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

August 6, 2003

The Honorable Nils J. Diaz
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Diaz:

SUBJECT: SUMMARY REPORT - 504th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, JULY 9-11, 2003, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

During its 504th meeting, July 9-11, 2003, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports and memoranda:

REPORTS:

The following reports were issued to Nils J. Diaz, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Safety Culture, dated July 16, 2003
- Proposed Criteria for the Treatment of Individual Requirements in a Regulatory Analysis, dated July 17, 2003
- Security of Nuclear Facilities, dated July 18, 2003 (Internal Use Only)

MEMORANDA:

The following memoranda were issued to William D. Travers, Executive Director for Operations, NRC, from John T. Larkins, Executive Director, ACRS:

- Revision to Section 9.5.1, "Fire Protection Program," of the Standard Review Plan, dated July 15, 2003
- Draft Final Regulatory Guide DG-1105, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," dated July 15, 2003

HIGHLIGHTS OF KEY ISSUES

1. Safeguards and Security

The Committee met with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and their contractors to discuss safeguards and security matters, including Commission papers on risk-informed guidance for vulnerability assessment and on risk-informed decisionmaking, integration of the results of the vulnerability studies, potential vulnerability to sabotage of spent fuel storage facilities, and NEI-sponsored work in the area of safeguards and security. This meeting was closed pursuant to 5 U.S.C. 552b(c)(1).

Committee Action

The Committee issued a report to Chairman Diaz on this matter, dated July 16, 2003.

2. Mixed Oxide Fuel Fabrication Facility

The Committee heard presentations by and held discussions with representatives of Duke Cogema Stone and Webster (DCS) and the NRC Office of Nuclear Material Safety and Safeguards (NMSS) regarding the Mixed Oxide (MOX) fuel fabrication facility construction authorization request submitted by DCS in 2001, and the open items associated with the request. NMSS indicated that, currently, there are twelve open items to be resolved. It is noted that the facility is to be sited within the Savannah River Plant, Aiken, DC.

DCS presented information on the MOX facility mission, layout, and safety philosophy. NMSS presented information on the technical aspects associated with the facility such as the estimated risk to the public, criticality and fire safety, "red" oil, and the remaining open items.

The purpose of the MOX facility is to fabricate mixed-oxide fuel rods and assemblies from plutonium oxide powder which has been purified from weapon-grade plutonium taken from U.S. nuclear weapons stockpile. The above ground facility will be approximately 400 x 400 feet and about 65 feet tall and comprised of an aqueous polishing area, shipping and receiving, and the MOX processing area. The facility safety strategy is prevention and redundancy, nested ventilation zones, and high-efficiency particulate air filtration.

Committee Action

The Committee deferred action on a letter report to a later date because of the number of open items remaining.

3. Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the proposed criteria for treatment of individual requirements in regulatory analyses. To address the concern that aggregating or "bundling" different requirements in a single regulatory analysis could potentially mask the inclusion of an inappropriate individual requirement, the staff has developed proposed criteria for treating individual requirements in a regulatory analysis. The staff plans to develop final criteria and provide them to the Commission as part of its response to a Staff Requirements Memorandum (SRM) dated December 31, 2001.

Committee Acton

The Committee issued a report to NRC Chairman Diaz on this matter, dated July 17, 2003. The Committee found that the proposed criteria are appropriate and responsive to the Commission's direction as provided in the December 31, 2001 SRM.

4. ESBWR Pre-Application Review

The Committee heard presentations by and held discussions with representatives of the NRC staff and the General Electric (GE) Company regarding the design aspects of the Economic and Simplified Boiling Water Reactor (ESBWR) design.

The ESBWR is a 1380 Mwe boiling water reactor with improved safety margins and passive safety systems. The ESBWR design is based on the GE Simplified Boiling Water Reactor (SBWR) and the Advanced Boiling Water Reactor (ABWR) components with natural circulation and passive safety systems. All pipes and valves are inside containment with significant reduction in systems and buildings.

The ESBWR design has several diverse means of decay heat removal. Regulatory challenges need to be addressed associated with the use of the extensive new testing and NRC approved SBWR and ABWR programs. The question of whether the regulatory hurdle is too high for new plants needs to be explored.

Committee Action

This briefing was for the Committee's information. The Committee will continue to follow-up on this matter during future meetings.

5. Expert Elicitation in Support of Risk-Informing 10 CFR 50.46

The Committee met with representatives of the NRC staff to discuss the on-going expert elicitation to reconcile loss of coolant accident (LOCA) frequency distributions to support a risk-informed alternative to the present maximum LOCA break size. A proposed rule change is being developed in response to the Commission's March 31, 2003 SRM.

Committee Acton

This was an information briefing. The Committee plans to work closely with the NRC staff in the development of the proposed rule change.

6. Recent Operating Events

The Committee, in its efforts to continue awareness of recent operating events, heard an information briefing by the NRC Office of Nuclear Reactor Regulation on the South Texas Project Unit 1 bottom mounted instrumentation nozzle leakage issue. The Committee also briefly discussed events involving dryer cracking, a stuck open relief valve, leaks in GE and Framatome fuels, and automatic scrams.

Committee Action

This was an information briefing. No Committee action was taken.

RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS

- The Committee considered the response from the EDO dated July 2, 2003, to the ACRS report dated May 16, 2003, concerning the Proposed Revisions of Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," and the associated SRP Chapter 3.9.8.

  The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the response from the EDO dated June 23, 2003, to the ACRS report dated April 17, 2003, concerning the "Proposed NRC Generic Letter 2003-XX: Control Room Habitability."

  The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from June 12, 2003 through July 8, 2003, the following Subcommittee meetings were held:

- Thermal-Hydraulic Phenomena - July 8, 2003

The Subcommittee was briefed on the application of the TRACG code to the ESBWR design and scaling analysis.

- Planning and Procedures - July 8, 2003

The Subcommittee discussed proposed ACRS activities, practices, and procedures for conducting Committee business and organizational and personnel matters relating to ACRS and its staff.

- Reliability and Probabilistic Risk Assessment, and Plant Operations - July 8, 2003

The Subcommittees discussed an update on the development of the mitigating system performance indices.

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- In the Fall 2003, the Committee plans to review the results of the expert elicitation to reconcile LOCA frequency distributions. Also, the Committee plans to review proposed rule changes in a timeframe to support the SRM's due date of March 31, 2004.

- The Committee plans to continue its discussion of the construction authorization application for the MOX Fuel Fabrication Facility during a future meeting.

- The Committee plans to continue its discussion of the ESBWR design during future meetings.

- The Committee decided to refer Draft Final Regulatory Guide DG-1105, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," to the ACNW for possible consideration.

- The Committee decided that the ACRS Subcommittees on Reliability and Probabilistic Risk Assessment and on Human Factors should hold a joint meeting to discuss the Draft NUREG Report on Updated SPAR Human Reliability Analysis Methodology.

PROPOSED SCHEDULE FOR THE 505th ACRS MEETING

The Committee has tentatively agreed to consider the following topics during the 505th ACRS meeting, to be held on September 10-13, 2003:

- Final Review of St. Lucie License Renewal Application
- Draft Final Regulatory Guide DG-1107, "Water Sources for Long-Term Recirculation Cooling Following a LOCA" and Draft Final Generic Letter 2003-XX, "Potential Impact of Debris Blockage on Emergency Recirculation Design-Bases Accident at PWRs"
- Proposed Resolution of GSI-186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants"

Future topics (continued)

- Development of NRC Review Standard for Review of Core Power Uprate Requests
- Review of PIRT Process
- Draft Final Regulatory Guide, DG-1122: Determining the Technical Adequacy of PRA Results for Risk-Informed Activities
- Subcommittee Report on Fire Protection Issues

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

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

September 5, 2003

MEMORANDUM TO: Sherry Meador, Technical Secretary
Advisory Committee on Reactor Safeguards

FROM: Mario V. Bonaca
Chairman

Mario V. Bonaca

SUBJECT: CERTIFIED MINUTES OF THE 504th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), JULY 9-11, 2003

I certify that based on my review of the minutes from the 504th ACRS full Committee meeting, and to the best of my knowledge and belief, I have observed no substantive errors or omissions in the record of this proceeding subject to the comments noted below.

CERTIFIED

Date Issued: 8/27/2003
Date Certified: 9/5/2003

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MINUTES OF THE 504th ACRS MEETING

JULY 9-11, 2003

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MEMORANDA:

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APPENDICES

I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
IV. Future Agenda and Subcommittee Activities
V. List of Documents Provided to the Committee

MINUTES OF THE 504th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS JULY 9-11, 2003 ROCKVILLE, MARYLAND

The 504th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on July 9-11, 2003. Notice of this meeting was published in the *Federal Register* on June 26, 2003 (65 FR 38106) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance. There were no written statements or requests for time to make oral statements from members of the public regarding the meeting.

A transcript of selected portions of the meeting is available in the NRC's Public Document Room at One White Flint North, Room 1F-19, 11555 Rockville Pike, Rockville, Maryland. Copies of the transcript are available for purchase from Neal R. Gross and Co., Inc. 1323 Rhode Island Avenue, NW, Washington, DC 20005. Transcripts are also available at no cost to download from, or review on, the Internet at http://www.nrc.gov/ACRS/ACNW.

ATTENDEES

ACRS Members: ACRS Members: Dr. Mario V. Bonaca (Chairman), Dr. Graham B. Wallis (Vice Chairman), and Mr. Stephen L. Rosen, (Member-at-Large), Dr. George E. Apostolakis, Dr. F. Peter Ford, Dr. Thomas S. Kress, Mr. Graham M. Leitch, Dr. Dana A. Powers, Dr. Victor H. Ransom, Dr. William J. Shack, and Mr. John D. Sieber. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

Dr. Mario V. Bonaca, Committee Chairman, convened the meeting at 8:30 a.m. and reviewed the schedule for the meeting. He summarized the agenda topics for this meeting and discussed the administrative items for consideration by the full Committee.

II. Safeguards and Security (Closed)

[Note: Dr. Richard P. Savio and Mr. Richard K. Major were the Designated Federal Officials for this portion of the meeting.]

The Committee met with representatives of the NRC staff, the Nuclear Energy Institute (NEI), and their contractors to discuss safeguards and security matters, including Commission papers on risk-informed guidance for vulnerability assessment and on risk-informed decisionmaking, integration of the results of the vulnerability studies, potential vulnerability to sabotage of spent fuel storage facilities, and NEI-sponsored work in the area of safeguards and security. This meeting was closed pursuant to 5 U.S.C. 552b(c)(1).

Committee Action

The Committee issued a report to Chairman Diaz on this matter, dated July 16, 2003.

III. Mixed Oxide Fuel Fabrication Facility (Open)

[Note: Mrs. Maggalean W. Weston was the Designated Federal Official for this portion of the meeting.]

Dr. Dana A. Powers, Chairman of the Reactor Fuels Subcommittee, introduced this topic to the Committee. The purpose of this meeting was to hear a summary of the information presented at the April 21, 2003 Subcommittee meeting regarding a mixed oxide (MOX) fuel fabrication facility construction authorization request and the open items associated with the request. Presentations were made by the industry and the NRC staff.

Industry and NRC Staff Presentation

The industry presentation was made by Mr. Gary Kaplan, Duke Cogema Stone and Webster. The NRC presentation was made by Andrew Persinko, Christopher Tripp, Rex Wescott, Alex Murray, and William Troskoski, of the Office of Nuclear Material Safety and Safeguards.

Mr. Kaplan provided information on the MOX facility mission, layout and safety philosophy. NMSS provided information on the technical aspects associated with the facility such as the estimated risk to the public, criticality and fire safety, "red" oil, and the remaining open items.

NRC received the environmental report in December 2000, and the construction authorization request in February 2001. These were revised in July 2002 and October 2002, respectively. The NRC issued a draft safety evaluation report (SER) in April 2002, a draft environmental impact statement for public comment in February 2003, and a revised draft SER in April 2003.

The MOX facility is being constructed to irreversibly transform excess plutonium (Pu) into an unusable form for weapons. Its purpose is to fabricate plutonium oxide powder into mixed oxide fuel assemblies. Weapons-grade Pu coming into the Savannah River Site will go to a Pit Disassembly and Conversion Facility, under the jurisdiction of DOE, and then to the MOX facility. The above ground facility will be approximately 400 x 400 feet and approximately 65 feet tall. This building comprises an aqueous polishing area, shipping and receiving, and the MOX processing area. The MOX fuel fabrication facility safety strategy is prevention and redundancy, nested ventilation zones, and HEPA filtration. Some of the major hazards associated with the facility are criticality and fire. The fire hazards include kerosene, "red" oil, nitroamine nitrate, sintering furnaces, zirconium metal, and the waste handling facilities. NMSS indicated that there are twelve open items to be resolved.

Committee Action

The Committee deferred action on a letter report to a later date because of the number of open items remaining.

IV. Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses (Open)

[Note: Mr. Michael R. Snodderly was the Designated Federal Official for this portion of the meeting.]

Dr. Tom Kress, the cognizant Committee member for this issue, introduced the topic. He explained that there is a concern that bundling different requirements in a single regulatory analysis could potentially mask the inclusion of an inappropriate individual requirement. To address this concern the staff has developed proposed criteria for the treatment of individual requirements in a regulatory analysis. Dr. Kress said that the staff would present the proposed criteria to the Committee before forwarding them to the Commission. Dr. Bonaca reminded the Committee that this concern was identified during the risk-informing of hydrogen control requirements.

NRC Staff Presentation

The main presenter from the staff was Mr. Brian Richter, Office of Nuclear Reactor Regulation (NRR). He was supported by Frank Gillespie, NRR, and Geary Mizuno, Office of the General Counsel (OGC). Mr. Richter provided background and presented the proposed criteria for the treatment of individual requirements in a regulatory analysis.

During the above discussions, the NRC staff and the ACRS Members made the following points:

- Mr. Richter stated that the staff is interested in the Committee's comments on proposed criteria for the treatment of individual requirements in a regulatory analysis before forwarding them to the Commission. Mr. Richter said that the proposed criteria had been issued for public comment and that the public comment period ended July 2, 2003. The staff received one comment from NEI.

- Dr. Kress recalled that a cost-benefit analysis performed during the risk-informing of hydrogen control requirements showed that an alternative power supply for hydrogen igniters was indeterminate. He asked how such a case would be handled with the proposed criteria. Mr. Gillespie explained that excess benefit derived from eliminating the design basis hydrogen source term would not be applied to the excess cost associated with an additional power supply for hydrogen igniters.

- Mr. Richter stated that the proposed criteria were: (1) if an individual requirement is "necessary" (i.e., it is needed in order to meet the objectives of the rule or maintain consistency with Commission policies), it does not need to be analyzed separately, and (2) if an individual requirement is supportive but not necessary, it should be included only if it makes the bundled initiative more cost-beneficial.

- Dr. Kress asked how one would determine if an individual requirement was "necessary." Mr. Richter responded that an individual requirement is considered "necessary" if it is integral to the purpose of the rule.

- Mr. Leitch referred to the latest revision to the hydrogen rulemaking and asked if the two technical issues were separated because they dealt with phenomena and different time frames. Mr. Richter answered in the affirmative. Dr. Bonaca followed up by asking if hydrogen monitoring was retained because it was deemed a fundamental part of the rule and, therefore, it did not require a separate cost beneficial analysis. Mr. Richter answered in the affirmative.

- Mr. Gillespie used the 10 CFR 50.69 rulemaking as an example of "necessary." He said the industry might say that the need for a quality probabilistic risk assessment (PRA) in and of itself is not cost justifiable. The staff considers a quality PRA as being an integral part of the rule.

- Dr. Kress asked what was meant by the bullet lack of scrutable guidance by the NRC. Mr. Richter said that NEI feels that to much subjective judgement is allowed when determining the possibilities.

- Mr. Richter commented that the Office of Research (RES) wants the treatment of uncertainties in the Regulatory Analysis Guidelines to more closely conform to OMB's Information Quality Guidelines. This position is contained in COMSECY-02-0037, dated July 31, 2002.

- Mr. Richter ended the presentation by stating that the staff intends to forward the proposed criteria to the Commission as a revision to the Regulatory Analysis Guidelines and then issue Revision 4 to NUREG/BR-0058.

- Dr. Powers asked about international use of risk-informed inservice inspection (ISI). Ken Balkey of Westinghouse said he was aware that the French have developed their own methodology. Mr. Balkey said that Spain is following very closely to the NRC regulations and that they use the ASME code directly. Other countries in Europe are still evaluating either method for application. There are trial applications in Switzerland and Sweden, where they have looked at both. The Japanese are still deciding, and they have not made any movement towards a risk-based inspection effort. Korea has followed the lead of the United States and they are using that as the pilot for their plants.

- Dr. Powers asked what is the difference in the French methodology. Mr. Ali committed to looking into the differences and trying to find out what the French are doing.

- Mr. Leitch questioned if the risk-informed ISI program is approved for a 10-year interval. Mr. Ali answered in the affirmative.

<u>Committee Action</u>

The Committee issued a report to the Commission, dated July 17, 2003, on this matter. In its report, the Committee concluded that the proposed criteria for the treatment of individual requirements in a regulatory analysis were responsive to the Commission's Staff Requirements Memorandum dated December 31, 2001.

V. ESBWR Pre-Application Review (Open)

[Note: Dr. Medhat El-Zeftawy was the Designated Federal Official for this portion of the meeting.]

Dr. Thomas Kress, Future Plant Designs Subcommittee Chairman, stated that the Committee would hear presentations from representatives of the NRC staff and the General Electric (GE) Company regarding the design aspects of the Economic and Simplified Boiling Water Reactor (ESBWR) design.

Ms. Amy Cubbage, Project Manager/New Reactor Licensing Projects-NRR, outlined the ESBWR pre-application scope. This included TRACG application for ESBWR LOCA and containment analysis; TRACG code qualification; test and analysis program description and phenomena identification and ranking tables; and an ESBWR scaling report. The NRC staff plans to issue a safety evaluation report on TRACG application and testing program. The ESBWR design certification scope included a design, transients, anticipated transients without scram (ATWS), stability, LOCA events, severe accidents, and PRAs.

Dr. Atam Rao, Project Manager/GE Nuclear Energy, stated that the ESBWR is a 1380 Mwe boiling water reactor with improved safety margins and passive safety systems. The ESBWR design is based on the GE Simplified Boiling Water Reactor (SBWR) and the Advanced Boiling Water Reactor (ABWR) components with natural circulation and passive safety systems. All pipes and valves are inside containment with significant reduction in systems; components such as pumps, motors, controls, heat exchangers, and buildings. The ESBWR apply load following through control rod drives with minimal impact on maintenance.

Dr. Rao indicated that GE will apply the 15+ year comprehensive technology for the approval of the use of TRACG code that includes the vessel response to pipe break, containment response to pipe break, vessel response to anticipated operational occurrences, and plant response to ATWS and normal operation stability.

The ESBWR design has several diverse means of decay heat removal. The passive safety systems have simplified the plant design and evaluations. The ESBWR has a low parameter uncertainty and substantial safety margins. The ESBWR has a large vessel with greater water inventory that results in improved plant LOCA performance. It also has a larger steam volume for improved transient performance. The evolution of the ESBWR is within a small range to minimize operational risks. However, regulatory challenges need to be addressed and the use of the NRC approved SBWR and ABWR programs and tests need to be reviewed.

Committee Action

This briefing was for the Committee's information. The Committee will continue to follow-up on this matter during future meetings.

VI. Expert Elicitation in Support of Risk-Informing 10 CFR 50.46 (Open)

[Note: Mr. Michael R. Snodderly was the Designated Federal Official for this portion of the meeting.]

Dr. William Shack, the cognizant Committee member for this issue, introduced the topic. He reminded the Committee of the Commission's SRM, dated March 31, 2003, that directed the staff to conduct a practical reconciliation of loss-of-coolant accident (LOCA) frequency distributions by (1) expert use of service-data, (2) Probabilistic Fracture Mechanics (PFM), and (3) expert elicitation to converge the results. Dr. Shack explained that RES is currently conducting this expert elicitation and the Committee would be briefed on how the expert elicitation is being conducted.

NRC Staff Presentation

The main presenter from the staff was Mr. Robert Tregoning, RES. He was supported by Scott Newberry and Dan Dorman, RES, and Eileen McKenna, NRR. Mr. Tregoning provided background and presented the staff's approach for developing LOCA frequencies as a function of leak rate and operating time considering both piping and non-piping contributions for all modes of plant operation.

During the above discussions, the NRC staff and the ACRS members made the following points:

- Mr. Tregoning pointed out that the panel is not trying to estimate the frequency of external events, including sabotage. Rather, the panel is defining loadings and assessing their impact on the piping. Therefore, what caused the loading is not as important as the loadings impact on the piping.

- Mr. Tregoning stated that they have 12 experts. He explained that the panel was solicited from industry, academia, national laboratories, contracting agencies, other government agencies, and international agencies. He said that most of the expertise was in the materials area but they also had experts in plant operation, piping, thermal hydraulics, stress analysis, and nondestructive evaluation. The final panel of 12 were chosen based on their broad relevant expertise and to ensure a diversity of opinion, expertise, and backgrounds. Mr. Rosen commented that the lack of industry representation was very disturbing and could lead to results that the industry will not accept. Dr. Bonaca and Dr. Shack thought that there was sufficient industry representation in the form of vendors and consultants who often represent industry interests.

- Dr. Wallis asked how seismic events, terrorism and sabotage were being considered. Mr. Newberry said the Commission is still looking at the guidance needed on all rule making activities with respect to terrorism. With regard to seismic events, Mr. Tregoning said that the elicitation will generate LOCA frequencies as a function of break size. Loadings corresponding to particular break sizes will be estimated. The likelihood of the loading will not be estimated but given a particular loading, resulting in a break size that will be estimated.

- Dr. Powers asked if the operating data considered non-nuclear systems. Mr. Tregoning said that the operating data only considered worldwide nuclear experience.

- Mr. Tregoning clarified for Dr. Apostolakis that the expert panel was estimating conditional LOCA probabilities. Dr. Shack clarified that if you apply a certain load to a piping system there is a probability that it will fail.

- Dr. Wallis questioned why leak rates are in gallons per minute which is applicable to a two-phase system. Mr. Tregoning responded that the break size in gallons per minute really referred to the make up rate required.

- Dr. Apostolakis recommended that instead of individual elicitations they should hold group elicitations so that the individuals can take advantage of each others expertise. Mr. Abramson responded that they have members with a broad range of experience. He said that they are trying to avoid the results being biased by a particular individual. He said that the members have had the benefit of each others expertise in the development of the base cases. This points out the importance of the members understanding the base cases so that they can make more informed relative judgements.

- Mr. Tregoning explained that there are five base cases and that the LOCA frequency contribution (per year) of each set of base case conditions will be calculated as a function of leak rate and operating time. Four panel members were chosen to perform the base case calculations. Two used operating experience and two used probabilistic fracture mechanics.

- Mr. Abramson explained that the base case is for small LOCAs, where there is the most data, and then the experts will be asked to consider comparable materials and degradation mechanisms and then estimate the chances of a large LOCA. Dr. Apostolakis agreed with this approach.

- Dr. Apostolakis suggested that the staff provide an example with numbers that would walk one through the whole exercise. Mr. Tregoning said that the staff would be happy to do so in the future.

Committee Action

This was an information briefing. The Committee agreed that a future subcommittee meeting on the results of the expert elicitation was needed. The Committee plans to work closely with the staff in the development of the proposed rule change.

X. Executive Session (Open)

[Note: Dr. John T. Larkins was the Designated Federal Official for this portion of the meeting.]

A. Reconciliation of ACRS Comments and Recommendations

[Note: Mr. Sam Duraiswamy was the Designated Federal Official for this portion of the meeting.]

- The Committee considered the response from the EDO dated July 2, 2003, to the ACRS report dated May 16, 2003, concerning the Proposed Revisions of Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping," and the associated SRP Chapter 3.9.8.

  The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the response from the EDO dated June 23, 2003, to the ACRS report dated April 17, 2003, concerning the "Proposed NRC Generic Letter 2003-XX: Control Room Habitability."

  The Committee decided that it was satisfied with the EDO's response.

B. Report on the Meeting of the Planning and Procedures Subcommittee (Open)

The Committee heard a report from the ACRS Chairman and the Executive Director, ACRS, regarding the Planning and Procedures Subcommittee meeting held on July 8, 2003. The following items were discussed:

- Review of the Member Assignments and Priorities for ACRS Reports and Letters for the July ACRS meeting

  Member assignments and priorities for ACRS reports and letters for the July ACRS meeting were discussed. Reports and letters that would benefit from additional consideration at a future ACRS meeting were considered.

- Anticipated Workload for ACRS Members

  The anticipated workload for ACRS members through October 2003 was addressed. The objectives were:

  - Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate

  - Manage the members' workload for these meetings

  - Plan and schedule items for ACRS discussion of topical and emerging issues

  During this session, the Subcommittee also discussed and developed recommendations on items included in the Future Activities List.

- Meeting with the NRC Commissioners

  The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:30 a.m. on Wednesday, October 1, 2003, to discuss items of mutual interest. Topics proposed by the Planning and Procedures Subcommittee for this meeting are as follows:

  I. Overview (MVB)
  Differences in regulatory requirements between U.S. and other countries -- status report

a) Safeguards and security
b) Future ACRS activities
c) Risk-informing 10 CFR 50.46 and proposed 10 CFR 50.69
d) License renewal activities
e) Preapplication review of ESBWR design
f) Power uprate review standard

II. Advancement of PRA technology in risk-informed decisionmaking (GEA)
III. Safety Culture (GEA)
IV. Mixed Oxide Fuel Fabrication Facility (DAP)

The Planning and Procedures Subcommittee considered eliminating these two items:

V. Interim review of AP1000 design (TSK/GBW)
VI. Reactor oversight process (JDS)

The Committee should select the topics during the July ACRS meeting. Subsequently, they will be sent to the Commission for approval. Since there is no ACRS meeting in August, the Committee should start preparing for the meeting with the Commissioners during its September meeting. To support this, cognizant members should complete their presentation slides in August.

The October ACRS meeting was previously scheduled for Thursday, October 2 through Saturday, October 4. Since the meeting now starts on Wednesday, October 1, the Committee should decide whether to have a four-day meeting (October 1-4, 2003).

- <u>A Critical Review of the PIRT Process</u>

The phenomena identification and ranking table (PIRT) process was originally formulated, as a major step in the code scaling, applicability and uncertainty (CSAU) evaluation methodology, to support a revised emergency core cooling system (ECCS) rule for light water reactors. This revised ECCS rule (10 CFR 50.46) was issued in September 1988 and allows, as an option, the use of best estimate plus uncertainty methods in safety analysis. The CSAU evaluation methodology was developed to demonstrate the feasibility of the best estimate plus uncertainty approach. The objective of the PIRT process was to define plant behavior in the context of identifying the relative importance of systems, components, processes, and phenomena.

The PIRT process, with some variations, has been used in many more applications than was originally envisioned. These applications include development of experimental programs and safety analysis requirements for proposed advanced light water reactors, identification of thermal-hydraulic phenomena of importance to pressurized thermal shock (PTS) evaluation, assessment of the adequacy of the planned research programs in addressing the high burnup and new cladding alloy issues, support to resolution of generic safety issues (GSIs) and providing technical guidance in allowing burnup credit in the criticality safety analysis of spent fuel in transport and storage configurations. The NRC Office of Nuclear Regulatory Research also plans to use the PIRT process for identifying and prioritizing the research needs to develop regulatory infrastructure including data, codes and standards, and analytical tools in support of regulatory review of advanced reactor applications.

In view of the widespread use of the PIRT process and its role in prioritization of research needs to address reactor safety technical issues, it is important to provide lessons learned from the past several years of experience with the PIRT process and to identify potential improvements for future PIRT development. Dr. Nourbakhsh provided a presentation to the Committee that reviewed the PIRT process and its prior applications and provided suggestions for enhancement of the process. Use of system dynamics techniques, such as influence diagrams, offers an attractive alternative for developing a phenomena identification and ranking table, which is the principal product of the PIRT process. The use of influence diagrams as a comprehensive framework to identify and prioritize the physical processes which need to be addressed for resolving a technical issue were also discussed.

