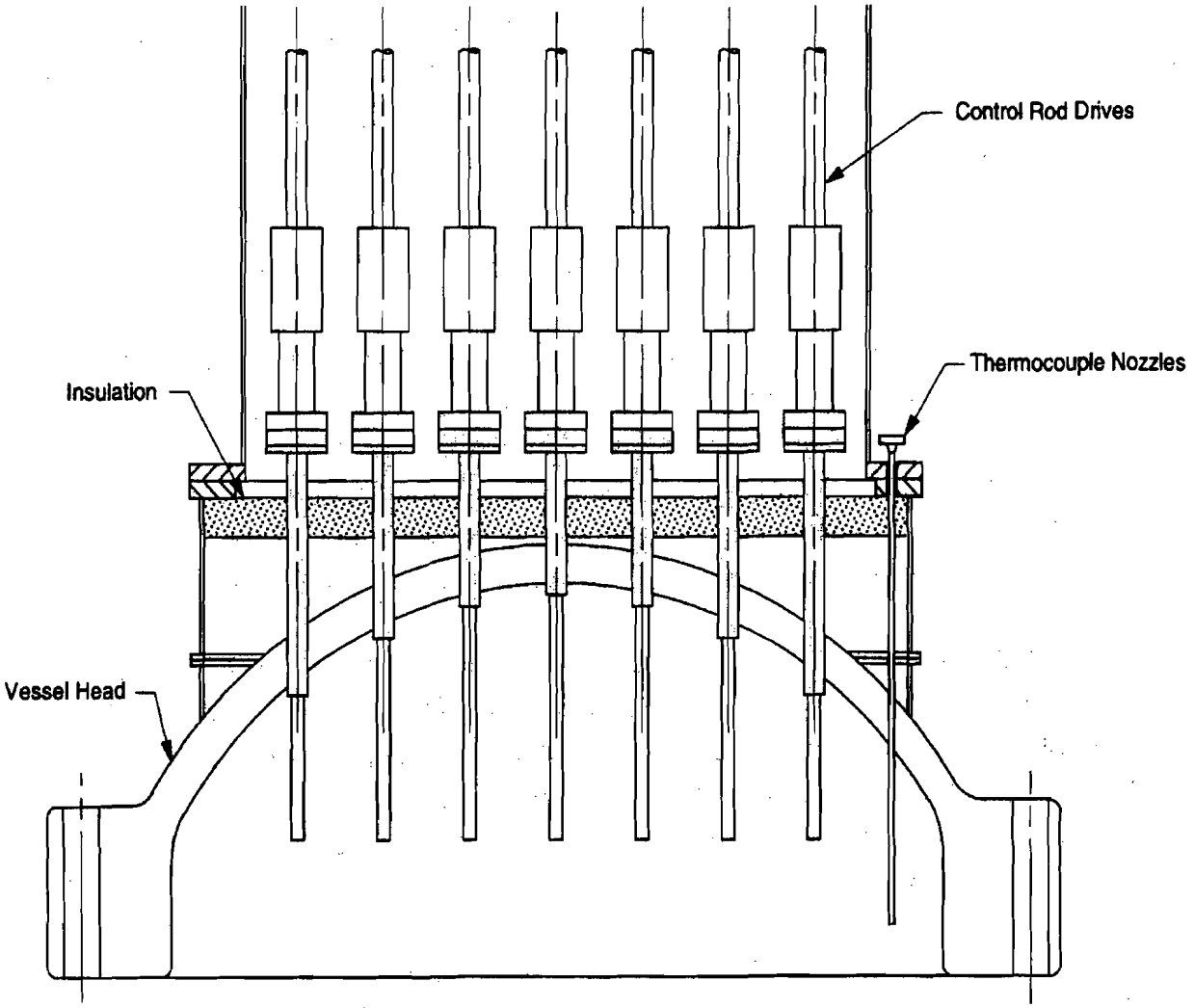
July 10, 2001

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

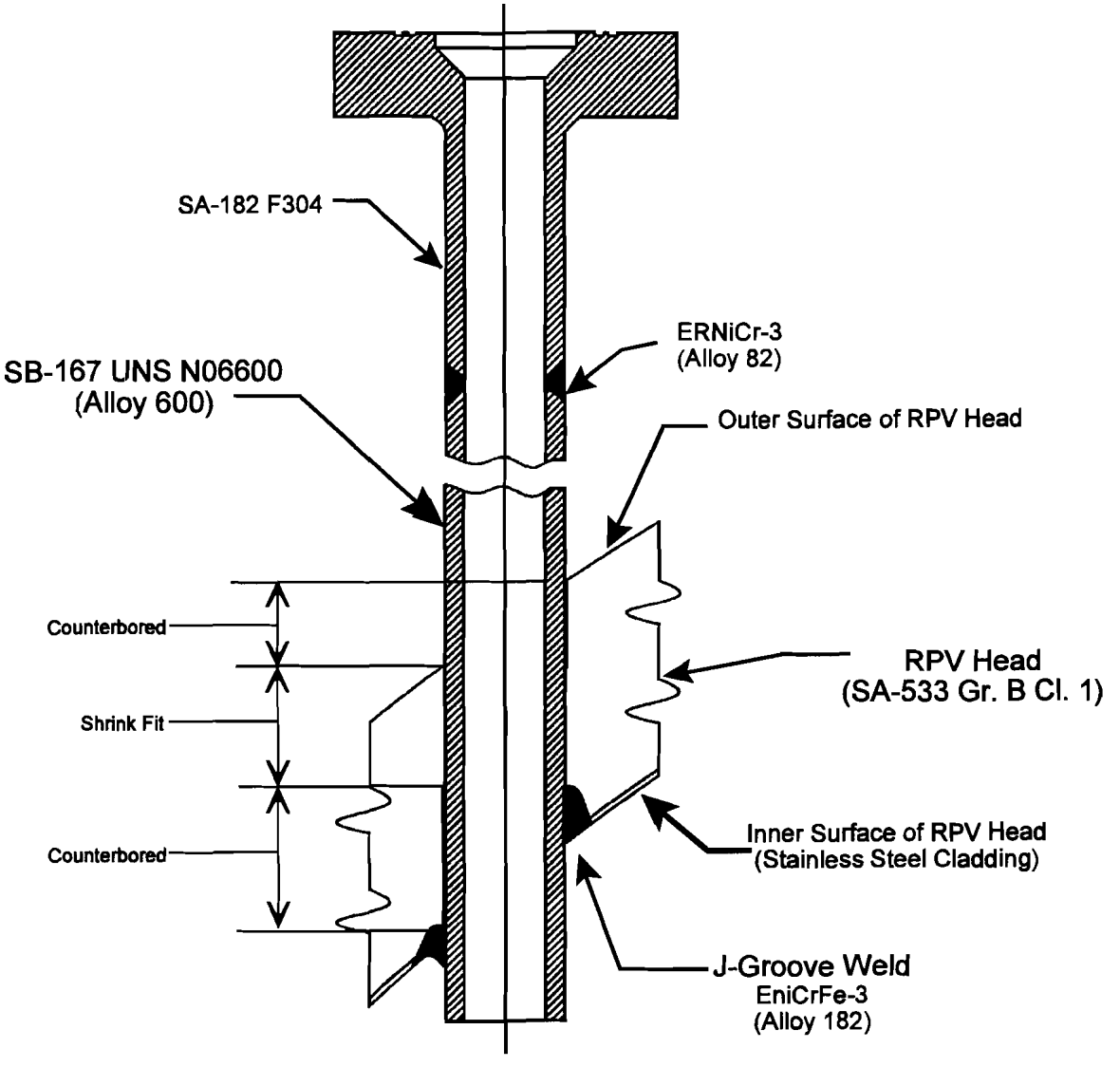
- NRC issued Generic Letter 97-01
  - Integrated industry resolution
  - Used susceptibility models to rank plants
  - Voluntary volumetric examinations at highest ranked plants
  - Boric acid walkdowns to detect throughwall leakage