- <u>Comments on NUREG/CR-6813, Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decisionmaking</u>

We recently published NUREG/CR-6813 prepared by Mr. Fleming under a contract with the ACRS/NRC. Mr. Lochbaum, Union of Concerned Scientists, sent comments on this report to the NRC Office of Public Affairs (OPA). Mr. Fleming prepared a response to Mr. Lochbaum, addressing each comment made by Mr. Lochbaum and sent it to Dr. Nourbakhsh. Mr. Lochbaum's comments and Mr. Fleming's response were e-mailed to all members by Dr. Nourbakhsh on May 5, 2003. The ACRS Executive Director e-mailed Mr. Fleming's response to OPA, NRR, and Mr. Lochbaum on May 5, 2003.

On July 7, 2003, the EDO submitted comments on Mr. Fleming's report. In summary, the EDO stated:

The author of NUREG/CR-6813 identified some key issues that should be addressed to enhance the use of PRA for risk-informed decisionmaking. While the NRC staff is aware of and is addressing these key issues, the staff would like to note that it is not in full agreement with all of the characterizations of the current state of PRA technology and its use. In particular, the staff is not in full agreement with some of the author's views expressed in the report regarding the Davis-Besse vessel head degradation issue.

- Meeting with the Executive Director for Operations

  The members of the Planning and Procedures Subcommittee met with the EDO and his deputies during lunch on Friday, July 11, 2003, to discuss items of mutual interest, including the following:

  - Differing views between the ACRS and the NRC staff on Reactor Oversight Process
  - NRC staff process for tracking commitments made by the EDO and staff in response to ACRS comments and recommendations
  - Timely submittal of documents for ACRS review
  - Staff Requirements Memorandum on risk-Informing 10 CFR 50.46
  - Safeguards and Security matters
  - Comments on NUREG/CR-6813, Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decisionmaking
  - Safety Culture

- Member Issues

  — Letter from Union of Concerned Scientists to Chairman Diaz

  Mr. Lochbaum, UCS, asked that the attached letter be forwarded to the ACRS regarding resolution of GSI-191 for Committee information and comment.

  — NRC System Simulation Capability

A memorandum from V. Ransom regarding NRC System Simulation Capability was discussed.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 505th ACRS Meeting, September 10-13, 2003.

The 504th ACRS meeting was adjourned at 6:15 p.m. on July 11, 2003.

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

August 27, 2003

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador *Sherry Meador*
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 504th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - JULY 9-11, 2003

Enclosed are the proposed minutes of the 504th meeting of the ACRS. This draft is being provided to give you an opportunity to review the record of this meeting and provide comments. Your comments will be incorporated into the final certified set of minutes as appropriate, which will be distributed within six (6) working days from the date of this memorandum.

Attachment:
As stated

reading room: *http://www.nrc.gov/reading-rm.html.* Any comments of Federal, State, and local agencies, Indian tribes or other interested persons will be made available for public inspection when received. Documents may also be obtained from NRC's Public Document Room located at U.S. Nuclear Regulatory Commission Headquarters, 11555 Rockville Pike (first floor), Rockville, Maryland. For those without access to the Internet, paper copies of any electronic documents may be obtained for a fee by contacting the NRC's Public Document Room at 1–800–397–4209.

SUPPLEMENTARY INFORMATION: A Settlement Agreement dated October 17, 1995, among the DOE, the U.S. Navy, and the State of Idaho requires, among other things, the transfer and dry storage of SNF until it can be removed from Idaho. As part of it's efforts to meet the Settlement Agreement, the DOE has contracted with FWENC to design, license, construct, and operate the proposed Idaho Spent Fuel Facility for portions of the SNF currently in storage at the INEEL. If approved, FWENC will be issued an NRC license, under the provisions of 10 CFR part 72, to receive, transfer, and store SNF. The proposed facility would store SNF and associated radioactive material from the Peach Bottom Unit 1 High-Temperature Gas-Cooled Reactor, the Shippingport Atomic Power Station, and various Training, Research, and Isotope reactors built by General Atomics (TRIGA reactors). The majority of this SNF is currently in storage at the Idaho Nuclear Technology Center located on the INEEL immediately adjacent to the proposed facility. DOE plans to transfer the SNF to the proposed facility using existing INEEL and DOE procedures. The transfers to the proposed facility would take place completely within the boundaries of the INEEL. Upon arrival at the proposed facility, the SNF would be (1) remotely removed from the containers in which it is currently stored, (2) visually inspected, (3) inventoried, (4) placed into new storage canisters, and (5) placed into interim dry storage.

The DEIS for the proposed Idaho Spent Fuel Facility was prepared by the staff of the NRC and its contractor, Center for Nuclear Waste Regulatory Analyses, in compliance with the National Environmental Policy Act (NEPA), and the NRC's regulations for implementing NEPA (10 CFR part 51). The proposed action involves a decision by NRC of whether to issue a license under the provisions of 10 CFR part 72 that would authorize FWENC to receive, transfer, and store SNF and associated radioactive materials at the proposed facility.

NRC published a Notice of Intent to prepare an EIS for the proposed Idaho Spent Fuel Facility and to conduct a scoping process in the **Federal Register** on July 26, 2002, (67 FR 48953). The NRC accepted scoping comments through September 16, 2002, and subsequently issued a Scoping Summary Report on December 2, 2002.

The DEIS describes the proposed action and alternatives to the proposed action, including the no-action alternative. The DEIS assesses the impacts of the proposed action and its alternatives on human health, air quality, water resources, waste management, geology, noise, ecology, land use, cultural resources, socioeconomics, accident impacts, and environmental justice. Additionally, the DEIS analyzes and compares the costs and benefits of the proposed action.

Based on the preliminary evaluation in the DEIS, the NRC environmental review staff have concluded that the proposed action will have small effects on the public and existing environment and should be approved. The DEIS is a preliminary analysis of the environmental impacts of the proposed action and its alternatives. The Final EIS and any decision documentation regarding the proposed action will not be issued until public comments on the DEIS have been received and evaluated. Notice of the availability of the Final EIS will be published in the **Federal Register.**

Dated at Rockville, Maryland, this 19th day of June, 2003.

For the Nuclear Regulatory Commission.

Lawrence E. Kokajko,

Acting Chief, Environmental and Performance Assessment Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 03–16174 Filed 6–25–03; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards; Meeting Notice

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on July 9–11, 2003, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, November 20, 2002 (67 FR 70094).

Wednesday, July 9, 2003, Commissioners' Conference Room O–1G16, One White Flint North, Rockville, Maryland

8:30 a.m.–6:30 p.m.: Safeguards and Security (Closed)—The Committee will meet with representatives of the NRC staff, Nuclear Energy Institute (NEI), and their contractors to discuss safeguards and security matters, including Commission papers on risk-informed guidance for vulnerability assessment and on risk-informed decisionmaking, integration of the results of the vulnerability studies, potential vulnerability to sabotage of spent fuel storage facilities, and NEI-sponsored work in the area of safeguards and security. Also, the Committee will discuss a proposed ACRS report on safeguards and security matters.

Thursday, July 10, 2003, Conference Room T–2B3, Two White Flint North Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–10:30 a.m.: ESBWR Pre-Application Review (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the General Electric Company regarding design aspects of the Economic and Simplified Boiling Water Reactor (ESBWR) design and requests for additional information submitted by the staff.

10:45 a.m.–11:45 a.m.: Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding the proposed criteria for treatment of individual requirements in regulatory analyses and related matters.

12:45 p.m.–2:45 p.m.: Mixed Oxide Fuel Fabrication Facility (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the applicant [Duke Cogema Stone and Webster (DCS)] regarding DCS application to construct a mixed oxide fuel fabrication facility at the Savannah River Site, Aiken, SC., and the resolution of open items.

3 p.m.–4:30 p.m.: Expert Elicitation in Support of Risk-Informing 10 CFR 50.46 (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff with regard to conducting an expert elicitation as directed by the

Commission in the March 31, 2003 Staff Requirements Memorandum related to risk-informing 10 CFR 50.46.

4:45 p.m.–7 p.m.: Preparation of ACRS Reports (Open/Closed)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee will discuss proposed ACRS reports on Safety Culture and on Safeguards and Security matters (Closed). The discussion of the Safeguards and Security report will be held in Room T–8E8.

Friday, July 11, 2003, Conference Room T–2B3, Two White Flint North Rockville, Maryland

8:30 a.m.–8:35 a.m.: Opening Remarks by the ACRS Chairman (Open)—The ACRS Chairman will make opening remarks regarding the conduct of the meeting.

8:35 a.m.–9:30 a.m.: Recent Operating Events (Open)—The Committee will hear a briefing by and hold discussions with representatives of the NRC Office of Nuclear Reactor Regulation on the South Texas Project Reactor Vessel Bottom Head Penetration Leakage.

9:30 a.m.–10:15 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future meetings. Also, it will hear a report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

10:30 a.m.–10:45 a.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations (EDO) to comments and recommendations included in recent ACRS reports and letters. The EDO responses are expected to be made available to the Committee prior to the meeting.

10:45 a.m.–7 p.m.: Preparation of ACRS Reports (Open/Closed)—The Committee will discuss proposed ACRS reports on matters considered during this meeting. In addition, the Committee will discuss proposed ACRS reports on Safety Culture and on Safeguards and Security (Closed). The discussion of the Safeguards and Security report will be held in Room T–8E8.

7 p.m.–7:15 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 11, 2002 (67 FR 63460). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Associate Director for Technical Support named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Associate Director prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Associate Director if such rescheduling would result in major inconvenience.

In accordance with Subsection 10(d) P.L. 92–463, I have determined that it is necessary to close a portion of this meeting noted above to discuss and protect information classified as national security information pursuant to 5 U.S.C. 552b(c)(1).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Dr. Sher Bahadur, Associate Director for Technical Support (301–415–0138), between 7:30 a.m. and 4:15 p.m., ET.

ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at pdr@nrc.gov, or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/* (ACRS & ACNW Mtg schedules/agendas).

Videoteleconferencing service is available for observing open sessions of ACRS meetings. Those wishing to use this service for observing ACRS meetings should contact Mr. Theron Brown, ACRS Audio Visual Technician (301–415–8066), between 7:30 a.m. and 3:45 p.m., ET, at least 10 days before the meeting to ensure the availability of this service. Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

Dated: June 20, 2003.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 03–16177 Filed 6–25–03; 8:45 am]

BILLING CODE 7590–01–P

RAILROAD RETIREMENT BOARD

Agency Forms Submitted for OMB Review

Summary: In accordance with the Paperwork Reduction Act of 1995 (44 U.S.C. chapter 35), the Railroad Retirement Board (RRB) has submitted the following proposal(s) for the collection of information to the Office of Management and Budget for review and approval.

Summary of Proposal(s)

(1) *Collection title:* Employer's Deemed Service Month Questionnaire.

(2) *Form(s) submitted:* GL–99.

(3) *OMB Number:* 3220–0156.

(4) *Expiration date of current OMB clearance:* 9/30/2003.

(5) *Type of request:* Extension of a currently approved collection.

(6) *Respondents:* Business or other for-profit.

(7) *Estimated annual number of respondents:* 150.

(8) *Total annual responses:* 4,000.

(9) *Total annual reporting hours:* 133.

(10) *Collection description:* Under section 3(i) of the Railroad Retirement Act, the Railroad Retirement Board may deem months of service in cases where an employee does not actually work in every month of the year. The collection obtains service and compensation information from railroad employers needed to determine if an employee may be credited with additional months of railroad service.

Additional Information or Comments: Copies of the forms and supporting documents can be obtained from Chuck Mierzwa, the agency clearance officer (312–751–3363).

Comments regarding the information collection should be addressed to Ronald J. Hodapp, Railroad Retirement

APPENDIX II

June 30, 2003

REVISED
SCHEDULE AND OUTLINE FOR DISCUSSION
504th ACRS MEETING
JULY 9-11, 2003

WEDNESDAY, JULY 9, 2003

[The meeting on Wednesday, July 9, 2003 will be closed pursuant to 5 U.S.C. 552b(c)(1)]

5:43
8:30 A.M. - ~~6:30~~ P.M. - Safeguards and Security (Closed) - The Committee will meet with representatives of the NRC staff, Nuclear Energy Institute (NEI), and their contractors to discuss safeguards and security matters, including Commission papers on risk-informed guidance for vulnerability assessment and on risk-informed decisionmaking, integration of the results of the vulnerability studies, potential vulnerability to sabotage of spent fuel storage facilities, and NEI-sponsored work in the area of safeguards and security. Also, the Committee will discuss a proposed ACRS report on safeguards and security matters.

THURSDAY, JULY 10, 2003, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. — Opening Remarks by the ACRS Chairman (Open)

1.1) Opening Statement (MVB/JTL/SD)

1.2) Items of current interest (MVB/SD)

2) 8:35 - ~~10:30~~ 10:50 A.M. — Mixed Oxide Fuel Fabrication Facility (Open) (DAP/MWW)

2.1) Remarks by the Subcommittee Chairman

2.2) Briefing by and discussions with representatives of the NRC staff and the applicant [Duke Cogema Stone and Webster (DCS)] regarding DCS application to construct a mixed oxide fuel fabrication facility at the Savannah River Site, Aiken, SC., associated draft Safety Evaluation Report prepared by the staff, and the resolution of open items.

10:50 - 11:05
~~10:30 - 10:45~~ A.M. — *****BREAK*****

3) ~~10:45 - 11:45~~ A.M. 11:05 - 11:50 — Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses (Open) (TSK/MRS)

3.1) Remarks by the Subcommittee Chairman

3.2) Briefing by and discussions with representatives of the NRC staff regarding the proposed criteria for treatment of individual requirements in regulatory analyses and related matters.

Representatives of the nuclear industry may provide their views, as appropriate.

11:50 - 12:50
~~11:45 - 12:45~~ P.M. — *****LUNCH*****

4) ~~12:45 - 2:45~~ P.M. 12:50 - 2:40 — ESBWR Pre-Application Review (Open) (TSK/MME)

4.1) Remarks by the Subcommittee Chairman

4.2) Briefing by and discussions with representatives of the NRC staff and the General Electric Company regarding the design aspects of the Economic and Simplified Boiling Water Reactor

	2:40-	(ESBWR) design and requests for additional information submitted by the staff.
	~~2:45~~ - 3:00 P.M.	***BREAK***
5)	3:00 - ~~4:30~~ 4:50 P.M.	Expert Elicitation in Support of Risk-Informing 10 CFR 50.46 (Open) (WJS/GBW/MRS) 5.1) Remarks by the Subcommittee Chairman 5.2) Briefing by and discussions with representatives of the NRC staff with regard to conducting an expert elicitation as directed by the Commission in the March 31, 2003 Staff Requirements Memorandum related to risk-informing 10 CFR 50.46.
	4:50-5:05 ~~4:30~~ - ~~4:45~~ P.M.	***BREAK***
6)	4:45 - 7:15 P.M.	Preparation of ACRS Reports (Open/Closed) Discussion of the proposed ACRS reports on:
	5:10-5:20	6.1) Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses (TSK/MRS) FINAL
		6.2) Mixed Oxide Fuel Fabrication Facility (DAP/MWW)
		6.3) ESBWR Pre-Application Review (Tentative) (TSK/MME)
	5:20-6:20	6.4) Safety Culture Report (GEA/MWW)
	6:20-7:00	6.5) Safeguards and Security Matters (Closed) (GEA/RPS). **This session will be held in Room T-8E8.**

FRIDAY, JULY 11, 2003, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7)	8:30 - 8:35 A.M.	Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)
8)	8:35 - ~~9:30~~ 9:50 A.M.	Recent Operating Events (Open) (JDS/GML/MWW) 8.1) Remarks by the Subcommittee Chairman 8.2) Briefing by and discussions with representatives of the NRC Office of Nuclear Reactor Regulation on the South Texas Project Reactor Vessel Bottom Head Penetration Leakage. Representatives of the nuclear industry may provide their views, as appropriate.
9)	~~9:30~~ - ~~10:15~~ A.M. 9:50-10:45	Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD) 9.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings. 9.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.
	10:45-11:00 ~~10:15~~ - ~~10:30~~ A.M.	***BREAK***
10)	~~10:30~~ - ~~10:45~~ A.M. 11:00-11:03	Reconciliation of ACRS Comments and Recommendations (Open) (MVB, et al./SD, et al.) Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.
11)	10:45 - 7:00 P.M. **(~~12:15-1:15~~ 11:45-1:00 P.M. LUNCH)**	Preparation of ACRS Reports (Open) The Committee will continue discussion of the proposed ACRS reports listed under Item 6.
12)	~~7:00~~ - ~~7:15~~ P.M. 1:00-6:15	Miscellaneous (Open) (MVB/JTL) Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

11:15-3:05 Safety Culture Report (FINAL)
11:03-11:15 Fire Protection Larkinsgram (FINAL)
SOIL Larkinsgram (FINAL)
4:20-6:15 Safeguards & Security Matters (FINAL)

Note:

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

APPENDIX III: MEETING ATTENDEES

504th ACRS MEETING
JULY 9-11, 2003

NRC STAFF (July 9, 2003)

M. Cunningham, RES
F. Eltawila, RES
J. Mitchell, RES
S. Newberry, RES
V. Perin, RES
J. Schaperow, RES
J. Strosnider, RES
N. Siu, RES
E. Thornsbury, RES
C. Tinkler, RES
J. Rosenthal, RES
A. Thadani, RES
R. Moody, NRR
R. Sullivan, NRR
C. Holden, NRR
T. Tate, NRR
P. Madden, NRR
R. Palla, NRR
M. Tschlitz, NRR
C. Jackson, OCM/NJD
T. McCartin, NMSS
R. K. Johnson, NMSS
S. Morris, NSIR
J. Aroldsen, NSIR
B. Schlapper, SFPO

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

S. Floyd, NEI
G. Hardy, NEI Consultat
D. Walters, NEI
S. Ashbaugh, LANL
V. Dandini, Sandia
R. Gauntt, Sandia

NRC STAFF (July 10, 2003)

A. Madison, NSIR
R. Landry, NRR
E. Throm, NRR
J. Staudenmeier, RES
S. Lu, NRR
G. Thomas, NRR
M. Razzazier, NRR
S. LaVie, NRR
J. Han, RES
A. Lewis, RES
J. Vora, RES
S. Newberry, RES
H. Hamzehee, RES
R. Tregoning, RES
T. Govan, RES
T. Powell, NMSS
L. Abramson, RES
B. Beach, RES
A. Buslik, RES
E. McKenna, NRR
D. Jackson, RES
J. Kramer, RES
B. Troskoski, NMSS
B. Smith, NMSS
T. C. Johnson, NMSS
B. Ibrahim, NMSS
M. Hilerman, RES
S. Crane, RES
R. Wescott, NMSS
C. Tripp, NMSS
J. Klein, NMSS
F. Burrows, NMSS
K. H. Gibson, NMSS
P. Justus, NMSS
A. Murray, NMSS
W. Smith, NMSS
B. Thomas, NRR
B. Richter, NRR
J. Morris, NMSS
H. Graves, RES
G. Mizuno, OGC

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

S. Thomas, DOE
T. St. Louis, DCS
J. Johnson, DOE
S. Zak, DOE
D. Alberstein, DOE
E. Brabazin, DCS
J. Kurakami, JNFCDI
D. Werder, MPR Assoc.
T. Touchstone, DCS
B. Foster, DCS
A. Polensky, Morgan, Lewis & Backus
S. Traiforos, Link Technology
A. Tabatabai, Link Technology
A. Rao, GE
R. Gamble, GE
P. Negas, GE
L. Hay, SERCH/Bechtel
N. Chapman, SERCH/Bechtel

NRC STAFF (July 11, 2003)
C. Gratton, NRR
S. Monarque, NRR
M. Mitchell, NRR
S. Coffin, NRR
B. Bateman, NRR
B. Cullen, RES
R. Barrett, NRR
J. Vora, RES
A. Hiser, NRR
M. Thadani, NRR
R. Gramm, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
S. Traiforos, Link
A. Tabatabai, Link
S. Thomas, STPNOC
M. McBurnett, STPNOC
D. Hurne, McGraw-Hill ✓

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

August 13, 2003

SCHEDULE AND OUTLINE FOR DISCUSSION
505th ACRS MEETING
SEPTEMBER 10-13, 2003

WEDNESDAY, SEPTEMBER 10, 2003

[The meeting on Wednesday, September 10, 2003 will be closed pursuant to 5 U.S.C. 552b(c)(1)]

1) 10:15 - 10:20 A.M. Opening Remarks by the ACRS Chairman (Closed) (MVB/JTL)

2) 10:20 - 7:00 P.M. **(12:30-1:30 P.M. LUNCH)** Safeguards and Security Matters (Closed) (MVB/TSK/RPS/RKM)
- 2.1) Remarks by the Subcommittee Chairman
- 2.2) Briefing by and discussions with representatives of the Office of Nuclear Regulatory Research and the Office of Nuclear Security and Incident Response to discuss safeguards and security matters. Also, the Committee will discuss a proposed ACRS report on safeguards and security matters.

THURSDAY, SEPTEMBER 11, 2003, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

3) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open)
- 3.1) Opening Statement (MVB/JTL/SD)
- 3.2) Items of current interest (MVB/SD)

4) 8:35 - 10:00 A.M. Final Review of the St. Lucie License Renewal Application (Open) (MVB/GML/BPJ/SD)
- 4.1) Remarks by the Subcommittee Chairman
- 4.2) Briefing by and discussions with representatives of NRC staff and Florida Power and Light Company regarding the St. Lucie license renewal application and the associated Final Safety Evaluation Report prepared by the staff.

10:00 - 10:15 A.M. *****BREAK*****

5) 10:15 - 11:30 A.M. Draft Final Regulatory Guide DG-1122, "Determining the Technical Adequacy of PRA Results for Risk-Informed Activities" (Open) (GEA/MRS)
- 5.1) Remarks by the Subcommittee Chairman
- 5.2) Briefing by and discussions with representatives of NRC staff regarding the draft final version of Regulatory Guide DG-1122.

Representatives of the nuclear industry may provide their views, as appropriate.

11:30 - 12:30 P.M. *****LUNCH*****

6) 12:30 - 2:00 P.M. Technical Assessment and Proposed Recommendations for Resolving GSI-186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants" (Open) (JDS/MWW)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of NRC staff regarding the technical assessment and recommendations proposed by the Office of Nuclear Regulatory Research for resolving GSI-186.

Representatives of the nuclear industry may provide their views, as appropriate.

2:00 - 2:15 P.M. *****BREAK*****

7) 2:15 - 3:45 P.M. Draft Final Review Standard for Reviewing Core Power Uprate Applications (Open) (VHR/RC)

7.1) Remarks by the Acting Subcommittee Chairman

7.2) Briefing by and discussions with representatives of NRC staff regarding the draft final review standard to be used by the staff for reviewing core power uprate applications.

Representatives of the nuclear industry may provide their views, as appropriate.

3:45 - 4:00 P.M. *****BREAK*****

8) 4:00 - 5:15 P.M. Draft Final Revision 3 to Regulatory Guide 1.82 (DG-1107), "Water Sources for Long-Term Recirculation Cooling Following a LOCA" (Open) (GBW/RC)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding draft final revision 3 to Regulatory Guide 1.82 (DG-1107) including resolution of public comments, and related matters.

Representatives of the nuclear industry may provide their views, as appropriate.

9) 5:15 - 6:00 P.M. Review of PIRT Process (Open) (GEA/HPN)
Briefing by Dr. Nourbakhsh, ACRS Senior Fellow, regarding his review of the phenomena identification and ranking table (PIRT) process.

6:00 - 6:15 P.M. *****BREAK*****

10) 6:15 - 7:30 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of proposed ACRS reports on:
10.1) Final Review of the St. Lucie License Renewal Application (MVB/GML/BPJ/SD)
10.2) Draft Final Regulatory Guide DG-1122 on PRA Quality (GEA/MRS) (Tentative)
10.3) Proposed Recommendations for Resolving GSI-186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants" (JDS/MWW)
10.4) Draft Final Review Standard for Reviewing Core Power Uprate Applications (VHR/RC)
10.5) Draft Final Revision 3 to Regulatory Guide 1.82 (DG-1107), "Water Sources for Long-Term Recirculation Cooling Following a LOCA" (GBW/RC)
10.6) Safeguards and Security Matters (Closed) (MVB/TSK/RPS/RKM)

FRIDAY, SEPTEMBER 12, 2003, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

11) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)

12) 8:35 - 9:30 A.M. Draft Final Revision 1 to Regulatory Guide 1.53, "Application of the Single Failure Criterion to Safety Systems" (Open) (WJS/MRS)
12.1) Remarks by the Subcommittee Chairman
12.2) Briefing by and discussions with representatives of the NRC staff regarding the draft final revision 1 to Regulatory Guide 1.53.

Representatives of the nuclear industry may provide their views, as appropriate.

13) 9:30 - 11:15 A.M. **(10:00-10:15 A.M. BREAK)** Preparation for Meeting with the NRC Commissioners (Open) (MVB/JTL)
The Committee will discuss proposed topics for discussion during the ACRS meeting with the NRC Commissioners which is scheduled to be held on Wednesday, October 1, 2003, between 9:30 and 11:30 a.m.

14) 11:15 - 11:30 A.M. Subcommittee Report on Fire Protection Issues (Open) (SLR/MDS)
The Fire Protection Subcommittee Chairman will provide a brief report on matters discussed during the September 9, 2003 meeting.

15) 11:30 - 12:15 P.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/RPS)
15.1) Discussion of the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings.

15.2) Report of the Planning and Procedures Subcommittee on matters related to the conduct of ACRS business, including anticipated workload and member assignments.

16) 12:15 - 12:30 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB, et al./SD, et al.)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

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

17) 1:30 - 7:30 P.M. Preparation of ACRS Reports (Open/Closed)
Discussion of the proposed ACRS reports on:

17.1) Final Review of the St. Lucie License Renewal Application (MVB/GML/BPJ/SD)
17.2) Draft Final Regulatory Guide DG-1122 on PRA Quality (GEA/MRS) (Tentative)
17.3) Proposed Recommendations for Resolving GSI-186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants" (JDS/MWW)
17.4) Draft Final Review Standard for Reviewing Core Power Uprate Applications (VHR/RC)
17.5) Draft Final Revision 3 to Regulatory Guide 1.82 (DG-1107), "Water Sources for Long-Term Recirculation Cooling Following a LOCA" (GBW/RC)
17.6) Safeguards and Security Matters (Closed) (MVB/TSK/RPS/RKM)
17.7) Draft Final Revision 1 to Regulatory Guide 1.53 regarding Single Failure Criterion (WJS/MRS)

SATURDAY, SEPTEMBER 13, 2003, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

18) 8:30 - 1:00 P.M. Preparation of ACRS Reports (Open/Closed)
The Committee will continue discussion of the proposed ACRS reports listed under Item 17.

19) 1:00 - 1:15 P.M. Miscellaneous (Open) (MVB/JTL)
Discussion of matters related to the conduct of Committee activities and matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTE:

- **Presentation time should not exceed 50 percent of the total time allocated for a specific item. The remaining 50 percent of the time is reserved for discussion.**

- **Thirty-Five (35) copies of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
504th ACRS MEETING
JULY 9-11, 2003

[Note: Some documents listed below may have been provided or prepared for Committee use only. These documents must be reviewed prior to release to the public.]