- 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

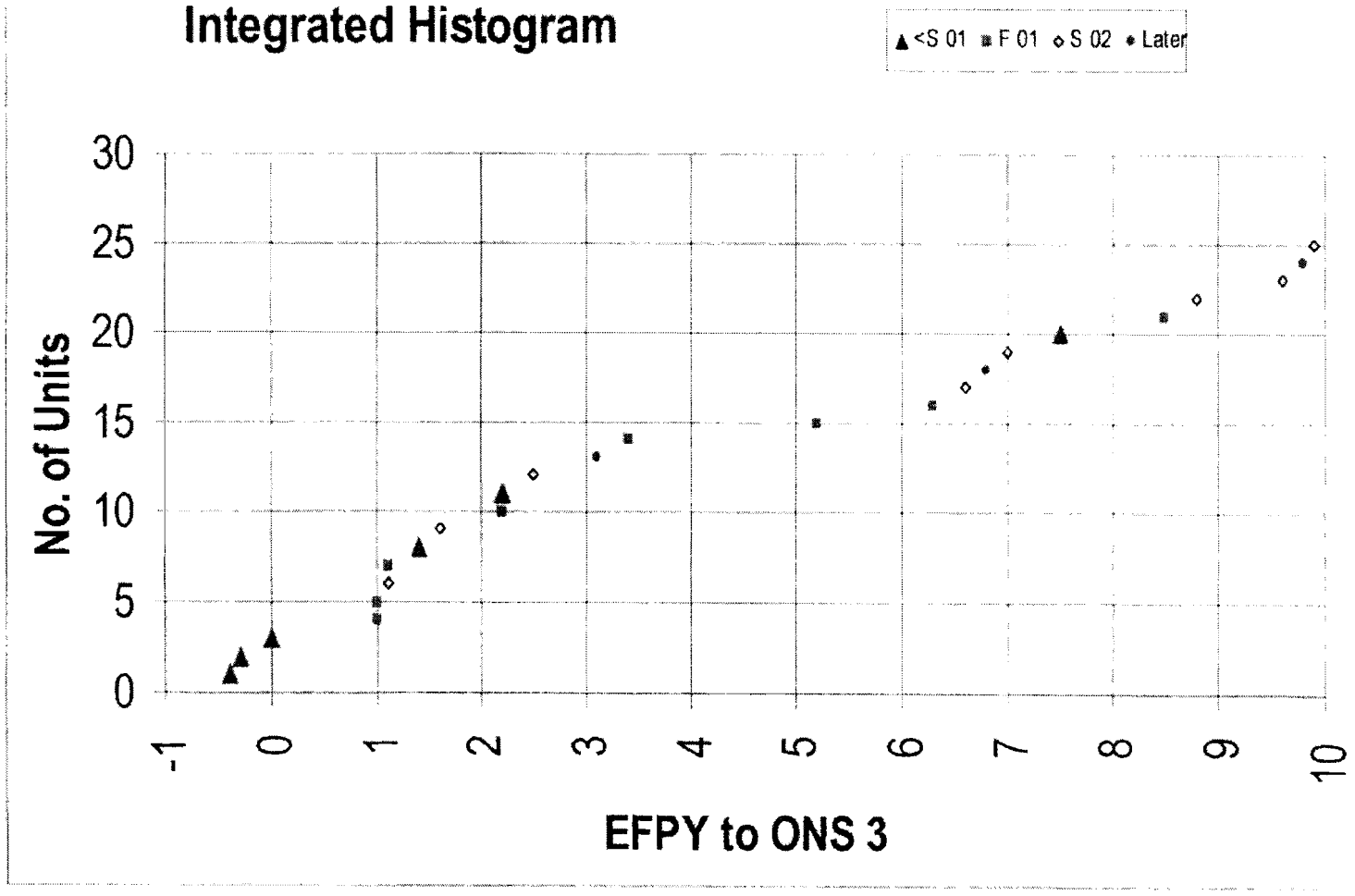
APPLICABLE REGULATORY REQUIREMENTS

- 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

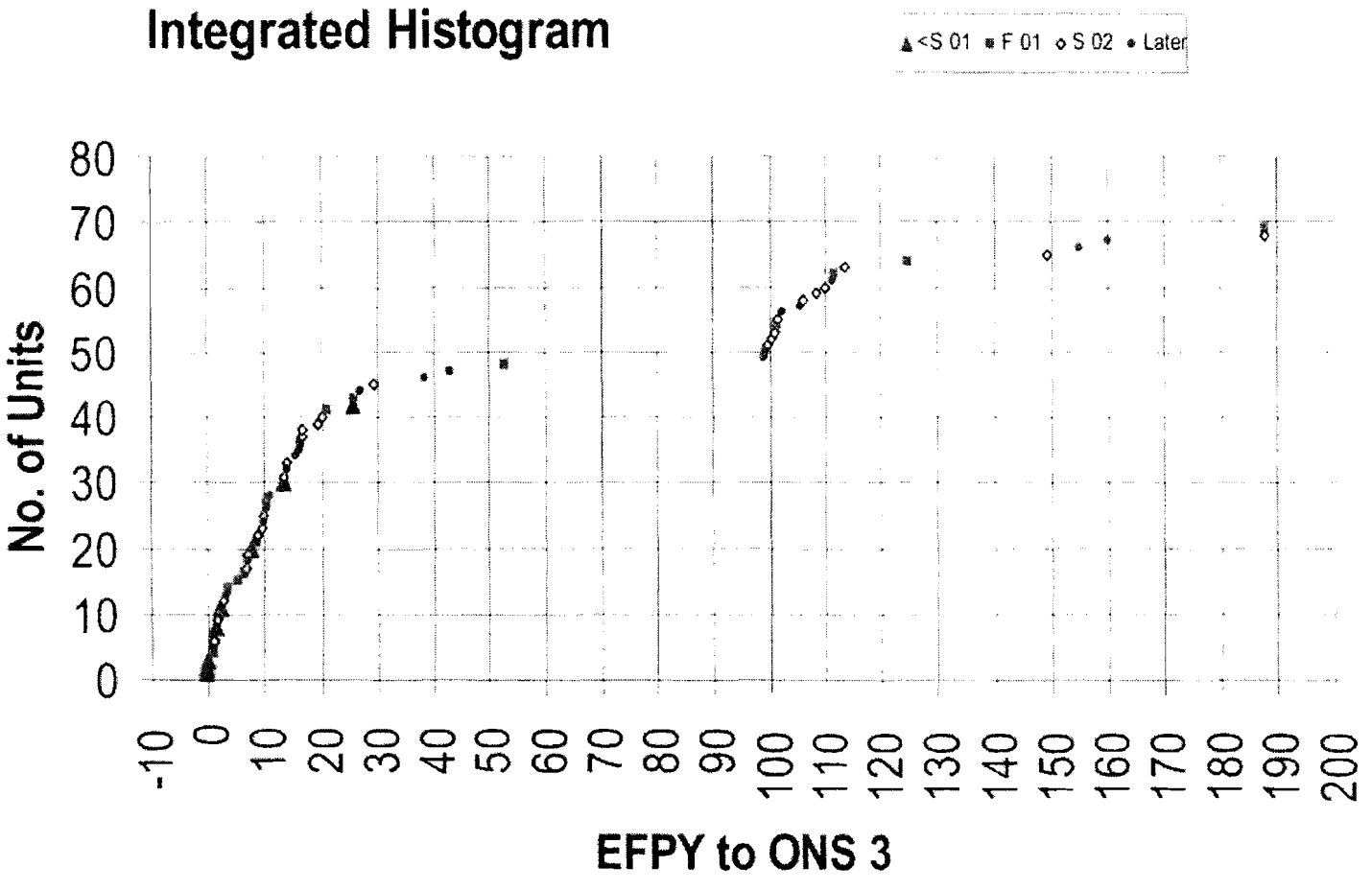
STAFF ASSESSMENT OF SUSCEPTIBILITY

- 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:

  - Environmentally Assisted Cracking
  - Non-destructive Evaluation
  - Structural Integrity/Fracture Mechanics
  - 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
  - Annulus region between the head and VHP will likely be a site for concentration of aggressive chemical species
  - Initiation frequency and crack growth rates for this situation are not known, but would likely be more rapid than those observed for PWSCC
  - Initiation at multiple sites around the circumference is likely
  - 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:
  - Interference fits
  - Occlusion of annulus by deposits
  - Quantity and differentiation of deposits
  - Configuration of head insulation
  - 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:
  - Volumetric examinations are indicated for plants with known cracking and would be the preferred inspection method for high susceptibility plants
  - Vendors have current equipment capabilities but not qualified inspection methods
  - Inspections can be effective if adequate pre-qualifications can be performed
  - 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:
  - On-line monitoring for leakage or cracking is technically feasible
  - On-line leakage monitoring for certain French plants is accomplished through N-13 monitoring
  - Acoustic emission monitoring has demonstrated potential for identifying cracking in nuclear plant applications
  - 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:
  - Expert group has verified structural margin calculations by the industry
  - Alloy 600 is capable of tolerating very large through-wall circumferential cracks while still maintaining adequate structural integrity
  - 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)

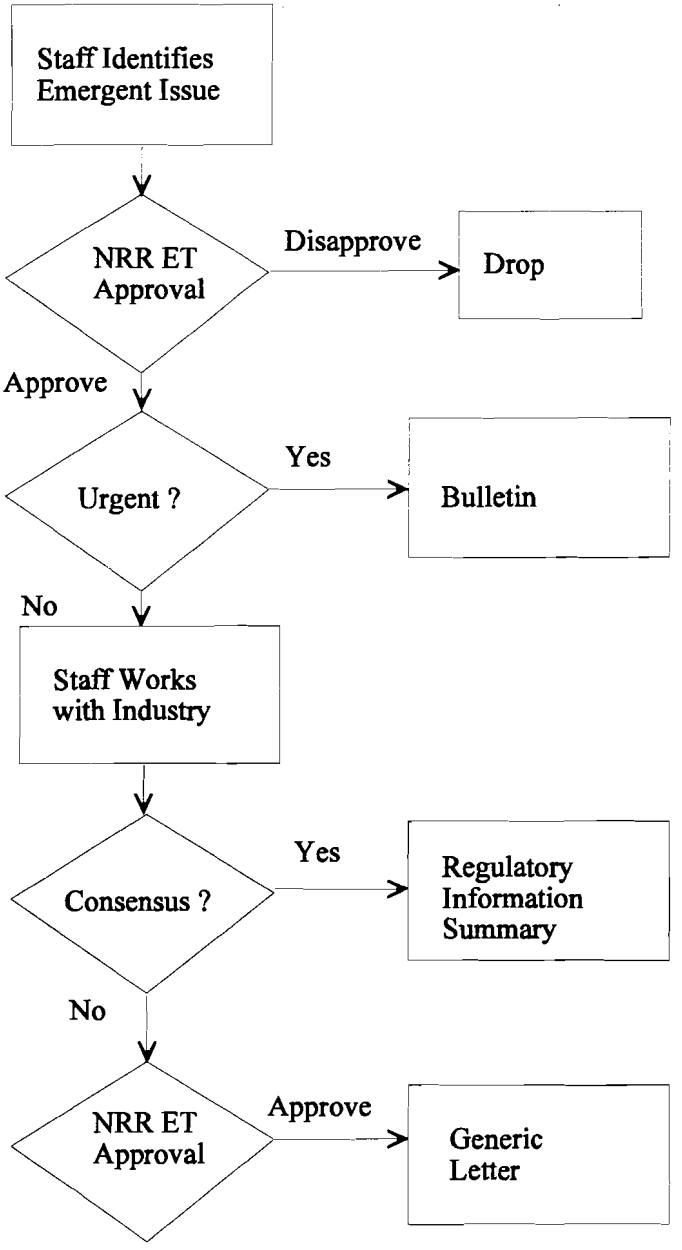
REQUEST INFORMATION

- 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