MEETING HANDOUTS

AGENDA ITEM NO.	DOCUMENTS
1	Opening Remarks by the ACRS Chairman 1. Items of Interest, dated July 9-11, 2003
2	Mixed Oxide Fuel Fabrication Facility 2. Mixed Oxide Fuel Fabrication Facility presentation by D. Persinko, NMSS [Viewgraphs] 3. MFFF General Facility Mission and Layout presentation by Duke Cogema, Stone & Webster [Viewgraphs]
3	Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses 4. Proposed Criteria for the Treatment of Individual Requirements in a Regulatory Analysis presentation by B. Richter, NRR [Viewgraphs]
4	ESBWR Pre-Application Review 5. ESBWR Pre-Application Review presentation by A. Cubbage, NRR [Viewgraphs] 6. ESBWR Design Overview and Technology Closure presentation by General Electric [Viewgraphs]
5	Expert Elicitation in Support of Risk-Informing 10 CFR 50.46 7. Expert Elicitation in Support of Risk-Informing 10 CFR 50.46 presentation by Office of RES [Viewgraphs]
8	Recent Operating Events 8. South Texas Project Unit 1 Bottom Mounted Instrumentation Nozzle Leakage Issue presentation by M. Mitchell, NRR [Viewgraphs] 9. South Texas diagram of Penetrations 1 and 46 [Viewgraphs] 10. Operating Events presentation by Graham Leitch, ACRS Member [Handout]

9 <u>Future ACRS Activities/Report of the Planning and Procedures Subcommittee</u>

11. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - July 8, 2003 [Handout #9.1]

10 <u>Reconciliation of ACRS Comments and Recommendations</u>

12. Reconciliation of ACRS Comments and Recommendations [Handout #10.1]

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
2	Mixed Oxide (MOX) Fuel Fabrication Facility (FFF) 1. Table of Contents 2. Proposed Schedule 3. Status Report 5. May 2003 Monthly Open Item Status Report 6. Environmental Report Revisions 1 & 2, Changes (provides some accident scenarios and consequences such as loss of confinement, internal fires, explosions, load handling, criticality, and chemical releases)
3	Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analyses 6. Table of Contents 7. Proposed Schedule 8. Status Report 9. Federal Register, Volume 68, Number 75, Page 19162-19166, Subject: Regulatory Analysis Guidelines: Proposed Criteria for the Treatment of Individual Requirements in Regulatory Analysis
4	ESBWR Pre-Application Review 10. Table of Contents 11. Proposed Schedule 12. Status Report 13. ESBWR Program and Regulatory Challenges, A. Rao, GE Nuclear Energy, USA
5	Expert Elicitation in Support of Risk-Informing 10 CFR 50.46 14. Table of Contents 15. Proposed Schedule 16. Status Report 17. Staff Requirements Memorandum dated March 31, 2003, from Annette L. Vietti-Cook, Secretary, to William D. Travers, EDO, Subject: Staff Requirements - SECY-02-057, Update to SECY-01-0133, "Fourth Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.46 (ECCS Acceptance Criteria)"

8 Recent Operating Events

18. Table of Contents
19. Proposed Schedule
20. Information Report
21. Licensee Event Report (LER), June 11, 2003

SAFEGUARDS AND SECURITY
JULY 9, 2003

8:30 am -
10:15 am
10:30 - 12:30p
12:35

ACRS Members
G. Apostolakis
M. Bonaca
P. Ford
T. Kress
G. Leitch
D. Powers
V. Ransom
S. Rosen
W. Shack
J. Sieber
G. Wallis

ACRS Staff
J. Larkins
S. Bahadur
H. Larson
R. Savio
R. Major
J. Gallo
T. Brown
S. Meador
~~B. J. White~~ M. Kelton

Office of Research
S. Ali
M. Cunningham
D. Dorman
F. Eltawila
T. Jensen-Otsu
A. Kuritzky
M. Mayfield
J. Mitchell
S. Newberry
V. Perin
J. Schaperow
J. Strosnider

R. Shaffer
N. Siu
B. Tegeler
E. Thornsbury
C. Tinkler
C. Trottier
J. Rosenthal
A. Thadani

SFPO
B. Schlapper
B. White
M. Shah
R. Parkhill
C. Bajwa
S. Bush-Goddard
C. Interrante
A. Snyder
J. Guttman
W. Hodges

Office of NRR
R. Moody
R. Sullivan
C. Holden
T. Tate
A. Adams
P. Madden
R. Palla

OCM/NJD
Chris Jackson

Office of NMSS
T. McCartin
D. Esh
A. Campbell
R. K. Johnson

Office of NSIR
R. Zimmerman
M. Weber
A. Tardiff
J. Tomlinson
W. Orders
E. McNeil
S. Stein
W. Orders
G. Tracy
W. Desmond
B. Wetzel
S. Morris
J. Arildsen
D. Gordon
E. Lee
W. Orders

OEDO
J. Shea
M. Fields

NEI
S. Floyd
G. Hardy (Consultant)
B. Nickell (Consultant) — not cleared
D. Walters

LANL
S. Ashbaugh

Sandia
V. Dandini
R. Gauntt

1:35—

SAFEGUARDS AND SECURITY
JULY 9, 2003

ACRS Members
G. Apostolakis
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R. Shaffer
N. Siu
B. Tegeler
E. Thornsbury
C. Tinkler
C. Trottier
J. Rosenthal
A. Thadani

M. ~~Tshlitcz~~ Tschlitz, NRR

SFPO
B. Schlapper
B. White
M. Shah
R. Parkhill
C. Bajwa
S. Bush-Goddard
C. Interrante
A. Snyder
J. Guttman
W. Hodges

Office of NRR
R. Moody
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OCM/NJD
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W. Orders
E. McNeil
S. Stein
W. Orders
G. Tracy
W. Desmond
B. Wetzel
S. Morris
J. Arildsen
D. Gordon
E. Lee
[illegible]

OEDO
J. Shea
M. Fields

NEI
S. Floyd
G. Hardy (Consultant)
B. Nickell (Consultant) Not coming
D. Walters

LANL
S. Ashbaugh

Sandia
V. Dandini
R. Gauntt

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504TH FULL COMMITTEE MEETING
JULY 9-11, 2003

JULY 10, 2003
TODAY'S DATE

NRC STAFF PLEASE SIGN IN BELOW

PLEASE PRINT

NAME	NRC ORGANIZATION
Alan Madison	NSIR
~~SPYROS TRAIFOROS~~	~~ACRS (LINK)~~
~~ALI TABATABAI~~	~~ACRS (LINK)~~
RALPH LANDRY	NRR/DSSA/SRXB
EDWARD D THROM	NRR/DSSA/SPSB
Joseph Staudenmeier	RES/DSARE/SMSAB
Shanlai Lu	NRR/DSSA/SRXB
GEORGE THOMAS	NRR/DSSA/SRXB.
Muhammad M. Razzaque	NRR/DSSA/SRXB
Steve LaVie	NRR/DSSA/SPSB
JAMES HAN	RES/DSARE/SMSAB
Alan Levin	RES/DD
Jit VORA	RES/DET/MEB
Scott Newberry	RES DRAA
HOSSEIN HAMZEHEE	RES/DRAA
ROB TREGONING	RES/DET
Tenia V. Govan	RES/DET/ERAB
Tamara Powell	NMSS/FCSS/SPIB
Lee Abramson	RES/DRAA/PRAB
Bennett Brady	RES/DRAA/OERAB
J ARTHUR BUSLIK	RES/DRAA/PRAB
Eileen McKenna	NRR/DRIP/RPRP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504TH FULL COMMITTEE MEETING
JULY 9-11, 2003

JULY 10, 2003
TODAY'S DATE

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PLEASE PRINT

NAME	NRC ORGANIZATION
Joel Kramer	RES/DSARE/REAHFB
Bill Troskoski	NMSS/FCSS/SPIB
Brian Smith	NMSS/FCSS/SPIB
TC Johnson	NMSS/FCSS
Bakr Ibrahim	NMSS/DWM
Mary K. Hileman	RES/DSARE
Samantha Crane	RES/DET
Rex Wescott	NMSS/FCSS
Christopher S Tripp	NMSS/FCSS/SPIB
Joel Klein	NMSS/FCSS/SPIB
FRED BURROWS	NMSS/FCSS/FM
Kathy Halvey Gibson	NMSS/FCSS/SPIB
PHILIP JUSTUS	NMSS/WM/HLWB
ALEX MURRAY	NMSS/FCSS/SPIB.
WILKINS SMITH	NMSS/FCSS/SPIB
BRIAN THOMAS	NRR/DRIP/RRAS
BRIAN RICHTER	NRR/DRIP/RPRP
JAMES E. MOREY	NMSS/IMNS/RGB
Herman Graves	RES/DET/ERAB
Geary Mizuno	NRC-OGC
[illegible] Jackson	NRC RES/DET

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504TH FULL COMMITTEE MEETING
JULY 9-11, 2003

JULY 10, 2003
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
SAM THOMAS	DOE/NNSA
Tom St. Louis	DCS
JAMIE JOHNSON	DOE
Stephen Zuk	DOE
David Alberstein	DOE
Ed Brabazin	DCS
Junichi Kurakami	Japan Nuclear Fuel Cycle Developement Istitution
DAVID WERDER	MPR ASSOCIATES / DOE
Tommy Touchstone	DCS
Bob Foster	DCS
Alex Polonsky	Morgan, Lewis & Bockius LLP
S. Traiforos,	Link
A. Tabatabai	Link

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504TH FULL COMMITTEE MEETING
JULY 9-11, 2003

JULY 10, 2003
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Atambir Rao	GE
Robert Gamble	GE
[illegible] Ngan	GE
Lane Hay	SERCH Bechtel
Nancy Chapman	SERCH / Bechtel

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504TH FULL COMMITTEE MEETING
JULY 9-11, 2003

JULY 11, 2003
TODAY'S DATE

NRC STAFF PLEASE SIGN IN BELOW

PLEASE PRINT

NAME	NRC ORGANIZATION
SPYROS TRAIFOROS	ACRS (CLINK)
Chris Grattan	NRR/DLPM.
STephen R Monarque	NRR/DLPM
Matthew A. Mitchell	NRR/DE/EMCB
ALI TABATABAI	ACRS (CLINK)
STEPHANIE COFFIN	NRR/DE/EMCB
Bill Bateman	"
Bill Cullen	RES/DET/MEB
Rich BARRETT	NRR/DE
Jit VORA	RES/MEB
Allen Hiser	NRR/DE/EMCB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504TH FULL COMMITTEE MEETING
JULY 9-11, 2003

<u>JULY 11, 2003</u>
Today's Date

ATTENDEES PLEASE SIGN IN BELOW
PLEASE PRINT

NAME	AFFILIATION
Steven Thomas	STPNOC
Mark McBurnett	STPNOC
Mohan Thadani	NRC/NRR/DLPM
Robert Gramm	NRR/DLPM
Daniel Horner	McGraw-Hill

ITEMS OF INTEREST

504th ACRS MEETING

JULY 9-11, 2003

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
504th MEETING
JULY 9-11, 2003

<u>CONGRESSIONAL CORRESPONDENCE</u>

UNITED STATES NUCLEAR REGULATORY COMMISSION

Announcement No. 048

Date: July 7, 2003

To: **All NRC Employees**

SUBJECT: SENIOR MANAGEMENT CHANGES

Announcement No. 44, dated June 17, 2003, informed you of the appointment of William F. Kane to the new position of Deputy Executive Director for Homeland Protection and Preparedness in the Office of the Executive Director for Operations (OEDO). Mr. Kane will execute OEDO responsibilities for homeland protection and preparedness, and carry out day-to-day supervision, guidance and direction for the Office of Nuclear Security and Incident Response. This will include providing oversight across agency lines of authority for all NRC policies, programs and activities related to homeland protection and preparedness, including matters dealing with the homeland security aspects of physical and personnel security, information security, information technology security, domestic and international safeguards, emergency response, and threat and vulnerability assessment.

As a result of Mr. Kane's new appointment, I am pleased to announce the following managerial assignments:

Samuel J. Collins has been named Deputy Executive Director for Reactor Programs in the Office of the Executive Director for Operations (OEDO). Mr. Collins will carry out the day-to-day supervision, guidance and direction for the Offices of Nuclear Reactor Regulation, Enforcement, Investigations, and the regional offices. Mr. Collins joined the NRC in 1980 as a Resident Inspector in Region I and was later selected as a Senior Resident Inspector. In 1983, he was selected as a Section Chief in Region I and later became Chief, Reactor Projects Branch. In 1987, Mr. Collins was selected for the Senior Executive Service (SES) position of Deputy Director, Division of Reactor Projects, Region I. In 1989, he transferred to Region IV where he served in a number of management positions including Director, Division of Reactor Projects; Director, Division of Reactor Safety; Director, Division of Radiation Safety and Safeguards; and Deputy Regional Administrator. In 1997, Mr. Collins was appointed to his most recent position of Director, Office of Nuclear Reactor Regulation (NRR). He is a graduate of the Maine Maritime Academy with a B.S. degree in Engineering.

James E. Dyer has been named Director, Office of Nuclear Reactor Regulation. Mr. Dyer joined the NRC in 1983 as an Inspection Specialist in the former Office of Inspection and Enforcement. From 1987 to 1989, he served as a Senior Operations Engineer and Section Chief in NRR. In 1989, Mr. Dyer joined the staff of the Executive Director for Operations where he served as a Regional Coordinator and Chief of the Regional Operations Staff. In 1990, he was appointed to the SES and served as a Project Director in NRR.

In 1994, Mr. Dyer transferred to Region IV where he served in a number of management positions including Deputy Director, Division of Reactor Projects; Director, Division of Reactor Projects; and Deputy Regional Administrator. In 1999, Mr. Dyer was appointed to his most recent position of Regional Administrator, Region III. Mr. Dyer received a B.S. degree in Chemistry from the University of California and a Master of Business Administration from Frostburg State College.

R. William Borchardt has been named Deputy Director, Office of Nuclear Reactor Regulation. This position has been vacant since the retirement of Jon R. Johnson in January. Mr. Borchardt joined the NRC in 1983 as a Reactor Engineer in Region I. He later served as a Resident Inspector and Senior Resident Inspector. In 1988, Mr. Borchardt transferred to Headquarters where he served in a number of progressively responsible positions including Senior Regional Coordinator in OEDO, and as Technical Assistant and Section Chief in NRR. In 1993, he was selected for the SES position of Project Director, Standardization Project Directorate and in 1995 as Chief, Inspection Program Branch. From 1998 to 1999, Mr. Borchardt was the Deputy Director, Office Enforcement, and in 1999 was appointed as Director. In 2001, he was appointed to his most recent position of Associate Director for Inspection and Programs, NRR. Mr. Borchardt received a B.S. degree in Chemistry from the U.S. Naval Academy.

John W. Craig has been named Associate Director for Inspection and Programs, NRR. Mr. Craig joined the NRC in 1979 in the former Office of Inspection and Enforcement where he held a number of positions including Reactor Engineer, Enforcement Specialist, Senior Reactor Operations Engineer, and Section Chief. In 1987, Mr. Craig was appointed to the SES position of Chief, Plant Systems Branch in NRR, and later served as a Project Director. From 1993 to 2000, he served in a number of management positions in the Office of Nuclear Regulatory Research including Deputy Director, Division of Engineering Technology; Director, Division of Regulatory Applications; and Director, Division of Engineering Technology. In 2000, Mr. Craig was appointed as Assistant for Operations in OEDO. Since April 2003, he has served as Chief of Staff in Chairman Diaz's office. Mr. Craig received a B.S. degree in Nuclear Engineering from the University of Maryland.

James L. Caldwell has been named Regional Administrator, Region III. Mr. Caldwell joined the NRC in 1983 as a Resident Inspector in Region II, and later was selected as a Senior Resident Inspector. In 1990, he transferred to Headquarters where he served as a Senior Regional Coordinator in OEDO and a Section Chief in NRR. In 1994, he was selected for the SES and transferred to Region III where he served in a number of management positions including Deputy Director, Division of Radiation Safety and Safeguards; Deputy Director, Division of Nuclear Materials Safety; Deputy Director, Division of Reactor Projects; and Director, Division of Reactor Projects. In 1997, Mr. Caldwell was selected for his most recent position of Deputy Regional Administrator, Region III. Mr. Caldwell received a B.S. degree in Electrical Engineering from the Virginia Polytechnic Institute and State University.

Geoffrey E. Grant has been named Deputy Regional Administrator, Region III. Mr. Grant joined the NRC in 1986 as a Resident Inspector in Region I, and later was selected as a Senior Resident Inspector. In 1989, he transferred to Headquarters where he served as a Senior Operations Engineer and later a Section Chief in the Division of Licensee Performance and Quality Evaluation in NRR. Following an assignment as a Senior Regional Coordinator in OEDO, Mr. Grant was selected in 1993 for the SES and transferred to Region III where he served as Director, Division of Reactor Safety. In 1997 he assumed his most recent position, Director, Division of Reactor Projects in R III. Mr. Grant received a B.S. degree from the United States Naval Academy in 1972.

These changes will be phased in over the next several weeks. Please join me in congratulating these executives on their new assignments.

/RA/
William D. Travers
Executive Director for Operations

[Top of Page | NRC Internal Home Page | Index of Yellow Announcements]

June 26, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - SECY-03-0047 - POLICY ISSUES RELATED TO LICENSING NON-LIGHT-WATER REACTOR DESIGNS

The Commission has approved the staff's recommendations as outlined in this paper for issues 2, 4, 5, and 7.

The Commission has approved the staff's recommendation for issue 1 on implementation of the Commission's expectations for enhanced safety in future non-light-water reactors, with the exception of accounting for the integrated risk posed by multiple reactors. The staff should provide further details on the options for, and associated impacts of, requiring that modular reactor designs account for the integrated risk posed by multiple reactors. Historically, the NRC has issued operating licenses to sites with as many as three units, granted Construction Permits for four at one site (Shearon Harris), and docketed another application for five at one site (Palo Verde). The staff should review those dockets for relevant historical regulatory positions on these issues, including potential precedents. The staff will need to establish a usable definition of core damage and will need to determine if the concept of large early release frequency is meaningful or if a level 3 risk assessment would be needed.

The staff should consider whether it can accomplish the same goals associated with issue 2 in a more efficient and effective manner by updating the Commission Policy Statement on Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities to include a more explicit discussion of defense-in-depth, risk-informed regulation, and performance-based regulation.

The Commission has disapproved the staff's recommendation for issue 3 to proactively participating in development of and endorsing international codes and standards where such codes and standards have been identified by applicants or pre-applicants for use in their submittals or by staff as needed to fill gaps in the NRC's non-LWR infrastructure. The staff should pursue option (a), specifically to "Review international codes and standards only as part of an application or pre-application review." The staff should gain experience through review of international codes and standards during the pre-application and application reviews of non-LWRs then apply the lessons-learned from these reviews to their activities involving our domestic codes and standards committees.

The Commission has disapproved the staff's recommendation for issue 6 related to the requirement for a pressure retaining containment building. At this time there is insufficient information for the Commission to prejudge the best options and make a decision on the viability of a confinement building. The staff should develop performance requirements and criteria working closely with industry experts (e.g., designers, EPRI, etc.) and other stakeholders regarding options in this area, taking into account such features as core, fuel, and cooling systems design. The staff should pursue the development of functional performance standards and then submit options and recommendations to the Commission on this important policy decision.

cc: Chairman Diaz
Commissioner Dicus
Commissioner McGaffigan
Commissioner Merrifield
OGC
CFO
OCA
OIG
OPA
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR

IN RESPONSE, PLEASE
REFER TO: M030515

June 10, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RESULTS OF AGENCY ACTION REVIEW MEETING, 9:30 A.M., THURSDAY, MAY 15, 2003, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff and representatives of Entergy Operations, Inc. and Greenpeace on the Reactor Oversight Process (ROP) and the results of the Agency Action Review Meeting.

The staff should inform the Commission of significant regional ROP inspection challenges, including sites where there are currently less than two fully qualified inspectors assigned.

The Commission supports flexibility in conducting effective ROP public meetings, including giving consideration to measures such as providing NRC personnel knowledgeable in areas of interest on the national level and allowing the resident inspectors to conduct ROP meetings held at the plant sites, as appropriate.

The staff should inform the Commission of the actions planned to respond to the issues raised by Mr. Riccio in his statement document dated May 15, 2003. The staff should follow the established process for evaluating stakeholder comments to evaluate the ROP changes suggested at the meeting, including increasing the threshold for a degraded cornerstone to three white performance indicators (PIs) or inspection findings.

(EDO) (SECY Suspense: 8/15/03)

In addition, the staff should review the action matrix thresholds to determine if changes are needed to ensure the action matrix categorization adequately reflects the safety significance of PIs and inspection findings. The staff should then provide a recommendation to the Commission in the calendar year 2003 ROP self-assessment report.

(EDO) (SECY Suspense: 3/31/04)

NRC NEWS

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No. S-03-019

REDEFINING THE LARGE BREAK LOSS OF COOLANT ACCIDENT (LOCA)
"... an Idea Whose Time Has Come"

by

Nils J. Diaz, Chairman
U.S. Nuclear Regulatory Commission

Keynote Speech

presented before the Committee on the Safety of Nuclear Installations/Committee on Nuclear Regulatory Activities (CSNI/CNRA) at the Joint CSNI/CNRA Workshop

Zurich, Switzerland
June 23, 2003

I am very pleased to be here with so many technical experts from around the world to discuss a subject which I feel very strongly about. I would like to express my thanks to the NEA committees, the CNRA and the CSNI, and to our Swiss colleagues for organizing and hosting the meeting, and for the opportunity to share my views with you. And of course, to the NRC staff, and Ashok Thadani in particular, for their work in making this meeting on this subject, possible.

Let me say from the beginning that redefining the Large Break LOCA is for me, and I hope for all of us, a significant safety initiative. I cannot stress that fact enough ... a safety initiative. We in the US experienced our most serious reactor accident at Three Mile Island (TMI) in 1979 -- twenty-four years ago, yet still fresh in our memories. The TMI accident was not a Large Break LOCA, it was not the event that we had invested so much of our time and technical resources in. The TMI accident was a small LOCA, an event given significantly less attention because of the overwhelming amount of attention on the Large Break LOCA concern. During the four decades since nuclear power plants began operation, each of our nations has experienced important reactor safety events, yet none were Large Break LOCAs. The only power or production reactor accident - Chernobyl - that resulted in loss of life on-site and massive radioactivity releases was many things but not a Large Break LOCA. All the other reactor safety

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events include occurrences such as small LOCAs, or loss of decay heat removal or fires or reactivity events. With today's improved know-how, shouldn't we be focused on the right safety issues? Shouldn't we assure the public, whom we are protecting, that our attention and the attention of our licensees is focused on the most important issues and activities for preserving their health and safety? I believe the record shows that we do a good job, but we can do a better job by using what we now know is more safety-focused, cognizant of the past and of present and future needs, and dedicated to the task at hand: protection of public health and safety and the environment.

I believe the nuclear regulatory agencies, cognizant of the present safety experiences and assessment capabilities, need to take the next step. The licensees and reactor vendors cannot change their focus until we change ours. That's a fact. Regulation and technology need to progress in parallel, in phase. And in this particular case, the regulators are currently lagging the technological capabilities. We also need to recognize, consider, and address the technical, legal, and political impediments to change, so whatever we do has to be right, scrutinized and well communicated.

Let me remind you of a quote from the well known 19th century author Victor Hugo, who said,

"Nothing else in the world ... is so powerful as an idea whose time has come."

Well, I believe that redefining the Large Break LOCA through a risk-informed and performance-based approach, <u>is</u> an idea whose time has come. And I am not overestimating its importance; it plays large in many areas. The double-ended rupture of the largest pipe in the RCS should be moved from the design basis to severe accident management space. This change will not create a void, it will create the opportunity for safety improvements per se, and will establish the due process and requirements to eventually replace design bases with a better, living and dynamic safety basis.

We have a good reason for a change; we need to have the technical basis to support that change. Therefore our first expectation for this meeting should be to identify, clarify, and, if possible, agree upon the current state of knowledge on the probability and consequences of various LOCAs.

As a second expectation, and as I alluded to above, we should also explore a related question (and answer it as best we can); that is, "If we change the Large Break LOCA, what should replace it?"

There is no doubt that, we will need to consider all of the design and operational implications of redefining the Large Break LOCA, and do it better than well. These include issues such as fuel and core design; containment design basis; ECCS design; RCS supports; emergency diesel generator start time; maximum hypothetical accident for dose assessment, emergency preparedness and control room habitability. These sets of issues need to be reduced for <u>holistic</u> system <u>and</u> probabilistic analysis.

Before discussing possible changes to the Large Break LOCA, let me first speak about the current NRC regulations in this area, that is, 10 CFR 50.46 and Appendix K, which establish the requirements for Emergency Core Cooling Systems. I will also mention some of the history of these requirements.

50.46 requires that "... ECCS cooling performance must be calculated ... for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." In this context, "loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system." In Appendix K, "ECCS Evaluation Models," the word "instantaneous" is added to the phrase "double-ended breaks" making the traditional maximum LOCA (but not necessarily the worst LOCA) the instantaneous double-ended break of the largest pipe in the reactor coolant system (usually the hot and occasionally the cold leg of the RCS).

50.46 analyses are all about consequences. And understanding consequences without understanding the associated probabilities is particularly meaningless for this case. We know that now very well, but the US Atomic Energy Commission (AEC) also knew that back in the 1970's. Qualitative judgements were made about the probability of a LOCA. That's why pipe failures are included in 50.46 but reactor vessel failures are not. The reactor vessel is the largest "pipe" in the RCS, but a judgement was made that vessel failures were so unlikely that protection was not necessary. That was a qualitative judgement about probability.

The approach to classifying events as "anticipated operational occurrences" and "postulated accident," is more than three decades old. It is a qualitative (or at best semi-quantitative) approach to event probability.

As operating experience and research data become available over time, those qualitative judgements are first validated and later replaced with quantitative information. It is a normal technical progression to go from qualitative judgements to quantitative estimates over time. That's expected progress.

In the December 28, 1973, "Opinion of the Commission," on the rulemaking hearing on 50.46, the Commission stated:

"In adopting this course [the 50.46 rule], we are not blinding ourselves to new knowledge acquired as a result of ongoing research. On the contrary, we believe that it is important that research programs - both analytic and experimental - continue, in order that we may increase knowledge relevant to ECCS performance ... As new knowledge is acquired, the Commission will analyze it, and at the appropriate time consider the possibility of amending the rule we announce today."

The Commission expected the regulatory requirements to change and progress along with the technology. However, they probably didn't think it would take 30 years!

In developing WASH-1400, the original "Reactor Safety Study," the AEC used the best information at that time to estimate the probability of various LOCA's -- including Large Break LOCAs and even vessel failures -- that was 1974.

Following the TMI accident, the NRC undertook a deep and serious look into its regulations and regulatory practices in the "NRC Special Inquiry" often referred to as the "Rogovin Report." In that report, a number of recommendations call for the increased use of risk analysis and risk insights. These recommendations include the following:

"The best way to improve the existing design review process is by relying in a major way upon quantitative risk analysis" and added,

"What we [the NRC Special Inquiry] are suggesting is that [the existing review process] be augmented and that quantitative methods be used as the best available guide to which accidents are the important ones, and which approaches are the best for reducing their probability and consequences." and again, it included a recommendation,

"We strongly urge that NRC begin the long and perhaps painful process of converting as much as is feasible of the present review process to a more accident-sequence-oriented approach."

I agree with their recommendations and with their predictions that the transition would be "long" and "painful." It should not have been that long and that painful, but it has been. The wheels of "nuclear" progress turn slow because predictability became equated to success. I do not disagree with that; I just disagree with the interpretation of predictability and success. Predictability must be rooted in today's know-how and success (in our case safety success) has to be meaningful for 2003 and beyond.

In 1995, eight years ago, the Commission issued a formal Commission Policy Statement supporting the increased use of PRA. We have made significant progress in the use of PRA since then, but we are far from done. That's our history and we cannot change it. But we have the opportunity to change the future, and I submit to you that we have the obligation to do so.

Now, in 2003, LOCA probabilities are routinely included in Probabilistic Risk Assessments (PRAs or PSAs). They are calculated every day and all around the world and are used in operational safety decisions . . . why not in the basic design requirements too? We have a sound understanding of the probabilities and consequences sufficient to progress to the next rational level of regulation to improve reactor safety.

The changes being considered by the NRC are headed in this direction. The situation is as follows:

The Commission has recently agreed to consider redefining the design basis large-break loss-of-coolant accident (LOCA) in view of the low risk associated with such events. The NRC staff was directed to provide the Commission a comprehensive "LOCA failure analysis and frequency estimation" that is realistically conservative and amenable to decision-making and to consider use of a 10-year period for the estimation of LOCA frequency distributions, with a rigorous re-estimation conducted every 10 years and a review for new types of failures every five years.

In that effort, the staff was directed to use Service-Data, Probabilistic Fracture Mechanics (PFM), and Expert Elicitation in a process that is risk-informed and consistent with the principles of RG 1.174. Where there is convergence, that is success, when there is divergence, there is work to be done.

The staff was also directed to credit leak-before-break where a licensee establishes a reliable and comprehensive means of detecting primary system leaks of the relevant size.

The staff was further directed to establish an appropriate risk "cutoff" for defining the maximum LOCA size and to require strict configuration controls during plant operation and a high quality PRA, including low power and shutdown operations.

These directives from the Commission to the NRC staff, highlight the two key technical issues involved with re-defining the LBLOCA; namely, LOCA frequencies and "PRA Quality". "PRA Quality" means having the appropriate scope level of detail reliability data and realism in accident progression and success criteria to support the regulatory decisions to be made. Since the risk assessment will play a significant role in this important change (i.e., re-defining the LBLOCA), we expect the PRA to be of high quality so that the results are both reliable and convincing. The PRA does not need to be perfect, but it does need to be "good enough". How good is "good enough" is an issue that we face for each risk-informed activity. And, as with previous activities, we will work with experts in the field to develop guidelines on "PRA quality" for this issue, and will probably use a NRC Regulatory Guide. The "PRA quality" issue will be difficult but it is well within our technical capabilities, and will be resolved in a prudent manner.

I am convinced that, as a matter of improving safety, the consideration of very low probability Large Break LOCAs should be addressed as severe accident scenarios, in the severe accident management program, rather than as the design basis accident. Effectively, the current LBLOCA would not be a design basis accident when utilizing a risk-informed approach. With an alternative definition of the LOCA, the really important, risk-significant, accident scenarios would remain within the design basis; in fact, their consideration would be enhanced by a new focus on their risk-importance.

These activities are in the formative stage; the commitment to go forward is fully formed and the NRC staff will develop proposed rule changes and associated guidance for public review and comment over the next several months. In addition, we expect one or more pilot applications which would request risk-informed changes to the Large-Break LOCA requirements through the NRC exemption process. This will provide a way of getting direct and practical experience with

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some of the important decisions to be made. We have found this approach very useful in the past.

I have no doubt that some, perhaps many, of the details of the rules and guidance will change, will mature and will become clearer as the staff discusses alternatives with interested parties . . . and that is good. Some new alternative approaches may even be developed. Information from this meeting may also influence the NRC's plans -- and that would be good too.

What I believe will not change is our commitment to improving safety and modernizing the treatment of the Large Break LOCA through the use of the best available information on the likelihood and potential consequences of these events and the best available approaches. And beyond the Large Break LOCA? 10 CFR 50 Appendix A and all it touches.

Realistically: there might be a tendency to let things be; to not challenge the status quo; to think that it is "ok". This would be wrong; technically and for long-term national energy policies.

Remember:

"Nothing else in the world . . . is so powerful as an idea whose time has come"

I look forward to working with you and to your contributions to make it happen.

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REMARKS

by

Chairman Nils J. Diaz
U.S. Nuclear Regulatory Commission

at the

Joint NRC/DHS State Security Outreach Workshop

June 17, 2003

The Department of Homeland Security Under Secretary McQueary, all the NRC partners in securing our homeland, especially our guests from the States, representatives of DHS and our sister Federal Agencies, members of the NRC staff: it is my pleasure and privilege to address you at this first, but surely not the last, Joint NRC/DHS State Security Outreach Workshop.

On behalf of the Commission, I thank Under Secretary McQueary and DHS for jointly sponsoring this very important workshop. In many ways, this meeting highlights a very important transition: from a period of policy-making and preparation that is taking place for America's homeland security to a more systematic, disciplined, coordinated implementation of security measures. This does not mean that, if called for, we all would not have jumped in and taken care of our people, with passion and thoughtfulness. We would not have worried about "who does it". We would have focused on "what" needs doing. We are now ready for a more cohesive and comprehensive program to defend our homeland from attacks. Your participation in this workshop is a tribute to how far we have come and your commitment to developing an integrated Federal, State, and local response.

Nuclear Regulatory Commission and Agreement State licensees were better prepared to respond to the spectrum of credible threats before 9/11 than most other facilities that are part of the civilian infrastructure and, continue to be so. Of course, we can always make further enhancements. Security against threats of diversion at our nuclear fuel facilities or sabotage at nuclear power plants are a 25year old business for the NRC and for our licensees. The awareness, resources and vigilance were there, but all went to a higher level when 9/11 showed the determination of enemies of the United States to attack our people and our way of life.

September 11, 2001 was a defining moment in American history and, in a very practical way, for all of us here today. The terrorist attacks focused public concern on the vulnerability of the national infrastructure to hostile action. For many in the public, the media, and the Congress, one immediate question especially concerning the NRC's licensees was: suppose the terrorists had chosen to attack a nuclear power plant? What then? There was nothing unreasonable about asking that question; on the contrary, it would have been unreasonable not to ask it, given the public prominence of anything related to nuclear power or radioactivity. The first answer, as the Commission has been stating, is that nuclear power plants, to a greater extent than any other kind of facility in our entire civilian infrastructure, are built to withstand powerful impacts. The second is that nuclear power plants have been required for a generation to assume that attack by well-armed terrorists is a real possibility, to be guarded against 24 hours a day, 365 days a year. Third, we have mitigation systems in place, including emergency planning and response, to minimize any impact on public health and safety. There is no doubt that today, both in our understanding and in our actions on how these three work together, there are significant improvements in the protection of public health and safety. They are not easily seen -- and sometimes that is intentional, for security doesn't always advertise -- but they are there. And they get better, everyday.

In the aftermath of the September 11 attacks, the Commission, unanimously, undertook a number of measures to improve security at nuclear power plants, to assess areas of possible vulnerability and to define corresponding mitigation strategies. The enhanced security construct we have established for the defense of nuclear power reactors includes three strongly interdependent elements, all of them directed to one fundamental goal: how to best protect our people, with the appropriate resources placed at the right places. These three elements are:

1) enhanced access controls, to prevent unauthorized entry of persons and materials to nuclear facilities;

2) enhanced work hour and training requirements for security personnel to increase their capability to detect and respond to threats; and

3) a revised Design Basis Threat that describes those adversary characteristics that are credible and reasonable for a private sector organization to protect against, based on the current threat, demonstrated terrorist attributes and intelligence as well as law enforcement information.

The aim of the security construct is clear enough: to deny access to potential wrongdoers and to ensure an ever-present security force that serves as a strong deterrent and as a tactically and weaponry-qualified defensive team that is capable of defending a facility with high assurance against a Design Basis Threat.

A key Commission statement, with strong implications for the NRC, DHS, and you, was made by the NRC regarding what was accomplished and the path forward: "The Commission believes that this DBT represents the largest reasonable threat against which a regulated private security force should be expected to defend under existing law." I repeat, "The Commission believes that this DBT represents the largest reasonable threat against which a regulated private secuity force should be expected to defend under existing law." This security framework includes both strengthened security by licensees and a clear role for the government in providing security beyond the licensee's capability while maintaining the ability of these industries and users to fulfill their intended functions. Let me repeat again, "providing security beyond the licensee's capability while maintaining the ability of these industries and users to fulfill their intended functions."

In addition to enhancements at the nuclear power facilities, NRC has also taken actions to enhance security at all our licensees -- from fuel cycle facilities to those licensees possessing discrete radioactive sources. For example, NRC and the Department of Energy are working to strengthen the U.S. regulatory infrastructure to increase the protection of high-risk radioactive sources which could be used to make a radiological dispersal device (RDD). The Commission recently approved the initial study of a joint NRC/DOE Working Group which provided action thresholds for radioactive materials of greatest concern. This report also addressed issues such as tracking and control of radioactive sources and recovery of unsecured radioactive material. We are also working closely with DHS in this area. For example, NRC staff is participating on the DHS Radiological Dispersal Device/Improvised Nuclear Device Working Group.

The Commission understands that it may not always be able to draw a bright line between security responsibilities of NRC-regulated entities and those of defense, security and law enforcement authorities. Responses may overlap for certain threats and coordination or integration of the responses of the various private and governmental organizations would be required. This is where the Commission, DHS, and other Federal Departments and agencies, and State and local authorities must work closely in developing integrated security contingency plans to complement licensee capabilities. The Commission believes that this integration is the responsibility of the Department of Homeland Security, and we have and stand ready to support DHS efforts in achieving integration. As we work to resolve integration issues at the Federal level, we also encourage efforts at the State and local level to develop the specific response protocols that will best serve the nation in enhancing homeland security.

At each step over the last 20 months, NRC has done what needed to be done to secure these facilities, but as we learn more, I am confident that the NRC, the Department of Homeland Security and you will do whatever it takes to protect the people of this country.

I would like to turn now towards another very important responsibility of the NRC. That is our responsibility to respond to radiological events of any kind, within our statutory responsibilities, and to coordinate Federal resources in support of State and local needs.

In May of 2003, the Department of Homeland Security issued an initial version of a "National Response Plan" (NRP) to address the management of domestic incidents, whether they are terrorist events or natural disasters. This Plan provides a framework from which Federal Agencies and Departments can begin to develop revisions to their existing incident response plans. This process will ultimately result in a comprehensive approach to domestic incident management. The Plan recognizes the vital roles that State and local authorities play in responding to all of the hazards that we face. We look forward to working with our Federal and State partners to implement this Plan, and encourage you to coordinate your activities closely with NRC licensees located within your states.

I have mentioned the NRC security construct and our commitment to incident response. I am pleased and proud to recognize that our new Office of Nuclear Security and Incident Response (NSIR) has done a tremendous job over its first year of existence in developing policies for the Commission, as well as in developing the requirements and oversight for other areas of homeland protection, and done so in a year of large uncertainties, as the nation is establishing its priorities. Our NSIR has done well for this country. I would like to recognize Roy Zimmerman, Director of NSIR, and all who labored long and well to increase our nuclear security and response capabilities. Roy, job well done!

With NSIR's transition to an implementation mode, the NRC is today taking another step to increase our attention to any cross-cutting issue of the agency's responsibilities that directly or indirectly affects security, incident response, emergency preparedness, vulnerability assessments and their mitigation strategies, and external integration of comprehensive strategies for these areas. At my request, and in consultation with the Commission, the Executive Director for Operations, Bill Travers, is establishing the position of Deputy Executive Director for Homeland Protection and Preparedness. The new Deputy, responding directly to the EDO, and both of them to me and the Commission, will have the authority to go across the agency lines of authority, to seek and resolve protection and preparedness issues, no matter who is doing it or where they reside in the agency. At NRC, the protection and preparedness of the homeland from nuclear events will respond to one senior manager, who will have the full support of every office in the agency. It is my pleasure to announce to you that Mr. William Kane has accepted the responsibility for Homeland Protection and Preparedness at the U.S. Nuclear Regulatory Commission.

You obviously note that I use the term homeland protection and preparedness rather than homeland security. There is a reason for the change. DHS has the overall responsibility for Homeland Security, and we are all responsible for different areas of concern. The NRC's main responsibility has always been the radiological <u>protection</u> of our people, as well as common defense and security. Our radiological protection requirements and expectations are based on multiple layers of defense, often referred to as defense-in-depth. The last layer of radiological protection is emergency preparedness, and there is no doubt that it has been, and is good. For example, the Commission believes that rapidly developing accident scenarios in nuclear power plants, whatever the initiator, are covered by the extensive emergency preparedness plans which are in place, and that the significant security improvements we have achieved, plant mitigation strategies, and emergency plans and off-site communications, are all contributors to robust and enhanced protective measures for the public. Yet, emergency preparedness must run deeper, covering the spectrum of radiological risks to our nation. There are concerns in this area that need to be addressed very clearly. I believe that mostly it is an issue of significantly enhancing the communication of what we do. But I am not discounting the probability that we can do better, especially in assuring that the communication and coordination links and resources are there if ever needed. I can assure you that the probability of a life-threatening radiation release from a nuclear plant is very small, yet we need to be prepared. These same concerns albeit to a much lesser degree, apply to events involving other licensed sources, the most noteworthy being an RDD event, yet, we need to be prepared.

I know this is an area of great concern to all the states, and particularly so for those high population areas where coordination of efforts is essential. As a key component of our focus on implementation, the NRC is going to pay increasingly close attention to homeland protection and preparedness.

I have provided my thoughts on some of the major issues facing the NRC, and focused them on the mutual dependence we have in ensuring the security of NRC-licensed facilities. I hope that I have provided some things for you to think about that will be useful to you in your discussions here over the next two days to build on what was good in the past, and move to what is better in the future.

The work of the NRC is, in microcosm, a reflection of the nation as a whole. There are competing interests and different points of view, strongly held, but what unites us is far greater than what divides us. All of us -- the NRC, its licensees, States and the public -- have a common interest in nuclear safety and security, and the well-being of our nation. All of us have different perspectives and insights to contribute; at its best, democracy permits a synthesis, in which we glean the best from divergent viewpoints and apply them to our common purposes. I look forward to the opportunity to join with you, our constituent stakeholders, toward a goal we all share: to benefit the American people.

Have a great workshop.

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REMARKS

BY

NILS J. DIAZ, CHAIRMAN
U.S. NUCLEAR REGULATORY COMMISSION AT THE
ALL EMPLOYEES MEETING

MORNING SESSION
10:30 A.M. WEDNESDAY, JUNE 11, 2003
PLAZA AREA, WHITE FLINT COMPLEX

Good morning, and welcome to the NRC's annual All Employees Meeting. Joining me this morning on the platform are my colleagues Edward McGaffigan and Jeffrey Merrifield. As you heard, Commissioner Greta Dicus is on official travel and will not be joining us today.

On behalf of the Commission, let me also welcome to this meeting those members of our staff who are located in the NRC Regional Offices, at the Technical Training Center in Chattanooga, and at remote sites throughout the country. You are an integral and highly valued part of this agency, notwithstanding your distance from your colleagues here at headquarters, and we look forward to your active participation in this session and the one that will follow this afternoon.

Today's meeting brings to mind three issues that are of special interest to me: communications, regulatory reforms, and homeland preparedness. I believe this is the 12th All Employees Meeting that the agency has held since the concept of an agency-wide meeting with the Commission intending to improve internal communications with the staff was first floated by one of my predecessors as Chairman, Dr. Ivan Selin. His idea was that such a meeting would be the most efficient way to explain to a concerned NRC staff the direction he planned to lead the agency and what his views on regulation and nuclear energy actually were. By most accounts, that first agency-wide meeting was, no pun intended, an electrifying experience as the NRC staff discovered that they could ask the Chairman and the Commission any question they wanted and receive an answer on the spot. Well, maybe not any question! Although the novelty of that first meeting has since worn off, the Commission has continued the practice because it has proven to be an important and effective tool for direct, two-way communication between the Commission and agency employees. My fellow Commissioners and I also hold "open doors" for your special concerns. One of the important lessons we have learned over the past decade is that effective internal agency communication is essential for improving our performance as a regulatory body. So I invite all of you to take advantage of the opportunity this meeting provides to express your interests or concerns in the form of questions, and we will do our best to respond to them, subject only to the usual limitations. Of course, I cannot promise you the novelty that marked the first All-Employees meeting -- my Commission colleagues and I have attended all but three or four of all the All-Employees Meetings ever held, so you know us better than any other Members of the Commission who have ever served in the entire history of the agency. Since most paths are two ways, we also know you well. That fact, I hope, will further encourage you to take an active part in these sessions -- we are, after all, "known materials" given our long, continuous association in the business of nuclear regulation and protection of the public health and safety.

The second theme that this meeting brings to mind is also a favorite of mine -- more safety-focused, less prescriptive, more risk-informed and performance-based regulation. As it so happens, Dr. Selin was also the Chairman who initiated the first concrete steps toward shifting the nuclear regulatory paradigm from the traditional prescriptive to a more risk-oriented approach to regulation by requesting the staff to prepare a report that explored the possibilities of expanded use of PRAs. As one of the agency's strong advocates of risk-informed regulation throughout my tenure as a Member of the Commission, I am pleased at the progress we have made, yet somewhat taken aback by the amount of time it has taken to get there. We have been steadily pursuing this very basic objective for more than ten years and still have a long way to go. This suggests to me, at least, that despite the progress we have clearly made, the nature of our business is complex and often driven by external events not subject to our control, and it is important to move forward steadily, but occasionally taking big steps. It is incumbent on us to be vigilant, persistent, patient, and committed in pursuit of our regulatory objectives as well as flexible and creative in responding to new challenges as they arise. I am both confident that we can do so and equally confident that we will have to do so.

One of the most important new challenges that the nation and the NRC face is public concern about homeland preparedness. By homeland preparedness, I mean the integrated coordination of the resources of the nation to prevent, respond to, or mitigate emergencies that would threaten the public health or safety. At the NRC, we usually refer to this issue as emergency preparedness for radiological protection, but present times are adding new dimensions not only to security, but also to emergency preparedness. The Commission believes that rapidly developing accident scenarios in nuclear power plants, whatever the initiator, are covered by the extensive emergency preparedness plans which are in place, and that the significant security improvements we have achieved, plant mitigation strategies, and emergency plans and off-site communications, are all contributors to robust and enhanced protective measures for the public. Yet, emergency preparedness must run deeper, covering the spectrum of radiological risks to our nation. Homeland preparedness is a serious concern for the citizens of the United States; it is an issue to which we are paying increasingly close attention.

Finally, let me return to where I began my remarks -- with the importance of communications. The challenges posed by regulatory reform and homeland preparedness have an important connection to the adequacy of our external communications efforts -- the need to explain clearly and accurately what we are doing and why. Improved communications, in my view, rests on two basic supports -- the need to communicate in clear, factual language without minimizing or exaggerating issues, and the quality of our actions. The actions of strong and active regulators carry a particularly strong message and can significantly enhance public confidence in the NRC.

Of course, a prerequisite to improving our external communications is the ability to communicate effectively within the agency. We all have a role to play in this effort, including the Commission. As I stated earlier, this All-Employees Meeting is part of the overall effort to improve our internal communications. In keeping with that objective, we would now like to turn the meeting over to you so that you may ask the questions you want to ask.

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NEI EMERGENCY PREPAREDNESS AND COMMUNICATIONS INFORMATION FORUM

Keynote Panel Remarks

by

The Honorable Greta Joy Dicus
Commissioner
U.S. Nuclear Regulatory Commission

June 9, 2003
Key Biscayne, Florida

Good morning everyone. I am delighted to be here today and delighted to join my distinguished co-panelists Angie and Art.

Many of you may know that I am in the last month of what has been a seven year run as a Commissioner. It has been a wonderful seven years and many of you, here today, have directly and indirectly contributed to my success and, most importantly, contributed to the success of the industry and the Nuclear Regulatory Commission in protecting public health and safety. So, thank you.

Many of you may also know that my background is in radiation biology and before joining the NRC, I was Director of Radiation Control and Emergency Management at the Arkansas Department of Health for nine years and conducted research in radiation health effects at Harvard Medical School, Rice University, and the University of Texas Southwestern Medical School.

I have been on both sides of the fence and can recall, as Arkansas Director of Emergency Management thinking how intractable the NRC was. Now as a NRC Commissioner, I see how immovable some States can be. I concluded that instead of focusing on solving each others problems, it might just be easier for all of us to focus on EPA. Is there anyone here from EPA? Just kidding !!

As I reflect on where we were in emergency planning and communication and compare that to where we are today, there have been some significant changes. Fast-breakers, KI, and security to name a few. We are better today, no question about that.

But the world today is more complex. Technology has advanced, our understanding of risk and vulnerabilities has improved, the public has become more involved, and the push for regulatory efficiency has become a theme. We are now dealing with emergency planning in a backdrop of potential malicious acts and uncertainty. All significant challenges individually, and together, these challenges present a daunting task.

To be sure, however, the focus of emergency planning has not changed. The focus of our emergency planning remains to protect public health and safety. It can be regarded as the last layer in our defense-in-depth strategy and if called upon must not fail. I know you have and will continue to meet the challenges before you simply because, as some have said, "failure is not an option".

I would like to share with you some impressions about elements of effective emergency planning and preparedness. Three elements that I would like to focus on are relationship and communication, management support, and the need to always look forward.

<u>**Relationship and Communication**</u>

The panel discussions over the next few days , I think, will be particularly helpful in framing some of the more important contemporary challenges before us. Few could have envisioned the sweeping impact that the events of September 11th had on security, emergency preparedness, and plant operations.

Even fewer could have been prepared for the unfolding and still developing complex relationships that are essential to functioning effectively in a post- September 11th environment. The key to building and maintaining these strong relationships is to continue to ensure that they are built on truth, trust, communication and mutual respect.

The post-September 11th security and safeguards environment has strained many relationships. NRC has strained relationships with many of its stakeholders, in part, because, the nature of dealing with sensitive security-related information often necessitates implementing NRC processes outside of public purview. When this happens, communication wanes and trust is more difficult to maintain.

Many of you may have new relationships with the community, law enforcement, and other government agencies as a result of the response to the events of September 11th. The challenge is to forge these new relationships in a meaningful manner while building on the foundation of previous relationships. As with any life-changing event, there is potential for profound positive outcomes. I believe that all parties will emerge from this with a better understanding of their roles and responsibilities and, ultimately, different and better relationships.

Just last year, I attended the security-related table top exercise at Fort Calhoun Nuclear Power Station just outside of Omaha, Nebraska. One afternoon session was devoted to an interactive forum with local law enforcement, armed forces representatives, federal law enforcement agencies, and emergency planning folks from the site and from Nebraska and Iowa.

It was truly an epiphany to see how far we have come in a short time since September 11th, yet how far we still need to go. It was important for all to understand how resources may be diverted from assigned emergency planning activities if a security-related event were to occur. It was critical to re-validate communication links, to ensure communication protocols are in place, and to understand decision-making roles and responsibilities.

My experience at Ft Calhoun further reinforced my long held beliefs, which took root when I was Director of Emergency Management in Arkansas, that communications are key during an emergency. Both communication to direct and mitigate the event and communication to members of the public must be clear and relevant. Communication is almost always a challenge. If the challenge is not in establishing communications, it is communicating clearly and ensuring the fidelity of the message as it passes through layers and branches of the organization and to members of the public.

I am sure I am preaching to the choir when it comes to the importance of communications. But in a post September 11th world, it cannot be overemphasized. Communication with all stakeholders is paramount.

<u>Management and Government Support</u>

Emergency planning will not be successful unless there is deep-rooted management support and support of local and State governments. Emergency planning is a team effort. We can learn a lot just by examining some contemporary issues around some of the nations nuclear facilities.

I understand that Art will talk about the malicious turkey at the Seabrook facility. We can chuckle about it today, it was a not so serious event but one that seriously challenged our coordination and communication. There are many lessons learned that I hope will improve future emergency planning, response and communication. The NRC is still considering hiring the turkey to assist in our security and emergency preparedness drills ! !

Another example of how crucial it is to have management and government support involves the circumstances surrounding emergency planning at the Indian Point plants. Around Indian Point, some local governments did not provide the necessary information for the Federal Emergency Management Agency to complete its review processes. We all know that licensees get "dinged" when the State and local governments do not perform well during an emergency exercise. But it is rare when some segments of government seemingly refuse to participate. In this case, unfortunately, emergency planning is gridlocked at the intersection of safety and security and politics is the erratic traffic light.

Despite the new challenges, there are numerous bright spots in emergency planning and communication post September 11th. I mentioned my observation of a security related table-top exercise and forum at Ft. Calhoun. Within the last year, I also visited the Cooper, Farley, and South Texas Project nuclear power plants. Each one of these plants, as I am confident the vast majority of plants, have strong working relationships with local law enforcement, with the local community, and with federal partners. When I visited South Texas for example, I met with the local sheriff, the local County judge, and a representative from the Houston FBI office. An impressive network of cooperation that was not strained by the events of September 11th, but instead was strengthened.

I challenge each of you to evaluate these support and communication networks. Is there a larger role for you to play in stimulating two-way communication among stakeholders, for example? Are you doing enough outreach to the community?

Looking Forward

I have mentioned some significant changes in the emergency planning arena and how, recently, many of these changes have been related to the emerging interrelationship between security and emergency planning. But there are many other challenges ahead that have taken a back seat because of the challenges of the last 15 months.

For example, the industry and NRC are beginning to exercise the early site permit process -- a key element of licensing a new nuclear power plant. All three sites currently under consideration are associated with existing nuclear power plant sites. One of the underlying thoughts behind this rationale is that it would be much easier to site a new plant where one is already sited, in part, because an emergency planning infrastructure is already approved and in-place.

Siting a new plant at an existing NPP site is a deliberate process. It does not, however, exercise all aspects of the early site permit process that might be involved in siting a nuclear power plant on a new site.

Siting a new plant on a new site is a significant challenge and one that will rightly need to build on successes. Once we have learned lessons from the initial early site permit reviews, we must prepare for the more complex emergency planning and communications challenges associated with a new site.

One aspect of emergency planning for new reactors that also may apply to currently operating reactors is the concept of a reduced source term. Some of the newer designs have such low source terms that policy questions arises about whether we can have smaller emergency planning zones or even whether we need to have a containment. I have always believed that our rules and regulations must be based on good science. In this case, however, I am not sure science will prevail quickly because I am uncertain whether the American public can accept nuclear power plants without containment or without emergency plans beyond the site boundary.

Of a more immediate nature are some of the challenges associated with changes in roles and responsibilities that have resulted from the formation of the new Department of Homeland Security. We recently completed a national exercise, TOPOFF2, and there are emerging lessons learned that may have some impact on emergency planning and communications.

We are currently reviewing Revision 11 of the draft Initial National Response Plan and are considering additional requirements for updating NRC's emergency plans and federal interagency plans. These updates will be submitted to the Secretary of Homeland Security. The extent of these changes is still unknown. It is still too early to tell.

It also seems as if we deal weekly with new legislative proposals. Some of these proposals are rooted in response to the events of September 11th, some more explicitly linked to issues at a specific site, and others more closely tied to defining roles and responsibilities between the newly formed Department of Homeland Security and other federal and state agencies. Regardless of the motivation, it is essential that everyone understand the dynamic nature of the emergency planning-related environment. My crystal ball is no better than your crystal ball and it is critical that you stay engaged and provide input into the processes by which some of these changes may occur.

Conclusion

Thank you for inviting me to this important forum. I wish you all the best in future endeavors. I look forward to the interesting agenda before us and look forward to continuing the dialogue with you throughout the forum.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555

June 9, 2003

NRC BULLETIN 2003-01: POTENTIAL IMPACT OF DEBRIS BLOCKAGE ON EMERGENCY SUMP RECIRCULATION AT PRESSURIZED-WATER REACTORS

Addressees

All holders of operating licenses for pressurized-water nuclear power reactors, except those who have ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to:

(1) Inform addressees of the results of NRC-sponsored research identifying the potential susceptibility of pressurized-water reactor (PWR) recirculation sump screens to debris blockage in the event of a high-energy line break (HELB) requiring recirculation operation of the emergency core cooling system (ECCS) or containment spray system (CSS).

(2) Inform addressees of the potential for additional adverse effects due to debris blockage of flowpaths necessary for ECCS and CSS recirculation and containment drainage.

(3) Request that, in light of these potentially adverse effects, addressees confirm their compliance with 10 CFR 50.46(b)(5) and other existing applicable regulatory requirements, or describe any compensatory measures implemented to reduce the potential risk due to post-accident debris blockage as evaluations to determine compliance proceed.

(4) Require addressees to provide the NRC a written response in accordance with 10 CFR 50.54(f).

Background

In 1979, as a result of evolving staff concerns related to the adequacy of PWR recirculation sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." To support the resolution of USI A-43, the NRC undertook an extensive research program, the technical findings of which are summarized in NUREG-0897,

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"Containment Emergency Sump Performance," dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," dated December 3, 1985. Although the staff's regulatory analysis concerning USI A-43 did not support imposing new sump performance requirements upon PWRs or boiling-water reactors (BWRs) that were then licensed or under construction, the staff's technical findings identified certain conditions that would inherently lead to these plants' design assumption of 50 percent sump blockage being nonconservative. Therefore, in GL 85-22 the NRC staff recommended that all reactor licensees replace the 50 percent blockage assumption with a comprehensive mechanistic assessment of plant-specific debris blockage potential for future modifications related to sump performance, such as thermal insulation changeouts. The staff also updated the NRC's regulatory guidance, including Section 6.2.2 of the Standard Review Plan (NUREG-0800) and Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," to reflect the USI A-43 technical findings documented in NUREG-0897.

Following the resolution of USI A-43 in 1985, several events challenged the staff's conclusion that no new requirements were necessary to prevent the clogging of ECCS strainers at operating BWRs:

- On July 28, 1992, at Barsebäck Unit 2, a Swedish BWR, the spurious opening of a pilot-operated relief valve led to the plugging of two containment vessel spray system suction strainers with mineral wool and required operators to shut down the spray pumps and backflush the strainers.

- In 1993, at Perry Unit 1, ECCS strainers twice became plugged with debris. On January 16, ECCS strainers were plugged with suppression pool particulate matter, and on April 14, an ECCS strainer was plugged with glass fiber from ventilation filters that had fallen into the suppression pool. On both occasions, the affected ECCS strainers were deformed by excessive differential pressure created by the debris plugging.

- On September 11, 1995, at Limerick Unit 1, following a manual scram due to a stuck-open safety/relief valve, operators observed fluctuating flow and pump motor current on the "A" loop of suppression pool cooling. The licensee later attributed these indications to a thin mat of fiber and sludge which had accumulated on the suction strainer.

In response to these ECCS suction strainer plugging events, the NRC issued several generic communications, including Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," dated February 18, 1994; Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," dated October 17, 1995; and Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," dated May 6, 1996. These bulletins requested that BWR licensees implement appropriate procedural measures, maintenance practices, and plant modifications to minimize the potential for the clogging of ECCS suction strainers by debris accumulation following a loss-of-coolant accident (LOCA). The NRC staff has concluded that all BWR licensees have adequately addressed these bulletins.

However, the findings from research to resolve the BWR strainer plugging issue in the late 1990s raised questions concerning the adequacy of PWR sump designs by confirming what the aforementioned BWR strainer plugging events had earlier indicated: (1) that the amount of debris generated by a HELB could be greater than estimated by the USI A-43 research program, (2) that the debris could be finer (and, thus, more easily transportable), and (3) that certain combinations of debris (e.g., fibrous material plus particulate material) could result in a substantially greater head loss than an equivalent amount of either type of debris alone. These BWR research findings, which may also affect the performance of PWR sumps, prompted the NRC to open Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The objective of GSI-191 is to ensure that post-accident debris blockage would not impede or prevent the operation of the ECCS and CSS in the recirculation mode at PWRs in the event of a LOCA or other HELB accidents for which sump recirculation is required.

Discussion

In the event of a HELB within the containment of a PWR, energetic pressure waves and fluid jets would impinge upon materials in the vicinity of the break, such as thermal insulation, coatings, and concrete, causing damage and generating debris. Debris could also be generated through secondary mechanisms, such as severe post-accident temperature and humidity conditions, flooding of the lower containment, and the impact of containment spray droplets. Through transport methods such as entrainment in the steam/water flows issuing from the break and in containment spray washdown, a fraction of the generated debris and foreign material in the containment would be transported to the pool of water formed on the containment floor. If the ECCS or CSS pumps subsequently took suction from the recirculation sump, the debris suspended in the containment pool would begin to accumulate on the sump screen. The accumulation of this suspended debris on the sump screen could create a roughly uniform mat over the entire screen surface, referred to as a debris bed, which would tend to increase the head loss across the screen through a filtering action. If a sufficient amount of debris accumulated, the debris bed would reach a critical thickness at which the head loss across it would exceed the net positive suction head (NPSH) margin required to ensure the successful operation of the ECCS and CSS pumps in the recirculation mode. A loss of NPSH margin for the ECCS or CSS pumps as a result of the accumulation of debris on the recirculation sump screen, referred to as sump clogging, could result in degraded pump performance and eventual pump failure.

To assess the likelihood of the ECCS and CSS pumps at domestic PWRs experiencing a debris-induced loss of NPSH margin during sump recirculation, the NRC sponsored a GSI-191 research program, which culminated in a parametric study. The parametric study mechanistically treated phenomena associated with debris blockage using analytical models of domestic PWRs that were generated with a combination of generic and plant-specific data. As documented in Volume 1 of NUREG/CR-6762, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," dated August 2002, the GSI-191 parametric study concludes that recirculation sump clogging is a credible concern for the population of domestic PWRs. However, as a result of limitations with respect to plant-specific data and other modeling uncertainties, the parametric study does not definitively identify whether or not particular PWR plants are vulnerable to sump clogging when phenomena associated with debris blockage are modeled mechanistically.

The methodology employed by the GSI-191 parametric study is based upon the substantial body of test data and analysis documented in technical reports generated during the NRC's GSI-191 research program and earlier technical reports generated by the NRC and the industry during the resolution of the BWR strainer clogging issue and USI A-43. The following pertinent technical reports, which cover debris generation, transport, accumulation, and head loss, are incorporated by reference into the GSI-191 parametric study:

- NUREG/CR-6770, "GSI-191: Thermal-Hydraulic Response of PWR Reactor Coolant System and Containments to Selected Accident Sequences," dated August 2002.

- NUREG/CR-6762, Vol. 3, "GSI-191 Technical Assessment: Development of Debris Generation Quantities in Support of the Parametric Evaluation," dated August 2002.

- NUREG/CR-6762, Vol. 4, "GSI-191 Technical Assessment: Development of Debris Transport Fractions in Support of the Parametric Evaluation," dated August 2002.

- NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," dated October 1995.

In addition to demonstrating the potential for debris to clog containment recirculation sumps, operational experience and the NRC's technical assessment of GSI-191 have also identified three integrally related modes by which post-accident debris blockage could adversely affect the sump screen's design function of intercepting debris that could impede or prevent the operation of the ECCS and CSS in the recirculation mode.

First, as a result of the 50 percent blockage assumption, PWR sump screens were typically designed assuming that relatively small structural loadings would result from the differential pressure associated with debris blockage. Consequently, PWR sump screens may not be capable of accommodating the substantial structural loadings that would occur due to mechanistically determined debris beds that may cover essentially the entire screen surface. Inadequate structural reinforcement of a sump screen may result in its deformation, damage, or failure, which could allow large quantities of debris to be ingested into the ECCS and CSS piping, pumps, and other components, potentially leading to their clogging and failure. The ECCS strainer plugging and deformation events that occurred at Perry Unit 1— further described in Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993, and Licensee Event Report (LER) 50-440/93-011, "Excessive Strainer Differential Pressure Across the RHR [Residual Heat Removal] Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation," submitted May 19, 1993—demonstrate the credibility of this concern for screens and strainers that have not been designed with adequate reinforcement.

Second, in some PWR containments, the flowpaths by which containment spray or break flows return to the recirculation sump may include "chokepoints," where the flowpath becomes so constricted that it could become blocked with debris following a HELB. For example, chokepoints may include drains for pools, cavities, or isolated containment compartments, and other constricted drainage paths between physically separated containment elevations. As a result of debris blockage at certain chokepoints, substantial amounts of water required for adequate recirculation could be held up or diverted into containment volumes that do not drain to the recirculation sump. The holdup or diversion of water assumed to be available to support sump recirculation could result in an available NPSH for ECCS and CSS pumps that is lower than the analyzed value, thereby reducing assurance that recirculation would function successfully. A reduced available NPSH directly impacts sump screen design because the NPSH margin of the ECCS and CSS pumps must be conservatively calculated to determine correctly the required surface area of passive sump screens when mechanistically determined debris loadings are considered. The NRC's GSI-191 research identified the holdup or diversion of recirculation sump inventory as an important and potentially credible concern, and a number of LERs associated with this concern have also been generated, which further confirms both its credibility and potential significance. These LERs include:

- LER 50-369/90-012, "Loose Material Was Located in Upper Containment During Unit Operation Because of an Inappropriate Action," McGuire Unit 1, submitted August 30, 1990.
- LER 50-266/97-006, "Potential Refueling Cavity Drain Failure Could Affect Accident Mitigation," Point Beach Unit 1, submitted February 19, 1997.
- LER 50-455/97-001, "Unit 2 Containment Drain System Clogged Due to Debris," Byron Unit 2, submitted April 17, 1997.
- LER 50-269/97-010, "Inadequate Analysis of ECCS Sump Inventory Due to Inadequate Design Analysis," Oconee Unit 1, submitted January 8, 1998.
- LER 50-315/98-017, "Debris Recovered from Ice Condenser Represents Unanalyzed Condition," D.C. Cook Unit 1, submitted July 1, 1998.

Third, debris blockage at flow restrictions within the ECCS and CSS recirculation flowpaths downstream of the sump screen is of potential concern for PWRs. For this mode of debris blockage to occur, pieces of debris would need to have spatial dimensions that would allow them to pass through the sump screen's intended openings, or through screen defects such as gaps or breaches, and then become lodged at downstream flow restrictions such as pump internals, high-pressure safety injection (HPSI) throttle valves, fuel assembly inlet debris screens, or containment spray nozzles. In particular, conditions conducive to downstream debris blockage may be present at PWRs with adverse screen defects, and at PWRs where the maximum dimension of the sump screen's intended openings (e.g., the diagonal dimension of a rectangular mesh) is not the most restrictive point in the ECCS and CSS recirculation flowpaths. Downstream debris blockage at restrictions in the ECCS flowpath could impede or prevent the recirculation of coolant to the reactor core, thereby leading to inadequate core cooling. Similarly, downstream debris blockage at restrictions in the CSS flowpath could impede or prevent CSS recirculation, thereby leading to inadequate containment heat removal. Numerous operational events concerning the discovery of inadequate sump screen configurations that could have led to downstream blockage are cited in Attachment 2 to GL 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," dated July 14, 1998.

Three emergent items have increased the urgency of the NRC staff's efforts to ensure that PWR licensees are aware of and appropriately responding to the above concerns regarding the potential for debris blockage to impede or prevent the operation of the ECCS and CSS in the recirculation mode: (1) an LER submitted by the licensee for Davis-Besse Unit 1 that declared the recirculation sump inoperable, (2) a subsequent LER submitted by the Davis-Besse licensee that declared the high-pressure injection (HPI) pumps inoperable, and (3) an NRC-sponsored risk study concerning operator actions to mitigate sump clogging.

On December 11, 2002, the licensee for Davis-Besse Unit 1 submitted LER 50-346/02-005-01, "Potential Clogging of the Emergency Sump Due to Debris in Containment." In this LER, the licensee stated that the recirculation sump had been declared inoperable as a result of the potential for sump clogging due to unqualified coatings and other potential sources of post-accident debris (e.g., fibrous insulation and improperly applied qualified coatings) and the potential for downstream debris blockage to occur due to a 6-inch by 3/4-inch gap discovered in the screen. The information provided in this LER, and in a public meeting with the licensee on November 26, 2002, also showed that key information documented in NUREG/CR-6762, Vol. 2, "GSI-191 Technical Assessment: Summary and Analysis of U.S. Pressurized Water Reactor Industry Survey Responses and Responses to GL 97-04," dated August 2002, and other assumptions used in the parametric study to model Davis-Besse Unit 1 were not conservative with respect to design basis assumptions regarding sump screen surface area, minimum containment pool depth at switchover to recirculation, and particulate debris generation.

On May 5, 2003, the Davis-Besse licensee submitted LER 50-346/03-002-00, which stated that the HPI pumps had been declared inoperable as a result of the potential for debris to damage the pump internals during the recirculation phase of certain postulated LOCAs when the HPI pumps are required to take suction from the containment recirculation sump. This LER stated that, when an HPI pump takes suction from the recirculation sump, small particles of debris may result in localized erosion of the mating surfaces around rotating parts, and that the flow of sump water that lubricates the hydrostatic bearing (which is drawn from the volute of the HPI pump) could be blocked by entrained debris, resulting in bearing damage.

In February 2003, Los Alamos National Laboratory published the NRC-sponsored technical report LA-UR-02-7562, "The Impact of Recovery From Debris-Induced Loss of ECCS Recirculation on PWR Core Damage Frequency." The report analyzes the potential risk benefit of operator actions to recover from sump clogging events using a generic probabilistic model to demonstrate that the potential increase in risk due to sump clogging could be reduced by approximately one order of magnitude if PWR licensees have appropriate mitigative measures in place.

In response to these emergent items associated with the potential post-accident debris blockage concerns identified in this bulletin, the NRC is requesting that individual PWR licensees submit information on an expedited basis to document that they have either (1) analyzed the ECCS and CSS recirculation functions with respect to the identified post-accident debris blockage effects, taking into account the recent research findings described in the Discussion section, and determined that compliance exists with all applicable regulatory requirements, or (2) implemented appropriate interim compensatory measures to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions while evaluations to determine compliance proceed. The NRC staff recognizes that it may be necessary for addressees to undertake complex evaluations to determine whether regulatory compliance exists in light of the concerns identified in this bulletin, and the staff is preparing a generic letter that would request this information.

To assist in determining whether the ECCS and CSS recirculation functions are in compliance with existing applicable regulatory requirements, addressees may use the guidance in Draft Regulatory Guide 1107 (DG-1107), "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated February 2003. The NRC has also published a technical report entitled NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," dated February 2003, which is designed to serve as a reference for plant-specific analyses with regard to whether a sump would perform its function without preventing the operation of the ECCS and CSS pumps. In addition, the NRC staff supports the development of generic industry guidance and the coordination of addressees' responses to this bulletin as a means of increasing efficiency and streamlining the regulatory verification process. Individual addressees may also develop alternative approaches to those mentioned in this paragraph for determining the status of their regulatory compliance; however, additional staff review may be required to assess the adequacy of alternative approaches.

Conditions at specific PWRs are expected to vary with respect to susceptibility to post-accident debris blockage and various options may be available to addressees for preventing or mitigating the effects of debris blockage. For these reasons, addressees that are unable to confirm compliance with all existing regulatory requirements within 60 days in light of the potential debris blockage effects identified in this bulletin may consider a range of possible interim compensatory measures and may elect to implement those which they deem appropriate, based upon the specific conditions associated with their plants. As stated above, the risk benefit of certain interim compensatory measures is demonstrated by the NRC-sponsored technical report LA-UR-02-7562. Possible interim compensatory measures may include, but are not limited to, the following:

- operator training on indications of and responses to sump clogging
- procedural modifications, if appropriate, that would delay the switchover to containment sump recirculation (e.g., shutting down redundant pumps that are not necessary to provide required flows to cool the containment and reactor core, and operating the CSS intermittently)
- ensuring that alternative water sources are available to refill the RWST or to otherwise provide inventory to inject into the reactor core and spray into the containment atmosphere
- more aggressive containment cleaning and increased foreign material controls
- ensuring containment drainage paths are unblocked
- ensuring sump screens are free of adverse gaps and breaches

In addition to the measures listed above, addressees may also consider implementing unique or plant-specific compensatory measures, as applicable. Commensurate with the potential risk-significance of post-accident debris blockage effects, addressees electing to implement interim compensatory measures in response to this bulletin should ensure the interim measures are implemented as soon as practical. The NRC staff recognizes that the implementation of certain compensatory measures involving containment entry may not be feasible until the next outage.

Approximately two weeks after the issuance of this bulletin, the NRC plans to hold a public meeting to further clarify the intent of the bulletin and respond to any questions from addressees regarding the bulletin. The NRC plans to publish the notice for this public meeting promptly after the bulletin is issued.

Applicable Regulatory Requirements

NRC regulations in Title 10 of the *Code of Federal Regulations*, Section 50.46, (10 CFR 50.46) require that the ECCS satisfy five criteria, one of which is to provide the capability for long-term cooling of the reactor core. As set forth in 10 CFR 50.46(b)(5), the ECCS must have the capability to remove decay heat so that the core temperature is maintained at an acceptably low value for the extended period of time required by the long-lived radioactivity remaining in the core. For PWRs licensed to the General Design Criteria (GDCs) in Appendix A to 10 CFR Part 50, GDC 35 specifies additional ECCS requirements.

Similarly, for PWRs licensed to the GDCs in Appendix A to 10 CFR Part 50, GDC 38 provides requirements for containment heat removal systems, and GDC 41 provides requirements for containment atmosphere cleanup. Many PWR licensees credit a CSS, at least in part, with performing the safety functions to satisfy these requirements, and PWRs that are not licensed to the GDCs may credit a CSS to satisfy similar plant-specific licensing basis requirements. In addition, PWR licensees may credit a CSS with reducing the accident source term to meet the limits of 10 CFR Part 100 or 10 CFR 50.67.

Technical specifications pertain to the ECCS and CSS insofar as they require the operability of these systems for the mitigation of certain design basis accidents. Other plant-specific licensing commitments concerning the ECCS and CSS are also documented in the Final Safety Analysis Report (FSAR).

Applicable Regulatory Guidance

Draft Regulatory Guide 1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," dated February 2003.

Requested Information

All addressees are requested to provide a response within 60 days of the date of this bulletin that contains either the information requested in Option 1 or Option 2:

Option 1: State that the ECCS and CSS recirculation functions have been analyzed with respect to the potentially adverse post-accident debris blockage effects identified in this bulletin, taking into account the recent research findings described in the Discussion section, and are in compliance with all existing applicable regulatory requirements.

Option 2: Describe any interim compensatory measures that have been implemented or that will be implemented to reduce the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions until an evaluation to determine compliance is complete. If any of the interim compensatory measures listed in the Discussion section will not be implemented, provide a justification. Additionally, for any planned interim measures that will not be in place prior to your response to this bulletin, submit an implementation schedule and provide the basis for concluding that their implementation is not practical until a later date.

Required Response

In accordance with 10 CFR 50.54(f), the NRC requires each addressee to respond as described above. The NRC needs this information to verify addressees' compliance with NRC regulations and to ensure that any interim risks associated with post-accident debris blockage are minimized while evaluations to determine compliance proceed.

Within 60 days of the date of this bulletin, each addressee is required to submit a written response that includes the information requested above in the Requested Information section. Addressees who choose not to submit the requested information must describe in their responses any alternative course of action that they propose to take, including the basis for the acceptability of the proposed alternative course of action.

The required written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). A copy of the response should be sent to the appropriate regional administrator.

The NRC staff will review the responses to this bulletin and, if concerns are identified, will notify affected addressees. The staff may also conduct inspections to determine addressees' effectiveness in addressing this bulletin.

Reasons for Information Request

As discussed above, recent research and analysis suggests that (1) most PWR licensees' current safety analyses do not adequately address the potential for the failure of the ECCS and CSS recirculation functions as a result of debris blockage, and (2) the ECCS and CSS recirculation functions at a significant number of operating PWRs could become degraded as a result of the potential effects of debris blockage identified in this bulletin. An ECCS that is incapable of providing long-term reactor core cooling through recirculation operation would be in violation of 10 CFR 50.46. A CSS that is incapable of functioning in the recirculation mode may not comply with GDCs 38 and 41, or other plant-specific licensing requirements or safety analyses. Furthermore, to address the risk which may be associated with potentially degraded or nonconforming ECCS and CSS recirculation functions, addressees that are unable to confirm regulatory compliance may find it appropriate to implement compensatory measures until a determination can be made. Therefore, the NRC needs the information requested in this bulletin to assess plant-specific compliance with NRC regulations and to ensure the safe operation of PWR facilities as addressees resolve the concerns identified in this bulletin.

Related Generic Communications

- Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.
- Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in the Suppression Pool Cooling Mode," October 17, 1995.
- Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," May 11, 1993.
- Bulletin 93-02, Supplement 1, "Debris Plugging of Emergency Core Cooling Suction Strainers," February 18, 1994.
- Generic Letter 98-04, "Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," July 14, 1998.
- Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," October 7, 1997.
- Generic Letter 85-22, "Potential For Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," December 3, 1985.
- Information Notice 97-13, "Deficient Conditions Associated With Protective Coatings at Nuclear Power Plants," March 24, 1997.
- Information Notice 96-59, "Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris," October 30, 1996.
- Information Notice 96-55, "Inadequate Net Positive Suction Head of Emergency Core Cooling and Containment Heat Removal Pumps Under Design Basis Accident Conditions," October 22, 1996.
- Information Notice 96-27, "Potential Clogging of High Pressure Safety Injection Throttle Valves During Recirculation," May 1, 1996.
- Information Notice 96-10, "Potential Blockage by Debris of Safety System Piping Which Is Not Used During Normal Operation or Tested During Surveillances," February 13, 1996.

- Information Notice 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," October 4, 1995.
- Information Notice 95-47, Revision 1, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," November 30, 1995.
- Information Notice 95-06, "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," January 25, 1995.
- Information Notice 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994.
- Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993.
- Information Notice 93-34, Supplement 1, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," May 6, 1993.
- Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992.
- Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," September 30, 1992.
- Information Notice 89-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," December 1, 1989.
- Information Notice 89-79, Supplement 1, "Degraded Coatings and Corrosion of Steel Containment Vessels," June 29, 1990.
- Information Notice 89-77, "Debris in Containment Emergency Sumps and Incorrect Screen Configurations," November 21, 1989.
- Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," May 19, 1988.

Backfit Discussion

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this bulletin transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements. Specifically, the requested information will enable the NRC staff to determine whether PWR licensees are in compliance with plant-specific regulatory requirements concerning the ECCS and CSS recirculation functions and ensure the safe operation of their facilities as they resolve the concerns identified in this bulletin. No backfit is either intended or approved by the issuance of this bulletin and, therefore, the staff has not provided a backfit analysis.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this bulletin is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

Federal Register Notification

A notice of opportunity for public comment on this bulletin was not published in the *Federal Register* because the NRC is requesting information from PWR licensees on an expedited basis to assess compliance with existing applicable regulatory requirements and the necessity for interim compensatory measures. As the resolution of this matter progresses, the opportunity for public involvement will be provided. Nevertheless, comments on the information requested and the technical issues addressed by this bulletin may be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC, 20555-0001.

Paperwork Reduction Act Statement

This bulletin contains an information collection that is subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). This information collection was approved by the Office of Management and Budget (OMB), clearance number 3150-0012, which expires on July 31, 2003. The burden to the public for this mandatory information collection is estimated to average 150 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or any other aspect of this information collection, including suggestions for reducing the burden, to the Records Management Branch, Mail Stop T-6 E6, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0012), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contacts or lead project manager listed below.

/RA/
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OMB Control No.: 3150-0011

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

June 12, 2003

NRC GENERIC LETTER 2003-01: CONTROL ROOM HABITABILITY

Addressees

All holders of operating licenses for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel and more than 1 year has elapsed since fuel was irradiated in the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to:

(1) alert addressees to findings at U.S. power reactor facilities suggesting that the control room licensing and design bases, and applicable regulatory requirements (see section below) may not be met, and that existing technical specification surveillance requirements (SRs) may not be adequate,

(2) emphasize the importance of reliable, comprehensive surveillance testing to verify control room habitability,

(3) request addressees to submit information that demonstrates that the control room at each of their respective facilities complies with the current licensing and design bases, and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance, and

(4) collect the requested information to determine if additional regulatory action is required.

Background

The control room is the plant area, defined in the facility licensing basis, from which actions are taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. For most facilities, the habitability criteria of General Design Criterion 19 (GDC 19) in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," apply to this area. The control room envelope (CRE) is the plant area, defined in the facility licensing basis, that encompasses the control room and may encompass other plant areas. The structures that make up the CRE are designed to limit the inleakage of radioactive and hazardous materials from areas external to the CRE. Control room habitability

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systems (CRHSs) typically provide the functions of shielding, isolation, pressurization, heating, ventilation, air conditioning and filtration, monitoring, and the sustenance and sanitation necessary to ensure that the control room operators can remain in the control room and take actions to operate the plant under normal and accident conditions. The personnel protection features incorporated into the design of a particular plant's CRHSs depend on the nature and scope of the plant-specific challenges to maintaining control room habitability. In the majority of the CRHS designs, isolation of the normal supply and exhaust flow paths and pressurization of the CRE relative to adjacent areas are fundamental to ensuring a habitable control room.

During the design of a nuclear power plant, licensees perform analyses to demonstrate that the CRHSs, as designed, provide a habitable environment during postulated design basis events. These design analyses model the transport of potential contaminants into the CRE and their removal. The amount of inleakage of assumed contaminants is important to these analyses. Unaccounted-for contaminants entering the CRE may impact the ability of the operators to perform plant control functions. If contaminants impair the response of the operators to an accident, there could be increased consequences to the public health and safety.

There are two typical CRE designs. These designs are referred to as positive-pressure and neutral-pressure CREs. Both designs focus on limiting the amount of contaminants entering the CRE. For radiological challenges, the positive-pressure CRE intentionally pressurizes the CRE with air from outside the CRE. The pressurization air is treated by a high-efficiency particulate air filter and iodine adsorption media to remove contaminants. The neutral-pressure CRE does not intentionally pressurize the CRE, but limits inleakage of contaminants by isolating controlled flow paths into the CRE. Most plants with a positive-pressure CRE have a technical specification SR to verify that those ventilation systems serving the CRE can maintain the CRE at a positive differential pressure relative to adjacent areas. These surveillance tests (typically referred to as a ΔP surveillance) are generally implemented through a technical specification SR for the CRHSs. Plants with a neutral-pressure CRE design typically do not have a CRE integrity testing program. (The term "neutral-pressure" means only that the CRE is not intentionally pressured. The actual pressure of the CRE may be positive, neutral, or negative relative to adjacent areas.)

In addition to the ΔP surveillance described above, licensees have performed CRE integrity testing at approximately 30 percent of the power reactor facilities using the standard test method described in American Society for Testing and Materials (ASTM) consensus standard E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." Unlike the ΔP surveillance, the ASTM E741 test determines the total CRE inleakage from all sources. It is well suited for assessing the integrity of positive-pressure or neutral-pressure CREs. The test basically involves homogeneously dispersing a nontoxic tracer gas throughout the CRE and measuring the dilution of the tracer gas caused by inleakage.

The results of the ASTM E741 tests indicate that the ΔP surveillance is not a reliable method for demonstrating CRE integrity. For all but one facility tested using the ASTM E741 standard, the measured inleakage was greater than the inleakage assumed in the design basis analyses. In some cases, even though the licensees had routinely demonstrated a positive ΔP relative to adjacent areas at their facilities, the measured inleakage was several orders of magnitude greater than the value previously assumed. Affected facilities were subsequently able to achieve compliance with the control room radiation protection regulatory requirements by sealing, adding new ductwork, changing their CRE, or reanalyzing their control room habitability.

Use of the ΔP surveillance as an indicator of CRE integrity has two inherent deficiencies. First, it does not measure CRE inleakage. The ΔP surveillance infers that no contamination can enter the CRE if the CRE is at a higher pressure than adjacent areas. Second, the ΔP surveillance cannot determine whether there may be unrecognized sources of pressurization of the CRE that could introduce contaminants into the CRE under accident conditions. Two possible unrecognized contamination pathways are the CRHS fan suction ductwork that is located outside the CRE, and the pressurized ducts that traverse the lower pressure CRE en route to another plant area.

The ASTM E741 testing has helped to identify a spectrum of CRHS deficiencies that affect (1) system design, construction, and quality, (2) system boundary construction and integrity, and (3) technical specification SRs. Licensees have determined that the performance of the CRHSs can be affected by (1) the gradual degradation in associated equipment such as seals, floor drain traps, fans, ductwork, and other components, (2) the drift of throttled dampers, (3) maintenance on the CRHSs, and (4) inadvertent misalignments of the CRHSs. Since inleakage is influenced by pressure differentials between the CRE and adjacent areas, changes in ambient pressure in these adjacent areas can affect the CRE inleakage. These changes can be the result of a modification, the degradation of the ventilation systems serving these areas, or inadequate preventive and corrective maintenance programs.

Licensees and NRC staff have identified other deficiencies in CRHS design, operation, and performance from the review of license amendments, licensee event reports, and records and reports prepared pursuant to 10 CFR 50.59. These deficiencies showed that the licensees' CRHSs did not meet their design bases. Some of these deficiencies are discussed in Regulatory Issue Summary 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests." For example, some licensees credited the operation of CRHSs based upon actuation of high-radiation signals from instrumentation. Further investigation revealed that for some licensees the system would not be actuated due to incorrect setpoints or placement of the instrumentation. Other CRHS designs appear not to have considered unfiltered or once-filtered inleakage through idle CRHS ventilation trains. Without adequate consideration of such design issues, design basis radiation exposure limits may be exceeded.

Previous to the ASTM E741 testing, a group of licensees had trouble meeting the control room criteria in Three Mile Island (TMI) Action Item III.D.3.4, "Control Room Habitability Requirements," that the NRC ordered most licensees to implement after the accident at TMI. At that time, radiological source term research suggested that the distribution of the chemical forms of iodine released during an accident could be different from the distribution in the traditional source term defined in U.S. Atomic Energy Commission Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Because of the possible differences, the staff allowed licensees to postpone changing their control rooms until the ongoing source term research was completed or until a generic letter on control room habitability was issued. The staff believed that postponing changes was reasonable since the source term research or improved methods of analyses might prove that the changes were unnecessary. Many of these licensees that postponed changes incorporated compensatory actions into their operating procedures to assure that the control room operators would be protected in case of an accident. Since then, some licensees have found that they could not meet the thyroid dose limits for habitability without using compensatory actions. The NRC also allowed these facilities to use compensatory actions until completion of the source term research. In August 2000, the NRC staff incorporated the results of the source term research into Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," which is now available for use by licensees.

Although many CRE integrity testing programs focus on radiological concerns, radiation is only one potential design basis challenge to the protection of the operators. The inleakage of other contaminants may have a greater impact on control room habitability. An inleakage rate that is tolerable for one contaminant may not be tolerable for another. The control room licensing basis describes the hazardous chemical releases considered in the CRE design, the design features, and the administrative controls implemented to mitigate the consequences of these releases to the control room operators. Smoke and other byproducts of fire within the CRE or in adjacent areas are among the contaminants that can have an adverse impact on control room habitability.

Discussion

Information obtained by the NRC indicates that some licensees have not maintained adequate configuration control over their CREs and have not corrected identified design and performance deficiencies. The primary design function of CRHSs is to provide a safe environment in which the operator can control the nuclear reactor and auxiliary systems during normal operations and can safely shut down these systems during abnormal situations to protect the health and safety of the public. It is important for the operators to be confident of their safety in the control room to minimize errors of omission and commission. Errors of omission and commission are more likely if CRHSs do not properly perform as intended in response to challenges from off-normal or accident situations. The control room must be safe so that operators can remain in the control room to monitor plant performance and take appropriate mitigative actions. This is an underlying assumption in both the design basis and severe accident risk analyses. It is, therefore, imperative to the health and safety of the public that operators are safe in the control room at all times.

The scope and magnitude of the problems that NRC staff and certain licensees have identified raise concerns about whether similar design, configuration, and operability problems exist at other reactor facilities. The NRC staff is particularly concerned about whether licensees' programs to maintain configuration control of CRHSs are sufficient to demonstrate that the physical and functional characteristics of CRHSs are consistent with and are being maintained according to their design bases. It is emphasized that the NRC's position has been, and continues to be, that it is the responsibility of individual licensees to know the licensing basis for the CRHSs. Licensees should also have appropriate documentation of the design basis and procedures in place, in accordance with NRC regulations, for performing necessary assessments of plant or procedure changes that may affect the performance of the CRHSs.

The technical specifications for about 75 percent of the control rooms (mostly positive-pressure CREs) have an SR to measure the ΔP from the CRE to adjacent areas. The bases of the Improved Standard Technical Specifications state that this SR demonstrates control room integrity with respect to unfiltered inleakage. The ASTM E741 integrated testing proves that it does not. Because 10 CFR 50.36 requires technical specifications to be derived from the safety analyses, the staff believes that the existing deficiency should be corrected. This correction is consistent with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient To Assure Plant Safety," which describes the staff's expectation that licensees correct technical specifications that are found to "contain non-conservative values or specify incorrect actions."

Because of the importance of ensuring habitable control rooms under all normal and off-normal plant conditions, the addressees are requested to provide certain information that will enable the NRC staff to verify whether addressees can demonstrate and maintain the current design bases for the CRHSs at their facilities. Addressees are encouraged, but not required, to work closely with industry groups on the coordination of their responses. Coordinating the responses promotes efficiency since it leads to a uniform approach to demonstrating compliance with the design bases of their CREs.

NEI 99-03, "Control Room Habitability Assessment Guidance," provides industry generic guidance on control room habitability. The NRC staff reviewed NEI 99-03, but rather than fully endorse NEI 99-03, the NRC staff developed its own guidance. Regulatory Guide 1.196 (formerly DG-1114), "Control Room Habitability at Light-Water Nuclear Power Reactors," endorses NEI 99-03 to the extent possible and provides additional guidance. Licensees are not required to comply with Regulatory Guide 1.196, but may find it useful in responding to this generic letter. Licensees that are unable to confirm item 1 under the Requested Information section may use Regulatory Guide 1.196 to develop and implement corrective actions.

<u>Requested Information</u>

Addressees are requested to provide the following information within 180 days of the date of this generic letter.

1. Provide confirmation that your facility's control room meets the applicable habitability regulatory requirements (e.g., GDC 1, 3, 4, 5, and 19) and that the CRHSs are designed, constructed, configured, operated, and maintained in accordance with the facility's design and licensing bases. Emphasis should be placed on confirming:

    (a) That the most limiting unfiltered inleakage into your CRE (and the filtered inleakage if applicable) is no more than the value assumed in your design basis radiological analyses for control room habitability. Describe how and when you performed the analyses, tests, and measurements for this confirmation.

    (b) That the most limiting unfiltered inleakage into your CRE is incorporated into your hazardous chemical assessments. This inleakage may differ from the value assumed in your design basis radiological analyses. Also, confirm that the reactor control capability is maintained from either the control room or the alternate shutdown panel in the event of smoke.

(c) That your technical specifications verify the integrity of the CRE, and the assumed inleakage rates of potentially contaminated air. If you currently have a ΔP surveillance requirement to demonstrate CRE integrity, provide the basis for your conclusion that it remains adequate to demonstrate CRE integrity in light of the ASTM E741 testing results. If you conclude that your ΔP surveillance requirement is no longer adequate, provide a schedule for: 1) revising the surveillance requirement in your technical specification to reference an acceptable surveillance methodology (e.g., ASTM E741), and 2) making any necessary modifications to your CRE so that compliance with your new surveillance requirement can be demonstrated.

If your facility does not currently have a technical specification surveillance requirement for your CRE integrity, explain how and at what frequency you confirm your CRE integrity and why this is adequate to demonstrate CRE integrity.

2. If you currently use compensatory measures to demonstrate control room habitability, describe the compensatory measures at your facility and the corrective actions needed to retire these compensatory measures.

3. If you believe that your facility is not required to meet either the GDC, the draft GDC, or the "Principal Design Criteria" regarding control room habitability, in addition to responding to 1 and 2 above, provide documentation (e.g., Preliminary Safety Analysis Report, Final Safety Analysis Report sections, or correspondence) of the basis for this conclusion and identify your actual requirements.

Requested Response

If an addressee cannot provide the information or cannot meet the requested completion date, the addressee should submit a written response indicating this within 60 days of the date of this generic letter. The response should address any alternative course of action the addressee proposes to take, including the basis for the acceptability of the proposed alternative course of action and the schedule for completing the alternative course of action.

The written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001. A copy of the response should be sent to the appropriate regional administrator.

NRC staff will review the responses to this generic letter and, if concerns are identified, will notify affected addressees. The staff may conduct inspections to determine licensees' effectiveness in addressing this generic letter.

Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (technical specifications) pertain to the issue of control room habitability. The general design criteria for nuclear power plants (10 CFR Part 50, Appendix A), or, as appropriate, the quality assurance requirements in the licensing basis for a reactor facility (stated in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"), and the technical specifications, are the bases for the NRC staff's assessment of control room habitability.

Appendix A to 10 CFR Part 50 and the plant safety analyses require or commit licensees to design and test safety-related structures, systems, and components (SSCs) to provide adequate assurance that they can perform their safety functions. The NRC staff applies these criteria to plants with construction permits issued on or after May 21, 1971, and to those plants whose licensees have committed to them. The applicable GDC are GDC 1, 3, 4, 5, and 19. GDC 1 requires quality standards commensurate with the importance of the safety functions performed. GDC 3 requires SSCs to be designed and located to minimize the effects of fires. GDC 4 requires SSCs to be designed to accommodate the effects of accidents. GDC 5 requires that an accident in one unit will not significantly impair orderly shutdown and cooldown of the remaining unit.

GDC 19 specifies that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and maintain the reactor in a safe condition under accident conditions, including a loss-of-coolant accident. There must be adequate radiation protection to permit personnel to access and occupy the control room under accident conditions without receiving radiation exposures in excess of specified values.

Before the issuance of the GDC, proposed GDC (sometimes called "principal design criteria") were published in the *Federal Register* for comment. As they evolved, several of the proposed GDC addressed control room habitability. A facility may have been licensed before the issuance of the GDC, but the licensee may have committed to the proposed GDC as they existed at the time of licensing.

Following the accident at TMI, TMI Action Plan Item III.D.3.4, "Control Room Habitability Requirements," as clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements," required all licensees to assure that control room operators would be adequately protected against the effects of accidental releases of toxic and radioactive gases and that the nuclear power plant could be safely operated or shut down under design basis accident conditions. When licensees proposed modifications, the NRC issued orders confirming the licensees' commitments. As a result, most plants licensed before the GDC were formally adopted were then subsequently required to meet the TMI Action Plan III.D.3.4 requirements.

Appendix B to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of those SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of design. A suitable testing program is identified as one method of accomplishing this verification. Appendix B, Criterion XVI, "Corrective Action," requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected.

The regulations in 10 CFR 50.36, "Technical Specifications," require plant technical specifications to be derived from the safety analyses.

If, in the course of preparing a response to the requested information, an addressee determines that its facility is not in compliance with the Commission's requirements, the addressee is expected to take appropriate action in accordance with requirements of Appendix B to 10 CFR Part 50 and the plant technical specifications to restore the facility to compliance.

Reasons for Information Request

This generic letter transmits an information request that is necessary to permit the assessment of plant-specific compliance with applicable regulatory requirements. Specifically, this information will enable the NRC staff to determine whether the control rooms at power reactor facilities comply with the current licensing bases and whether additional regulatory actions are required.

The habitability of the control room and the operability of the CRHSs in the event of adverse environmental conditions external to the CRE have a direct link to maintaining public health and safety. Plant design bases and severe accident risk analyses both assume that the control room operators can remain safely within the control room to monitor plant performance and take appropriate mitigative actions. It is essential that operators be confident of their safety within the control room at all times.

Backfit Discussion

This generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this generic letter). This generic letter does not constitute a backfit as defined in 10 CFR 50.109(a)(1) since it does not impose modifications or additions to structures, systems, and components or to the design or operation of an addressee's facility. Nor does it impose an interpretation of the Commission's rules that is either new or different from a previous staff position. Therefore, no backfit is either intended or approved by this generic letter, and the staff has not performed a backfit analysis.

Small Business Regulatory Enforcement Fairness Act

The NRC has determined that this action is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

Federal Register Notification

A notice of opportunity for public comment was published in the *Federal Register* on May 9, 2002 (67 FR 31385). Comments were received from three licensees, three industry organizations, and one individual. The staff considered all comments that were received. The staff evaluation of these comments is accessible electronically from the Agencywide Documents Access and Management System (ADAMS) at ML030780493.

Paperwork Reduction Act Statement

This generic letter contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.) These information collections were approved by the Office of Management and Budget (OMB), approval number 3150-0011, which expires January 31, 2004.

The burden to the public for these information collections is estimated to average 200 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments regarding this burden estimate or on any other aspect of these information collections, including suggestions for reducing the burden, to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to INFOCOLLECTS@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, an information collection unless the requesting document displays a currently valid OMB control number.

/RA/
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EA-02-265 - Dresden 3 (Exelon Generation Co., LLC)

June 23, 2003

EA-02-264
EA-02-265

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: NOTICE OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTY - $60,000, AND FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING [NRC INSPECTION REPORT NO. 50-237/01-21(DRS); 50-249/01-21(DRS)] [NRC OFFICE OF INVESTIGATIONS REPORT NO. 3-2001-054]

Dear Mr. Skolds:

This refers to the inspection conducted September 24 to October 16, 2001, at Dresden Nuclear Station, Unit 3. The inspection focused on your staff's response to the discovery of damage to a piping system support in the high pressure coolant injection (HPCI) system which was indicative of a hydraulic transient. The NRC inspection report was provided to you on November 16, 2001. The NRC Office of Investigations (OI) conducted an investigation into the circumstances surrounding apparent incomplete and inaccurate information furnished to the NRC by your staff regarding the hydraulic transient. The Office of Investigations concluded that members of your staff willfully provided false information concerning: (1) the condition of HPCI system support; (2) air in the HPCI system; and (3) HPCI system peak pressure on July 5, 2001. The synopsis from the OI report and a summary of that report were provided to you on January 30, 2003. In addition, you were notified on January 30, 2003, of our preliminary determination of a White finding and an apparent violation pertaining to your failure to promptly correct a nonconforming pipe support on that system resulting in the HPCI system being inoperable from July 5 to September 30, 2001.

In review, the Dresden Station Unit 3 reactor scrammed on July 5, 2001. A HPCI actuation signal was received at about the same time as the reactor scram and the reactor operators believed they intervened in time to prevent the HPCI system from injecting. However, on July 19, 2001, your staff identified that HPCI system pipe support M1187D-80 was damaged. No other damage was observed on the HPCI system at that time. In an operability evaluation dated July 24, 2001, your staff concluded that the HPCI system was operable. In the Apparent Cause Evaluation (ACE) report dated August 24, 2001, your staff stated the apparent cause of the damage to HPCI support M1187D-80 was likely a transient (water hammer) possibly associated with the scram on July 5, 2001. On September 26, 2001, in anticipation of a scheduled telephone conference call with the NRC to discuss operability of the HPCI system, members of your staff conducted a walk down of the HPCI system and found that HPCI support M1187D-83 was loose.

During the telephone call on September 27, 2001, your staff argued that the HPCI system was operable because: (1) they had not observed other evidence of water hammer during the walk down on September 26, 2001; (2) HPCI system pressure had not exceeded 193 psig on July 5, 2001; and (3) alignment of the HPCI system to the condensate storage tank would have prevented air from entering the system. The NRC was not told during the call that your staff had found HPCI support M1187D-83 loose.

The NRC inspectors were skeptical of the position presented by your staff during the September 27, 2001, call, and continued to question your staff's position. When NRC inspectors visually examined HPCI support M1187D-83 on September 28, 2001, they observed that the support was loose and that there were chips of spalled concrete beneath the support. Based on the concerns from the NRC, your staff vented the HPCI system numerous times from September 30 to October 3, 2001, and discovered that air was trapped in the system. On October 16, 2001, your staff was able to retrieve data from the Transient Analysis Data System (TADS) indicating HPCI system pressure on July 5, 2001, had reached approximately 1000 psig. You concluded this information was not provided to the NRC in a timely manner.

A closed, transcribed predecisional enforcement conference (PEC) was held on April 23, 2003, in the NRC Region III office with members of your staff to discuss the apparent violation, its significance, its root causes, and your corrective actions. At the PEC, your staff admitted that Exelon Nuclear, without willfulness, failed to provide the NRC with complete and accurate information, in violation of 10 CFR 50.9, "Completeness and Accuracy of Information," concerning: (1) the status of HPCI support M1187D-83; (2) HPCI system pressure during the July 5, 2001 event; and (3) the need for venting the HPCI system.

Based on information developed during the NRC inspection, the OI investigation, and at the PEC, the NRC determined that violations occurred. The violations are cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding them were described in the previously provided inspection report, Licensee Event Report (LER) 2002-005-00, dated December 3, 2002, and the synopsis and summary of the OI report. The NRC recognizes that your staff admitted non-willful violations of 10 CFR 50.9 during the PEC. However, the NRC determined that the failure to describe the known condition of HPCI support M1187D-83 during the September 27, 2001, telephone conference call is a willful violation, representing at least careless disregard. During the telephone call, the NRC staff described bent and loose supports as well as spalled concrete as possible indications of water hammer, but your staff did not inform the NRC that HPCI support M1187D-83 was loose. Your staff, participating in the telephone call, stated that they had performed a walk down of the HPCI system, that support M1187D-83 was not damaged, and that no other signs of a water hammer were observed. However, on September 28, 2001, NRC inspectors found that HPCI support M1187D-83 was loose and did not carry any of its 4,000 pound design load. One of your employees commented during the PEC that he had said nothing during the September 27, 2001, conference call with the NRC about his observation that support M1187D-83 was loose on September 26, 2001. It was not brought up because, in the individual's opinion, water hammer did not cause the support to become loose, so the loose hanger was not relevant and the individual felt that discussion of the support would confuse the issue.

In assessing this violation, the NRC considered the training, education and experience of the Exelon employees involved and their knowledge that the ACE report documented that a transient (water hammer) had likely occurred. The inaccurate information was material to the NRC because the NRC staff was evaluating your operability determination for the HPCI system. It is essential that licensees disclose all pertinent information to the NRC particularly when technical disagreements arise. In this case, the condition of a specific support was being discussed and an accurate description of its condition was not provided to the NRC staff. Therefore, this violation is categorized in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (NRC Enforcement Policy) NUREG-1600, at Severity Level III (EA-02-265).

In accordance with the Enforcement Policy, a base civil penalty in the amount of $60,000 is considered for a Severity Level III violation. Because this was a willful violation of NRC requirements, the NRC considered whether credit was warranted for *Identification* and *Corrective Action* in accordance with the Enforcement Policy. Credit was not warranted for *Identification* because the NRC identified the violation. Credit was warranted for the *Corrective Action* civil penalty adjustment factor. Corrective actions included, but were not limited to: providing additional guidance and training on operability determinations and issue identification and management; providing written expectations to the technical staff on issue resolution; meeting with each involved individual on the need to provide complete and accurate information to the NRC; and informing the other Exelon Nuclear and AmerGen facilities of the lessons learned.

Therefore, to emphasize the importance of accurate and complete information, I have been authorized, after consultation with the Director, Office of Enforcement, to issue the enclosed Notice of Violation and Proposed Imposition of Civil Penalty (Notice) in the base amount of

$60,000 for the Severity Level III violation.

During the PEC, your staff admitted non-willful violations, concerning HPCI system pressure and venting, stating each represented poor performance by employees of Exelon Nuclear. No enforcement action is proposed for those two violations in accordance with Section IX of the NRC Enforcement Policy because your staff provided information they thought to be accurate at the time and provided the correct information after discovering their error. While enforcement action is not being proposed, each instance represents a failure by your staff to display the questioning attitude toward a safety issue that you should expect.

Licensee Event Report 2002-005-00 indicated that the HPCI system was inoperable for the period following the reactor scram on July 5, 2001, until September 30, 2001. On January 30, 2003, the NRC informed your staff that the NRC had preliminarily categorized the inspection finding for the inoperable HPCI system as White, in accordance with the NRC Significance Determination Process (SDP). The opportunity for a regulatory conference to discuss the White finding and associated violation was offered to Exelon Nuclear. On February 5, 2003, Mr. Keith Jury of your staff informed the NRC that a regulatory conference would not be requested. After considering the information developed during the inspection and in the LER, the NRC has concluded that the inspection finding is appropriately characterized as White, an issue with low to moderate increased importance to safety which may require additional inspections by the NRC. You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also determined that the failure to promptly correct the damaged HPCI system support resulting in the equipment being inoperable for greater than the allowed outage time is a violation of Technical Specification 3.5.1 and of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," as cited in the attached Notice. The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the violation is considered escalated enforcement action because it is associated with a White finding (EA-02-264).

Because plant performance for the corrective action issue was determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event and notify you, by separate correspondence, of that determination.

You are required to respond to this letter and should follow the instructions in the enclosed Notice when preparing your response. The NRC will use that response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web sire at http:wwwnrc.gov/reading. rm/adams-html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the Public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do, Enforcement,** then **Significant Enforcement Actions.**

Sincerely,

/RA/ James L. Caldwell for

J. E. Dyer
Regional Administrator

Docket No. 50-249
License No. DPR-25

Enclosures:
1. Notice of Violation and Proposed Imposition of Civil Penalty
2. NUREG/BR-0254 Payment Methods (Licensee Only)

cc w/encl. 1:
Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional
Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional
Operating Group
Manager Licensing - Dresden and Quad Cities
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
Document Control Desk - Licensing
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission
J. Mikan, Will County Executive/
Board Chairman
P. Kaupas, Will County Sheriff
W. Ferguson, Will County Emergency
Management Director
The Honorable Arthur Schultz
J. Mezera, City Manager
J. Church, Kendall County Board Chairman
R. Randall, Kendall County Sheriff
P. Nelson, Grundy County Board Chairman
J. L. Olson, Grundy County Sheriff
J. Lutz, Grundy County Emergency
Management Coordinator/Director
The Honorable Richard Kopczick
The Honorable C. Richard Ellis
The Honorable Gerald V. Pierard
The Honorable Joseph Fracaro
The Honorable Elmer Rolando
The Honorable Richard Girot
The Honorable Tony McGann
The Honorable Wayne Chesson
M. T. Gibson, Channahon Village Administrator
The Honorable Richard Chapman
K. Carroll, Shorewood Village Administrator
The Honorable Robert Blum
INPO

NOTICE OF VIOLATION
AND
PROPOSED IMPOSITION OF CIVIL PENALTY

Exelon Generation Company, LLC	Docket No. 50-249
Dresden Nuclear Station	License No. DPR-25
Unit 3	EA-02-264; EA-02-265

During an NRC inspection from September 24 to October 16, 2001, and an NRC investigation completed on April 22, 2002, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the NRC proposes to impose a civil penalty pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (Act), 42 U.S.C. 2282, and 10 CFR 2.205. The particular violations and associated civil penalty are set forth below:

A. Violation Assessed a Civil Penalty (EA-02-265)

10 CFR 50.9 provides, in part, that information provided to the Commission by a licensee shall be complete and accurate in all material respects.

Contrary to the above, during a telephone conference call on September 27, 2001, Exelon Nuclear failed to provide complete and accurate information to the NRC Region III staff concerning the high pressure coolant injection (HPCI) system for Dresden Nuclear Station, Unit 3. Specifically, during the call, the NRC staff described various indications of a potential water hammer, including damaged, bent, or loose pipe supports and spalled concrete. In response during the call, Exelon Nuclear staff told the NRC that, had a water hammer occurred following a reactor scram on July 5, 2001, HPCI support M1187D-83 would have been damaged. They stated that they had conducted a walk down of the system on September 26, 2001, that HPCI support M1187D-83 was not observed to be damaged, and that no other signs of a water hammer existed. One employee of Exelon Nuclear found that HPCI support M-1187D-83 was loose during a visual examination on September 26, 2001, and did not provide that information to the NRC on September 27, 2001. The incomplete and inaccurate information provided to the NRC on September 27, 2001, was material to the NRC because the NRC staff was evaluating the licensee's operability determination for the Dresden Nuclear Station, Unit 3, HPCI system.

This is a Severity Level III violation (Supplement VII).
Civil Penalty - $60,000.

B. Violation Not Assessed a Civil Penalty (EA-02-264)

Dresden Nuclear Station Technical Specification 3.5.1 requires, in part, that for operating Mode 1, should the high pressure coolant injection (HPCI) system become inoperable, the HPCI system must be restored to an operable status within 14 days or the plant be placed in Mode 3 in 12 hours .

10 CFR Part 50, Appendix B, Criterion XVI, provides, in part, that conditions adverse to quality be promptly corrected, and in the case of a significant condition adverse to quality, that corrective action be taken to preclude repetition.

Contrary to the above, as of September 30, 2001, the licensee had not promptly corrected damaged pipe support M1187D-80 on the Dresden Nuclear Station Unit 3 HPCI system, after it was identified on July 19, 2001. The licensee did not take corrective action to preclude repetition of the damage to support M1187D-80, a significant condition adverse to quality, until prompted by the NRC on September 30, 2001. As a result, while the plant was operating in Mode 1, the HPCI system was inoperable from July 5, 2001, to September 30, 2001, a period in excess of 14 days.

This violation is associated with a White SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Exelon Nuclear (Licensee) is hereby required to submit a written statement or explanation to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, within 30 days of the date of this Notice of Violation and Proposed Imposition of Civil Penalty (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation: EA-02-264; EA-02-265" and should include for each of the alleged violations: (1) admission or denial of the alleged violation, (2) the reasons for the violation if admitted, and if denied, the reasons why, (3) the corrective steps that have been taken and the results achieved, (4) the corrective steps that will be taken to avoid further violations, and (5) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked or why such other action as may be proper should not be taken. Consideration may be given to extending the response time for good cause shown. Under the authority of Section 182 of the Act, 42 U.S.C. 2232, this response shall be submitted under oath or affirmation.

Within the same time as provided for the response required above under 10 CFR 2.201, the Licensee may pay the civil penalty proposed above, in accordance with NUREG/BR-0254 and by submitting to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555, a statement indicating when and by what method payment was made, or may protest imposition of the civil penalty in whole or in part, by a written answer addressed to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission. Should the Licensee fail to answer within the time specified, an order imposing the civil penalty will be issued. Should the Licensee elect to file an answer in accordance with 10 CFR 2.205 protesting the civil penalty, in whole or in part, such answer should be clearly marked as an "Answer to a Notice of Violation" and may: (1) deny the violations listed in this Notice, in whole or in part, (2) demonstrate extenuating circumstances, (3) show error in this Notice, or (4) show other reasons why the penalty should not be imposed. In addition to protesting the civil penalty, in whole or in part, such answer may request remission or mitigation of the penalty.

In requesting mitigation of the proposed penalty, the factors addressed in Section VI.C.2 of the Enforcement Policy should be addressed. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate parts of the 10 CFR 2.201 reply by specific reference (e.g., citing page and paragraph numbers) to avoid repetition. The attention of the Licensee is directed to the other provisions of 10 CFR 2.205, regarding the procedure for imposing a civil penalty.

Upon failure to pay any civil penalty due which subsequently has been determined in accordance with the applicable provisions of 10 CFR 2.205, this matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234(c) of the Act, 42 U.S.C. 2282c.

The response noted above (Reply to Notice of Violation, EA-02-264; EA-02-265, statement as to payment of civil penalty, and Answer to a Notice of Violation) should be addressed to: Frank J. Congel, Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, and the Resident Inspector at the Dresden Nuclear Station.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated this 23rd day of June 2003.

EA-030-060 - Callaway (Union Electric Co.)

June 20, 2003

EA-03-060

Garry L. Randolph, Senior Vice
President and Chief Nuclear Officer
Union Electric Company
P.O. Box 620
Fulton, Missouri 65251

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC INSPECTION NO. 50-483/03-08; CALLAWAY PLANT)

Dear Mr. Randolph:

The purpose of this letter is to provide you the final results of our significance determination of the preliminary White finding identified in the subject inspection report. The inspection finding was assessed using the Significance Determination Process and was preliminarily characterized as White, a finding with low to moderate increased importance to safety, which may require additional NRC inspections. This White finding involved a failure to establish the means to notify certain members of the public in your emergency planning zone in the event of an emergency at your Callaway plant. The finding was based on the conclusions that from 1998 through November 2002: (1) your database of tone alert radio recipients was inaccurate and was continuing to degrade due to programmatic and implementation inadequacies; (2) the failure to maintain an accurate database resulted in the failure to distribute tone alert radios to members of the public that required tone alert radios for emergency alerting; and (3) your program was not capable of identifying the errors in a timely manner such that compensatory measures could be taken to alert affected members of the public.

In a telephone conversation with Mr. Troy Pruett of my staff on or about May 15, 2003, Mr. Mark Reidmeyer of your staff indicated that Union Electric Company did not contest the characterization of the risk significance of this finding and that you declined your opportunity to discuss this issue in a Regulatory Conference. He stated that you would provide a written response to the subject inspection report.

The NRC received your response letter dated June 10, 2003. This letter confirmed your acceptance of the White finding as preliminarily characterized, but also requested clarification of our characterization of the cross cutting aspects of the finding which are documented in the subject inspection report. The NRC acknowledges and agrees with your comments in the letter concerning the promptness and adequacy of the immediate actions you took following your November 2002 discovery of the inadequate distribution of tone alert radios. Our primary cross cutting concern related to the White finding was the failure of your audit programs and supervisory oversight of surveillance activities to identify the inaccurate tone alert radio database prior to the occurrence of an external event (change in electric service providers) which prompted its discovery. The subject inspection report inaccurately characterized these failures as a human performance cross cutting issue. This inspection did not evaluate the effectiveness of your corrective action programs and processes, but concluded that the White finding had cross cutting aspects related to problem identification. The cross cutting aspects of the White finding were documented in Section 4OA2 to facilitate future NRC inspection.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC has also determined that the failure to establish the means to notify certain members of the public in the emergency planning zone is a violation of 10 CFR 50.47(b)(5), as cited in the attached Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, NUREG-1600, the Notice of Violation is considered escalated enforcement action because it is associated with a White finding (50-483/0308-01).

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response.

Because plant performance for this issue has been determined to be in the regulatory response band, we will use the NRC Action Matrix to determine the most appropriate NRC response for this event. We will notify you, by separate correspondence, of that determination.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Thomas P. Gwynn
Acting Regional Administrator

Enclosure: Notice of Violation

Docket: 50-483
License: NPF-30

cc w/Enclosure:

Professional Nuclear Consulting, Inc.
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Derwood, Maryland 20855

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Regulatory Affairs Supervisor
Regulatory Affairs
AmerenUE
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Missouri Public Service Commission
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Ronald A. Kucera, Deputy Director
for Public Policy
Department of Natural Resources

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NOTICE OF VIOLATION

Union Electric Company
Callaway Plant

Docket No. 50-483
License No. NPF-30
EA-03-060

During an NRC inspection conducted on February 10 through March 21, 2003, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR 50.54(q) provides in part that "[a] licensee authorized to possess and operate a nuclear power reactor shall follow . . . emergency plans which meet the standards in [section] 50.47(b). . . ."
>
> 10 CFR. 50.47(b) requires that the onsite emergency response plans for nuclear power reactors must meet each of 16 planning standards, of which, standard (5) states, in part: the ". . . means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established." The licensee's emergency plan described the means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone (EPZ) to include tone alert radios and emergency sirens.
>
> Contrary to the above, from 1998 through November 2002, the licensee failed to follow its emergency plan designed to meet planning standard (5) in 10 C.F.R. 50.47(b). Specifically, the licensee failed to provide tone alert radios to 98 residences in portions of the EPZ that relied upon tone alert radios as the primary means of emergency notification (i.e., areas of the EPZ that were outside of the range of emergency sirens).
>
> This violation is associated with a White Significance Determination Process finding.

Pursuant to the provisions of 10 CFR 2.201, Union Electric Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, and a copy to the NRC Resident Inspector at the Callaway Plant, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-03-060," and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an Order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. ADAMS is accessible from the NRC Web site at the Public Electronic Reading Room, http://www.nrc.gov/reading-rm/adams.html. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 20th day of June 2003

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

June 24, 2003

NRC INFORMATION NOTICE 2003-07: WATER IN THE VENT HEADER/VENT LINE SPHERICAL JUNCTIONS

Addressees

All holders of operating licenses for boiling water reactors (BWRs) with a Mark I containment.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to inform addressees of recent issues involving the pressure suppression containment system in BWRs with a Mark I containment. During a recent refueling outage at Nine Mile Point Unit 1 (NMP1), unanticipated standing water was found inside the vent header/vent line (VH/VL) spherical junctions (vent system low point or "bowl"). The weight of this standing water inside the VH/VL spherical junctions was not included in the generic Mark I containment accident analysis because the spherical junctions are assumed to remain dry. This standing water inside the VH/VL spherical junction increases the thrust loads on the vent system. The primary concern is that this standing water will increase vent system thrust loads during reactor blowdown after a loss-of-coolant accident inside containment beyond design limits.

The licensees for other plants with Mark I containment designs have also noted standing water in the VH/VL spherical junctions (Pilgrim, Hope Creek, and Fermi). It is expected that recipients will review the information in this notice for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Background

The pressure suppression containment system of a Mark I BWR consists of a drywell, a torus-shaped pressure suppression chamber, which is approximately half filled with water, a connecting vent system between the drywell and the pressure suppression chamber, isolation valves, a vacuum relief system, a containment cooling system and other service equipment. An illustration of portions of this system is provided in Figure 1.

The vent pipe descending from the drywell joins the ring header at a VH/VL spherical junction. There are 8 to 10 vent pipes and spherical junctions in most Mark I containments. The ring header is arranged within the suppression chamber shell, with downcomer pipes from the header extending below the water surface in the suppression chamber. Submergence of the downcomer pipes is operationally maintained by a minimum required suppression chamber water level in Technical Specifications.

ML031750146

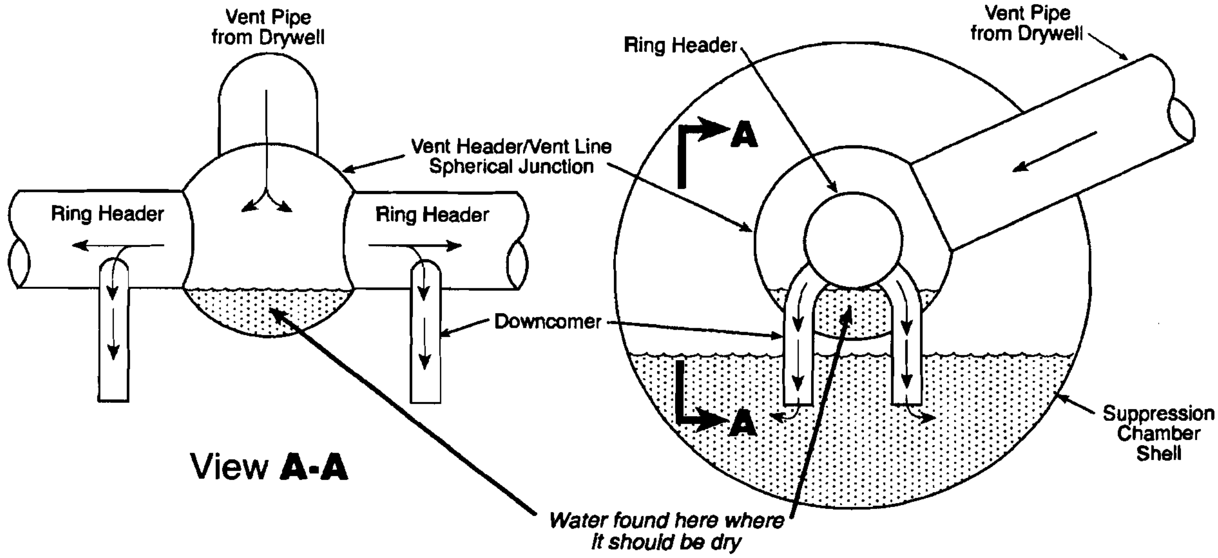

Figure 1 BWR Mark I Pressure Suppression Containment System (Partial View)

The limiting event for containment pressurization and vent system flow rates is the design basis loss-of-coolant accident (DBA-LOCA). The DBA-LOCA results in the maximum pressurization rate, maximum pressure, and highest vent system flow rates; therefore, it produces the highest vent system thrust loads. The vent system thrust load is a function of the vent system pressure relative to the suppression chamber air space and the mass flow and velocity through vent piping. The Mark I Containment Program for NMP1 has previously defined vent system thrust loads for the DBA-LOCA based on these parameters.

Description of Circumstances

During a refueling outage, NMP1 personnel discovered that the VH/VL spherical junctions contained approximately 3 feet of standing water or 1100 gallons per sphere (11,000 gallons total) where it should be dry. Because of the system geometry, the volume of water in the spherical junction was at its maximum. Addition of more water would result in spilling into the ring header and downcomers. The source of the water is believed to be condensation in the relatively cool vent header lines. The original plant design of some Mark I containments had drain lines from the spherical junctions to the torus. Some of the plants having these drain lines removed them in the early 1980s to eliminate a potential torus bypass path. These drain lines were not part of the original design at NMP1 and were not installed.

The licensee's analysis of the standing water in the spheres concluded that the mass could become entrained in the initial blowdown and would increase the thrust loads during a LOCA. A subsequent analysis demonstrated that the majority of the system components met American Society of Mechanical Engineers (ASME) Code allowable stress values with the exception of the VH/VL spherical junction at the connection to the ring header. The calculated stress level for the VH/VL junction exceeded the original design acceptance criteria (ASME Service Level A/B), but remained below ASME Service Level C and the higher acceptance stress level limits for operability (ASME Service Level D). The guidance provided in NRC Generic Letter 91-18 was used to demonstrate that the VH/VL spherical junction stress levels remained operable.

At the time of this information notice, analysis was continuing to determine if additional actions were needed to restore compliance with the original design criteria.

This information notice requires no specific action or written response. If you have any questions regarding the information notice, please contact the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
William D. Beckner, Program Director
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

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Attachment: List of Recently Issued NRC Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

June 19, 2003

NRC INFORMATION NOTICE 2003-06: FAILURE OF SAFETY-RELATED LINESTARTER RELAYS AT SAN ONOFRE NUCLEAR GENERATING STATION

Addressees

All holders of operating licenses or construction permits for nuclear power reactors, except those that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of recent failures of safety-related valves due to linestarter relay degradation. The degradation was caused by past use of excessive amounts of trichloroethane-based cleaners during preventive maintenance. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

San Onofre Nuclear Generating Station utilizes reversing linestarters manufactured by Square D to operate the motors on safety-related motor-operated valves. The linestarter consists of two relays that provide 480 volt power to the motor and contain auxiliary contacts associated with interlock and seal-in functions. The interlock function provides a means to avoid energizing both open and closed relays at the same time. The seal-in function keeps the relay energized until the valve has completed its stroke. All reversing linestarters have interlock auxiliary contacts. San Onofre has 172 Square D linestarters associated with safety-related motor-operated valves, 86 in each unit.

On August 30, 2002, a Unit 3 low-pressure safety injection (LPSI) pump mini-recirculation valve failed to open during surveillance testing. Subsequent analysis determined that the plastic housing on an auxiliary contact in the associated linestarter was degraded. The licensee determined that the auxiliary contact housing degradation was caused by the past use of excessive amounts of Inhibisol, a cleaning solvent based on trichloroethane (TCE). The cleaning solvent caused the plastic to break down. Over time, small amounts of the plastic came loose and interfered with the electrical contacts, resulting in the valve failure.

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In response to the LPSI pump mini-recirculation valve failure, San Onofre developed a plan to inspect a sample of other safety-related linestarters installed in Units 2 and 3. In October 2002, San Onofre completed the inspection of 19 additional linestarters. This sample inspection identified two auxiliary contacts that showed signs of chemical attack (i.e., cloudy plastic contact housing); however, both were found to be functional. Subsequently, the licensee developed a risk-informed plan to inspect all safety-related linestarters and replace all auxiliary contacts showing signs of chemical attack. The linestarter inspections were scheduled into online and outage maintenance windows, and will be completed by the end of the Unit 2 outage in 2004.

On January 18, 2003, during a Unit 3 refueling outage, the quench tank sample containment isolation valve failed to open during surveillance testing. Examination of the contact revealed that a similar chemical attack had occurred and caused the valve failure.

On February 10, 2003, during an inspection of Unit 3 LPSI header stop valve linestarters, an auxiliary contact failed on the 20th cycle of the auxiliary contact test. The linestarter inspections included a test to cycle each auxiliary contact 20 times. This auxiliary contact cycle test was performed to determine the functionality of the auxiliary contacts in the linestarter.

On Unit 3, all 86 linestarters have been inspected with two surveillance test failures noted and one maintenance test failure. The licensee replaced 42 auxiliary contacts from the linestarters due to evidence of chemical attack on the plastic auxiliary contact housing. On Unit 2, 33 linestarters have been inspected as of May 2, 2003, with no failures noted; however, four auxiliary contacts showed signs of chemical attack on the plastic contact case.

Discussion

As a result of the valve stroke failure on August 30, 2002, the licensee initiated a laboratory analysis of the suspect auxiliary contact from the linestarter. The contact was coated with a plastic residue from the deterioration of the plastic switch bodies. The licensee concluded that excessive use of cleaning solvents during previous preventive maintenance activities had caused the failure of the contacts.

The licensee believes that all damage to the auxiliary contact housings occurred prior to 1989 and is showing up in the recent safety-related valve failures. The original linestarter preventive maintenance procedure was issued in April 1984, and required the use of cleaning solvents on linestarters, but had no caution regarding the potential for damage to plastic components within the linestarter. Also, the procedure did not require visual inspection of internally mounted auxiliary contact assemblies. As a result, Inhibisol was used liberally, which allowed the cleaner to come in contact with plastics that were susceptible to chemical degradation. In April 1989, the licensee recognized that TCE-based cleaners were being used improperly and that controls needed to be implemented to prevent future damage to equipment containing plastics. The licensee revised the consumables controls manual to restrict the use of TCE-based cleaners on plastics, and provided guidance on the approved method for use of the cleaner (i.e., spray on cloth, then wipe component). Additionally, the linestarter preventive maintenance procedure was revised to caution that cleaning solvents should be used sparingly to avoid damage to plastic components. In response to the recent valve failures, the licensee took action on March 7, 2003, to prohibit the use of all TCE-based cleaners for electrical maintenance applications.

The licensee missed several opportunities from plant and industry experience to recognize the need for an extent-of-condition review. An extent-of-condition review could have identified any equipment degradation that occurred throughout the plant due to improper use of cleaning solvents. One of these prior opportunities was the review of Information Notice 93-76, "Inadequate Control of Paint and Cleaners for Safety Related Equipment," which the licensee performed in February 1994. The review determined that the programs in place were sufficient to avoid problems similar to those discussed in the notice. The licensee focused on the TCE-based cleaner controls in place at the time of the information notice review, but overlooked the fact that safety-related equipment could have been damaged prior to the implementation of the controls in April 1989. This oversight was a missed opportunity to correct the equipment deficiency that has been revealed by the recent linestarter failures and the discovery of degraded contacts.

The San Onofre linestarter experience emphasizes the need to perform an extent-of-condition review to determine equipment impact when an improper maintenance practice is recognized and corrected. Further, the root cause analysis revealed that past improper use of corrosive cleaners could result in degraded plant equipment that could remain undetected for a considerable length of time before showing up in equipment failures.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
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Attachment: List of Recently Issued NRC Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

June 5, 2003

NRC INFORMATION NOTICE 2003-05: FAILURE TO DETECT FREESPAN CRACKS IN PWR STEAM GENERATOR TUBES

Addressees

All holders of operating licenses or construction permits for pressurized-water reactors (PWRs).

Purpose

This information notice (IN) is being provided to inform licensees of a recent problem experienced at Comanche Peak Unit 1 concerning the detection of freespan outside diameter stress corrosion cracking (ODSCC) in steam generator (SG) tubes. This has led to tube integrity performance criteria not being met as defined in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines." The NRC anticipates that recipients will review the information for applicability to their facilities and consider taking appropriate actions. However, suggestions contained in this IN do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Comanche Peak Unit 1 is a four-loop Westinghouse PWR with four Westinghouse Model D4 recirculating SGs (1, 2, 3, 4). Each SG contains 4578 mill- annealed Alloy 600 tubes, which are nominally 0.750 inch in diameter and have a nominal wall thickness of 0.043 inch. The tubes are supported by a number of carbon steel tube support plates with circular holes and by V-shaped chrome-plated Alloy 600 anti-vibration bars (AVBs).

Comanche Peak Unit 1 was shut down approximately 1 week prior to its scheduled refueling outage as a result of a primary-to-secondary leak. A 5- to 15-gallon-per-day (gpd) leak was first observed in SG 2 on September 26, 2002. Over the next 2 days, the leakage spiked to higher values several times. On September 28, 2002, after a leakage spike to 52 gpd, the licensee elected to shut down the plant and to commence refueling (1RF09). In response to the leak, a special inspection by the NRC staff was conducted. The results of the special inspection were documented in an inspection report dated January 9, 2003, "Comanche Peak Steam Electric Station - Special Team Inspection Report 50-445/02-09" (ADAMS Accession No. ML030090566).

After shutting down the plant, the licensee began inspecting the SG tubes with eddy current testing techniques. A bobbin coil and a rotating probe were used during these inspections. The rotating probe was equipped with various types of coils including a +Point™ coil. The bobbin coil was used to inspect the full length of each tube while the rotating probe was used to inspect selected regions of the tube (e.g., the top of tubesheet region) and to confirm and/or characterize indications initially detected by the bobbin coil probe.

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The licensee determined that the leak was from an axially oriented flaw in the tube located at row 41, column 71 (R41C71) in SG 2. The flaw, located in the U-bend region, was estimated from the +Point™ coil to be approximately 0.9 inches in length, with a depth of approximately 90 percent over most of the indicated length. The licensee's structural assessment of the flaw indicated that the leaking tube did not meet the applicable structural and accident leakage performance criteria in NEI 97-06. These performance criteria were developed consistent with the plant design and licensing basis and include the three-times-normal-operating-pressure criterion against burst (3800 pounds per square inch (psi)), the 1.4 times main steam line break (MSLB) criterion against burst (3584 psi), and a 1- gallon-per-minute (gpm) MSLB- induced leak rate criterion. The licensee estimated the burst pressure of R41C71 to be 2727 psi at the location of the flaw based on analysis of the flaw profile as determined by the +Point™ coil. In situ pressure testing of this tube was terminated at a test pressure of 2100 psi when leakage exceeded the test system capacity of 2.5 gpm.

Review of the bobbin data for this tube (R41C71) from the previous inspection in 2001 (1RF08) revealed that a clearly detectable indication was present at the location of the leak. This bobbin indication did not meet the reporting criteria in the 1RF08 eddy current data analysis guidelines and was not reported by either the primary or the secondary analyst in 2001. These reporting criteria required a freespan bobbin indication in the absence of a dent or ding signal to be reported if the phase angle response of the indication was less than the phase angle response corresponding to a 0 percent through-wall flaw. Since a dent or ding signal can rotate a flaw signal out of the normal phase angle window, the applicable reporting criteria for bobbin indications in the presence of a detectable dent or ding signal were less restrictive (i.e., were increased). If a dent or ding signal had been reported at this location in 1RFO8, the bobbin indication in tube R41C71 would have been reportable. A reportable bobbin indication might have triggered additional inspections with a rotating probe. However, no ding signal was reported at the R41C71 location in 1RF08 by either the primary or the secondary analysts during their review of the bobbin data since there was no clear evidence of a ding in the 1RF08 signal response. However, a large amount of horizontal noise attributable to probe wobble was observed. This amount of horizontal noise could easily mask a 2 volt ding signal.

Based on these findings, the licensee revised its bobbin probe data analysis procedures for the 1RF09 inspection to increase the phase angle response reporting criteria for freespan indications. The ensuing inspections identified about 20 freespan flaws. These included freespan flaws associated with dents and dings and long freespan flaws not associated with dents or dings. However, examination of the inspection results called into question the reliability of the bobbin inspection. Of the 20 freespan flaws, only 5 had been detected during both the primary analysis of the bobbin data, performed using automated (computerized) data screening (ADS), and the secondary analysis of the bobbin data, performed by human analysts. The primary (ADS) analysis missed several of the bobbin indications called by the secondary (human) analysis and vice versa. In general, the bobbin indications missed by the primary (ADS) analysis exhibited bobbin amplitude responses less than the 0.2-volt ADS threshold. Furthermore, 8 of the 20 freespan flaws were not detected by either the primary or secondary analysis of the bobbin data. These eight freespan flaws were found fortuitously rather than by programmatic intent. They were found only because the licensee had performed a more comprehensive +Point™ examination of the region to investigate an indication or dent located elsewhere in the same region of tube where the flaw was eventually found.

Accordingly, the licensee retrained the analysts and manually performed a third (tertiary) independent analysis of the bobbin coil data, leading to the finding of additional freespan bobbin indications. Several of these additional freespan bobbin indications were confirmed as flaws during the +Point™ coil examination. Of these confirmed flaws, two had been detected during the aforementioned primary and/or secondary analysis of the bobbin coil data. These bobbin indications were not investigated with a +Point™ coil following the primary (ADS) and secondary (manual) analysis since the bobbin signals at these locations were perceived to be similar to those observed in 1999 (i.e., there was a perceived lack of change in the bobbin coil signal indicating that the bobbin indication was not a result of a flaw, but rather it was within the expected range of repeatability of the bobbin test). However, during the tertiary analysis, the review of the prior inspection data for these indications revealed clear indications of signal change, calling into question the effectiveness of the prior history reviews for bobbin indications.

To address this concern, the licensee prepared data analysis guidelines for the history reviews and performed a new, supplemental history review of all bobbin indications. Two qualified data analysts working as a team performed this supplemental review. They considered all data extending back to the first inservice inspection, including data from the low-frequency absolute channel. The analysts were also instructed to identify not only indications with changes exceeding change criteria specified in the data analysis guidelines, but also indications with changes which, in their experience and judgment, were beyond changes associated with normal eddy current signal repeatability. This review led to the finding of three additional flaws.

Discussion

Early detection of stress corrosion cracks is key to ensuring that such cracks do not impair tube integrity relative to the tube integrity performance criteria in NEI 97-06. It continues to be standard industry practice to use bobbin probes to screen for indications potentially associated with axially oriented stress corrosion cracks and, where such indications are found, to perform a followup inspection with a rotating, surface-riding coil such as a pancake or +Point™ coil to determine whether a crack is actually present. As evidenced by the recent experience at Comanche Peak Unit 1, appropriate data analysis procedures, analyst training, and process controls are critical to ensuring that all indications of actual stress corrosion cracking are being identified during the bobbin coil data analysis and subsequently inspected with a +Point™ coil. The following are some of the lessons learned from the recent experience at Comanche Peak Unit 1.

1. Care should be exercised when establishing reporting criteria for indications based on phase angle response. Dings, dents, and other artifacts can rotate a flaw indication outside the nominal range of phase angle response, even where the amplitude of such artifacts is relatively small or less than the reporting value for such artifacts.

2. The presence of artifact signals which may potentially distort flaw indications can themselves be masked by other artifacts such as probe wobble. Probe wobble signals tend to be particularly large in the U-bend region of a tube.

3. Depending on the value of the threshold criteria, indications with voltage responses less than the ADS threshold criteria may sometimes be associated with flaws whose maximum depths exceed the tube plugging limit (e.g., 40 percent through-wall). Thus, data analysis procedures (including ADS threshold criteria) should be sufficiently robust

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to reliably identify indications which may potentially exceed the plugging limit. For example, the use of ADS at some plants is supplemented by an independent review of the data by two teams of human analysts.

4. A comparative review of indications called by the primary and secondary analysis teams can provide insights on the effectiveness of the analysis effort. As an illustration, failure of the primary or secondary analysis team to detect a high fraction of the indications identified by the other team may be indicative of a need to evaluate the cause of the discrepancies and whether corrective actions are needed with respect to the examination technique, data analysis guidelines, and/or analyst training.

5. A robust approach is important for determining which bobbin indications exhibit change over time in order to ensure all potential flaws are further evaluated (e.g., with a rotating probe). A team could review the previous bobbin coil data for each indication identified during an inspection or multiple independent reviews of the previous bobbin coil data could be done. The analysts might be allowed to use their judgment and experience in determining whether there has been a change in addition to determining whether specific change criteria on phase angle and amplitude have been met. In addition, previous inspection data could be reviewed as far back in time as possible since the bobbin response for some of the flaws at Comanche Peak Unit 1 did not show a change when compared only to the most recent previous inspection data.

6. The bobbin data from the low-frequency absolute data channel can sometimes be helpful in detecting long freespan indications and for observing changes in these signals over time.

7. The insertion of known flaw signals from a "Judas" (or "Cobra") tube into the data stream being reviewed by each data analyst can provide additional confidence in the performance level of the analysts. This insertion could be done in such a manner that the data analysts could not tell that the inserted flaw signal did not belong to the population of actual flaws they were currently analyzing. At Comanche Peak Unit 1, the Judas tube was a tube containing indications missed during the primary and secondary analysis and found fortuitously during the subsequent +Point™ examination.

<u>Related Generic Communications</u>

The following documents describes other recent reactor operating experience with steam generator tubes:

1. IN 2002-02 and IN 2002-02 supplement 1, "Recent Experience With Plugged Steam Generator Tubes" dated January 8, 2002 and July, 17, 2002

2. IN 2002-21, "Axial Outside-Diameter Cracking Affecting Thermally Treated Alloy 600 Steam Generator Tubing" dated June 25, 2002

3. IN 2001-16, "Recent Foreign and Domestic Experience with Degradation of Steam Generator Tubes and Internals," dated October 31, 2001

4. NRC Generic Letter 97-05, "Steam Generator Tube Inspection Techniques," dated December 17, 1997

This information notice does not require any specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate project manager in the NRC's Office of Nuclear Reactor Regulation (NRR).

/RA/
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Attachments: List of Recently Issued NRC Information Notices

June 26, 2003

The Honorable David L. Hobson, Chairman
Subcommittee on Energy and Water Development
Committee on Appropriations
United States House of Representatives
Washington, D.C. 20515

Dear Mr. Chairman:

The FY 2003 Energy and Water Development Appropriations Act, (House Report 108-10 and Senate Report 107-220) directed the U.S. Nuclear Regulatory Commission (NRC) to report to the Congress by June 30, 2003, on regulatory efficiencies that would be gained by consolidating or eliminating regional offices. The NRC staff's report is enclosed.

As you will note in the report, the NRC has conducted several reviews of regional office effectiveness over the last two decades. Each of these reviews has led to changes in our regional structure, either through elimination and consolidation of our regional and field offices or a shift in functional responsibilities between the regions. In the latest review conducted in 2002, the Commission approved the consolidation of all regional fuel cycle facility inspection activities in our Region II Office in Atlanta and the transfer of Region II's materials licensing and inspection activities to Region I. Our objective was to maintain essential regulatory activities in the field while gaining efficiencies through functional realignment. As the chart on page 3 of the report indicates, NRC regional staffing levels, as expressed as a percentage of total NRC employees, have consistently been declining since 1992, reflecting efficiencies gained through consolidation and realignments as well as the assignment of additional responsibilities previously conducted by personnel in all of our regions to headquarters personnel.

The Commission believes that regulatory efficiencies would not be gained by consolidating or eliminating NRC's regional offices at this time. The regions provide direct, critical support to the accomplishment of the agency's health, safety, and security mission. Major changes beyond those already implemented by the Commission would disrupt NRC activities at a critical time for our Nation.

I hope this information is helpful to you. If you have questions, please contact me.

Sincerely,

/RA/

Nils J. Diaz

Enclosure:
Assessment of Efficiencies to be Gained by
Consolidating or Eliminating Regional Offices

cc: Representative Peter J. Visclosky

Identical letter sent to:

The Honorable David L. Hobson, Chairman
Subcommittee on Energy and Water Development
Committee on Appropriations
United States House of Representatives
Washington, D.C. 20515
cc: Representative Peter J. Visclosky

The Honorable Pete V. Domenici, Chairman
Subcommittee on Energy and Water Development
Committee on Appropriations
United States Senate
Washington, D.C. 20510
cc: Senator Harry Reid

Assessment of Efficiencies to Be Gained by
Consolidating or Eliminating Regional Offices

1. Background

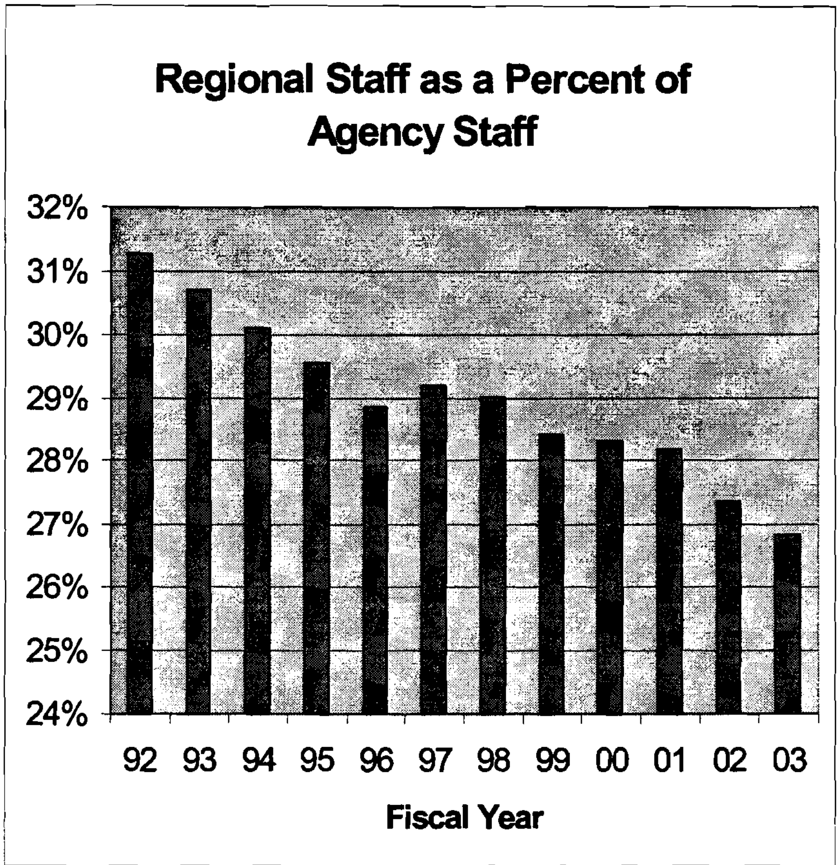
NRC maintains four regional offices: King of Prussia, Pennsylvania; Atlanta, Georgia; Lisle, Illinois; and Arlington, Texas. Approximately 850 agency employees are assigned to these offices. About 200 are resident inspector and support personnel stationed at reactor and fuel cycle facility sites; the rest are located in the four regional centers. Except for 28 employees stationed at our Technical Training Center in Chattanooga, Tennessee, the remainder of our 3000 employees are located at our Rockville, Maryland, headquarters.

NRC has periodically reviewed the workload and resources assigned to the regions, and taken several steps to improve their efficiency and effectiveness. In 1983, NRC established a field office in Denver, Colorado, to improve operations with regard to its responsibilities under the Uranium Mill Tailings Radiation Control Act of 1978. By 1994, the uranium recovery oversight program had matured to the point where the agency was able to eliminate the Denver field office and the infrastructure costs it entailed with no degradation in the uranium recovery program. Also in 1994, NRC reduced the number of regional offices from five to four, maintaining only a small field office in Walnut Creek, California. By 1998, the agency was in a position to maintain effective safety programs on the West coast without the necessity of a local field office. The field office was closed, with no adverse impact on public health and safety and some savings in overhead expenses.

In the 1994-1995 time frame, the agency conducted a thorough review of regional operations. A standard regional organization structure was developed to realign regional functions to be more consistent with those in headquarters program offices, to eliminate one layer of management, and to increase the span of managerial control. The new structure reduced the level of coordination needed between organizations, enhanced communications, and reduced the number of reviews and concurrences required to deliver regulatory services and products.

In 1998, the agency conducted another review of regional operations. The review considered consolidation and elimination scenarios, based on the most likely set of programmatic assumptions that could be made at that time. Cost projections made at the time indicated that the scenarios could generate some on-going savings. Implementation costs, however, would have been substantial. The estimated cost recovery time was approximately 7 years. Actual dollar amounts of costs and savings would be higher now, but given that the costs and savings would be similarly affected by inflation, cost recovery could be expected to continue to require 7-8 years. Adverse effects of the inevitable disruption to our program because of staff dislocations are less quantifiable but no less real. The most serious consequence would be the potential loss of highly skilled, difficult to replace staff.

Over time, these continuing efforts to align program activities and resource allocations to match changing workloads have resulted in a leaner regional operation. As the following chart indicates, the percent of NRC staff in the regions has steadily declined over the past several years.



2. Current Status

The Commission is continuing to explore ways to distribute workload and responsibilities that will improve operational efficiency and effectiveness in the regions. This effort is moving in the direction of specialized roles for some regions, stemming both from NRC regulatory developments and from fact-of-life changes in the external environment. For example, the Commission has recently approved the consolidation of fuel cycle facility inspection activities in Region II (Atlanta, Georgia), and the transfer of Region II materials licensing and inspection activities to Region I (King of Prussia, Pennsylvania). On one hand, most of the nation's current and planned fuel cycle facilities are already in, or in close proximity to, Region II. On the other, all but two of the states in Region II are Agreement States[1], and the two that are not, Virginia and West Virginia, are contiguous to Region I and much closer to the Region I office in King of Prussia than to Atlanta. This arrangement enables the agency to concentrate fuel cycle expertise in one region under one senior manager, and to eliminate the need for a materials licensing and inspection organization in Region II. Our expectation is that this will result in

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[1]Agreement States are states which, by agreement with, and in conformance with guidelines established by NRC, carry out materials licensing and inspection activities within their borders.

better staff development, more effective program implementation, and more efficient use of resources. Region IV (Arlington, Texas) has been equipped to serve as the alternate location for the agency's Emergency Response Center. The region is also expected to have specialized inspection and oversight responsibilities with regard to the Yucca Mountain high-level waste repository and related issues.

In the context of its fundamental mission, a robust regional presence is essential for the effective implementation of the agency's health, safety, and security programs. Public health and safety is better served with critical NRC expertise located close to the geographical area of our licensed activities. Whether overseeing routine licensed activities or reacting to unforseen circumstances, a regional office can rapidly muster critical resources to a facility when a situation needs immediate attention and time is of the essence. The NRC can facilitate a more rapid response from a regional office to a contingency or emergency event at a close-by licensed facility. If an event should occur at a licensed site, the regional response is equivalent to that of a first responder. In addition, the regional staff have unique expertise in the area of field inspections and are familiar with the licensee location, procedures, strengths and weaknesses. This knowledge has been obtained through years of inspections and interactions with the licensee. The four regional offices each oversee 21 to 33 operating reactors, which permits optimal usage of management and staff to carry out inspections and respond to events. This cadre of readily deployable first responders to incidents and emergencies in four different geographical locations is critical to sustaining a ready, reliable, and sufficiently redundant response capability.

The agency's regional structure provides an effective and efficient base for interaction and coordination with counterparts and stakeholders at the state level. Regional Administrators are in frequent contact with state officials, developing and maintaining relationships that promote effective communication and cooperation. This is particularly pertinent with regard to the 32 states that are Agreement States, with whom NRC shares responsibility for public health and safety, safeguards, and security in the utilization of nuclear materials. Radiation control programs, whether related to nuclear materials licensees or nuclear facility sites, require close and continuing coordination of state and Federal efforts by staff who have built relationships of mutual understanding and assistance with their counterparts.

The regional structure also aligns well with the Administration's emphasis on close coordination with constituents and stakeholders. Regional offices bring NRC closer to the public it serves, giving stakeholders access to NRC officials in their own region of the country, and sometimes in their own community. This is particularly beneficial to members of the general public, who are far less likely than utility executives and members of industry associations to come to NRC headquarters to participate in NRC activities. Through its regional offices and its resident inspectors, the NRC is not only a regulator but a neighbor in the nuclear community. This community concept builds public confidence and partnership. A regulator living in the area of a regulated facility is usually perceived as testament to its safe operations.

All the regional offices are involved in heightened security and safeguards activities in light of the current threat environment. In fact, homeland security initiatives and objectives are compelling reasons for the agency's current regional structure. In the event that NRC headquarters is disabled due to fire, natural disaster, or terrorist attack, the regions -- and Region IV in particular -- would play a significant role in continuing the mission of protecting the public health, safety, and security. In accordance with the October 1998, Presidential Decision Directive (PDD) 67 "Ensuring Constitutional Government and Continuity of Government Operations (COOP)," each executive branch agency was directed to prepare and maintain a plan for continuing its minimum essential functions at an alternate location, if necessary. The regions provide needed support to restore the ability of NRC to respond to security-related incidents within 3 hours, which meets the agency's goals. In the aftermath of 9/11 Region IV's role as the agency's alternate Emergency Response Center has been enhanced with the introduction of special secure communications capabilities. It is well situated for this; it is the closest regional office to the population center of the country, and it is connected to an electric power grid different from the power sources that support headquarters and the other regional offices.

3. Conclusion

In summary, program imperatives mandate maintenance of the agency's current regional structure. The four regional offices are integral and essential to achieving the agency's health, safety and security mission. They bring a critical dimension to – and are in many ways the heart of – the agency's safety culture. Regional personnel are usually the immediate deliverers of inspection services and the immediate responders in emergency situations. Their separation from headquarters fosters a sustained focus on, and vigilant commitment to, day-to-day operational safety. The regional office structure provides essential and highly effective support, guidance, and supervision both to the regional office-based inspectors and to the cadre of resident inspectors within the regional boundaries. Certainly, the costs of the regional offices should be kept as low as possible, and every opportunity to reduce them should be considered. But in the long run, their value far outweighs their costs. The regions epitomize the bottom line of NRC's commitment to public health and safety.

Gilena Monroe - Bio

Miss Gilena Monroe joined the ACRS staff on June 16th as a summer intern. Currently, Gilena is a full-time Graduate student attending North Carolina A&T State University. She has a B.S. degree in Computer Science and is presently majoring in Industrial and Systems Engineering with a concentration in Human-Machine Systems/Human Factors. She is working with the ACRS as a Student Engineer on topics of Human Factors Engineering and Human Reliability.

Marvin Sykes comes to the ACRS from NRR's Division of Inspection Program Management where he monitored regional implementation of the ROP's significance determination process (SDP) and recommended strategies for improving the effectiveness of the process. Marvin joined the NRC in 1991 as an NRR Reactor Engineer Intern. He later transferred to Region II where he completed Operator Licensing Examiner and Resident Operations Inspector certification and completed assignments at the Grand Gulf(BWR) and McGuire(PWR-ICE) Nuclear Stations and within the regional office. Before joining the NRC, Marvin worked as a Technical Support Engineer with the Southern Nuclear Operating Company Farley Project.

Mr. Sykes received a Bachelor of Science in Physics from Alabama A&M University.

Dr. Bhagwat P. Jain

Bio-Data

Dr. Jain will be joining the ACRS staff as a Senior Staff Engineer on July 14, 2003. He has been with the NRC for 5 years. During this period, he has served in NRR (Division of Engineering), NSIR, and RES (Division of Engineering Technology). Currently he is a project manager in RES. Prior to joining the NRC, Dr. Jain worked with Carolina Power & Light Company as a project engineer at the Brunswick nuclear power plant, AES Corporation as a supervising structural/mechanical engineer at the Prairie Island nuclear power plant, and with Sargent & Lundy Engineers, Chicago, as a supervising structural engineering specialist. He is a registered professional engineer with a Ph.D. in structural engineering which he received in 1976 from Illinois Institute of Technology, Chicago. In addition, Dr. Jain has experience in the evaluation of nuclear plant safety issues relating to plant structures, systems, and components of PWR and BWR containments. He is experienced in the structural design and licensing of several nuclear power plants (e.g., Lasalle, Zimmer, Byron, Braidwood). Dr. Jain has a broad range of over 25 years experience in progressively responsible positions in private nuclear power industry with design consultants and power utility companies, and the NRC.

MIXED OXIDE FUEL FABRICATION FACILITY

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

July 10, 2003

2

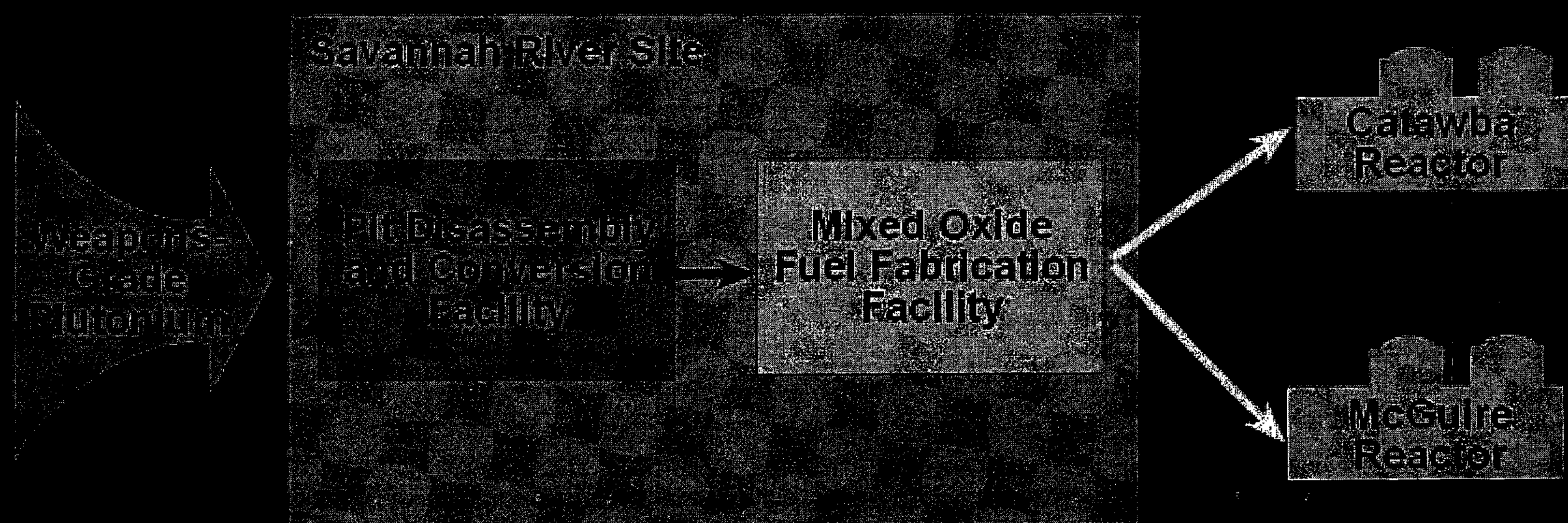
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Review of the Mixed Oxide Fuel Fabrication Facility
Construction Authorization Request

Introduction

Andrew Persinko, Sr. Project Manager
NMSS/FCSS/SPIB

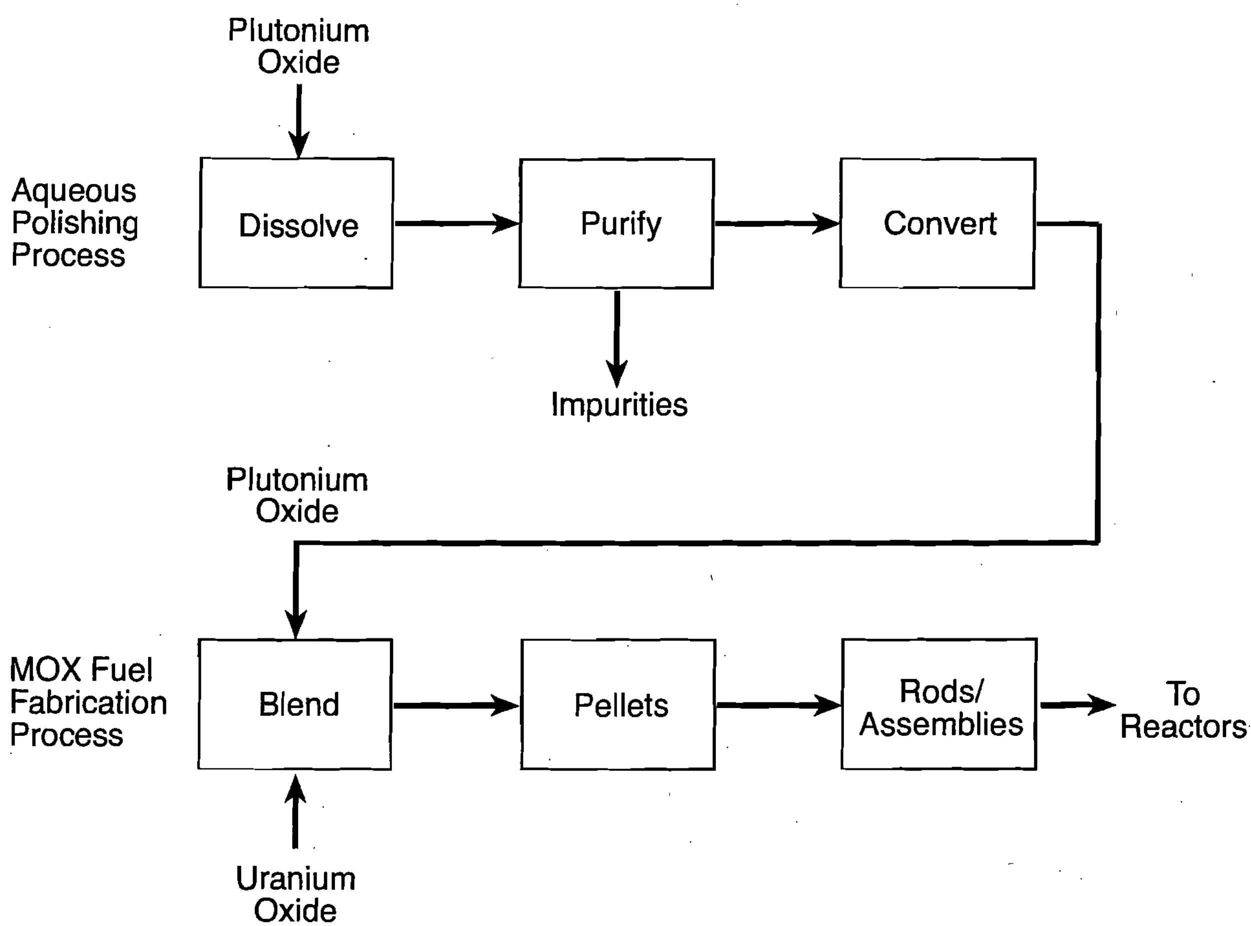
NRC Role in Regulating Mixed Oxide Fuel



Yellow = NRC regulated

Blue = DOE regulated

Mixed Oxide Fuel Fabrication Facility Process



Mixed Oxide Fuel Fabrication Facility

Licensing (10 CFR Part 70)

- 2-step approval:
  - Construction
  - Operation/possession of special nuclear material
- Approvals to start construction plutonium facility
  - <u>Design bases</u> of principal structures, systems, and components (PSSCs)
  - Quality assurance program
  - Environmental impact statement
- Principal structures, systems, and components / Items relied on for safety

Construction

Design Bases

- 10 CFR 50.2 Definition:

"Design Bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design..."

10 CFR 70.61 Performance Requirements

	Highly Unlikely	Unlikely	Not unlikely
High Consequence Publ Dose > 25 rem Worker Dose > 100 rem	Acceptable	Not Acceptable	Not Acceptable
Medium Consequence Publ Dose 5 - 25 rem Worker Dose 25 -100 rem Env releases > 5000 Tbl 2	Acceptable	Acceptable	Not Acceptable
Low Consequence Publ Dose < 5 rem Worker Dose < 25 rem	Acceptable	Acceptable	Acceptable

Schedule

Major Milestones

- Received Environmental Report 12/19/00
- Received Construction Authorization Request (CAR) 2/28/01
- Issued draft Safety Evaluation Report (SER) for construction 4/30/02
- Received revised Environmental Report 7/11/02
- Received revised CAR 10/31/02

Schedule

Major Milestones

- Issued draft Environmental Impact Statement (EIS) for public comment 2/28/03
- Issued revised draft SER for construction 4/30/03
- Issue final EIS and final SER 9/03
- Issue EIS Record of Decision (ROD) and construction licensing decision 10/03

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Nuclear Criticality Safety Review for the Mixed Oxide Fuel Fabrication Facility Construction Authorization Request

Christopher S. Tripp, Sr. Nuclear Process Engineer (Criticality)
NMSS/FCSS/SPIB

NCS Design Bases

- Double Contingency Principle (DCP)
- Maximum k-effective for criticality calculations
- Subcritical under normal and abnormal conditions
- Dominant parameters for major processes
- Preference for engineered over administrative control
- Criticality accident alarm system
- Management measures
- Organization and administration
- Technical practices (including ANSI standards)
- Balance of fire protection and criticality risk

Controlled Parameters (Waste Storage)

- Waste to be stored at MFFF
- Waste to be processed under DOE jurisdiction
- Control strategy:
  - Dual controls on concentration/mass
  - Adherence to DCP
  - May consist of active and passive engineered means, dual sampling
- Consistent with usual industry practice for auxiliary systems (e.g., ventilation, acid/solvent recovery)

NCS Open Issue (NCS-4)

- K-effective limits for 5 different AOAs:
  - Pu nitrate solutions
  - MOX pellets, rods, assemblies
  - PuO_2 powder
  - MOX powder
  - Pu compounds

- Methodology for normal condition minimum subcritical margin (abnormal = 0.05)

NCS Open Issue (NCS-4)

Criticality Code Validation

- Few critical benchmarks with required absorbers, range of parameters
- Use of sensitivity/uncertainty methods (SCALE 5) for powder systems
- Lumping all benchmarks into same AOA
- Rigor of methods used to demonstrate benchmark applicability

NCS Open Issue (NCS-4)

Criticality Code Validation

- Received Validation Report January 2003
- Meeting on major issues March 2003
- Received SCALE 5 (sensitivity/uncertainty code) May 2003
- Issued RAI June 2003
- Performing independent sensitivity/uncertainty analysis for solution, powder, and compounds

Conclusions

- Design bases acceptable except k-effective limits
  - Validation of AOAs
  - Normal case subcritical margin
- Identified early as main technical challenge for NCS
- Staff reviewing validation report
- SCALE-5 code being used

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Fire Protection Review of the Mixed Oxide Fuel Fabrication Facility Construction Authorization Request

Rex Wescott, Integrated Safety Analysis Specialist
NMSS/FCSS/SPIB

Fire Protection

OVERALL DESIGN BASES: Assure that the 10 CFR 70.61 Performance Requirements are complied with under all credible fire scenarios

- Prevention
  - AP process cells
  - Inerted gloveboxes
- Suppression and/or combustible loading controls
  - Truck bays
  - Secured warehouse
  - Glovebox areas (clean agent suppression)
  - Fuel rod and canister storage areas
- Fire barriers
  - Confinement of internal fires to one fire area
  - Protection against external fires

Fire Design Basis Values

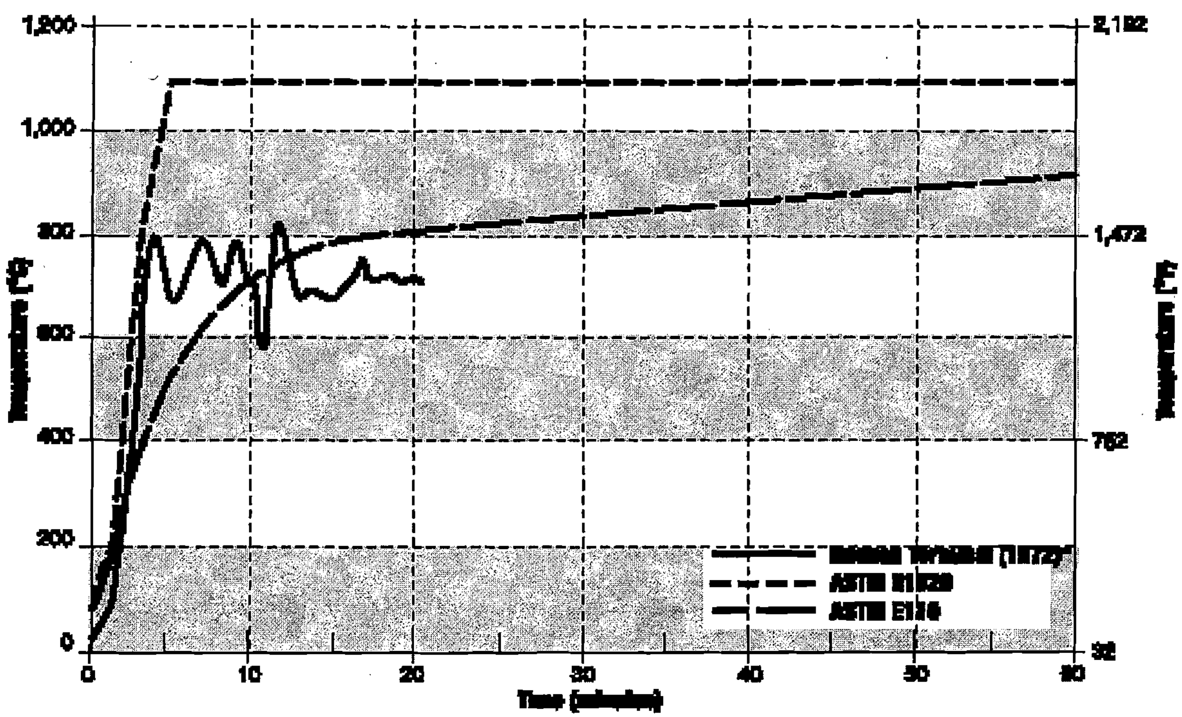
- Compartment or fire area boundaries -- 2 hour fire as per ASTM E-119
- Compartment air temperature into ventilation system -- 2000°F to protect final HEPA filters
- Material confinement barriers --
  - 3013 transport cask -- 1472°F for 30 minutes
  - MOX fuel transport cask -- 1472°F for 30 minutes

Open Issue: Fire barriers

- Applicant has evaluated fire scenarios where temperatures could exceed the ASTM E-119 curve (reagent storage area)

- It must be demonstrated that fire barriers can withstand the rapid fire development without loss of integrity

Comparison of standard fire test curves with office fire experiment



*Hudson Terminal experiment conducted with normal office fuel load (8 psf) (DeCicco, et al. 1972)

Figure A-3 Comparison of exposure temperatures in standard tests.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Tributyl-Phosphate (TBP) -Nitrate
(Red-Oil) Review for the Mixed Oxide Fuel Fabrication
Facility Construction Authorization Request

Alexander Murray, Sr. Chemical Process Engineer
NMSS/FCSS/SPIB

Description of Tributyl-Phosphate (TBP) -Nitrate (Red Oil)

- Chemical reaction of TBP/organics and nitric acid/nitrates
- Reaction can runaway - generate thermal energy and non-condensable gases
- Reaction rate is a function of chemical species, concentrations, temperature, and pressure; radiolysis can contribute to the phenomena
- Impurities and intermediates exacerbate the phenomena
- Red oil reactions are explosive under certain conditions

Background

- There are 4 reported accidents with equipment damage and facility release
- There is 1 accident with offsite release (Tomsk)
- Literature implies several "incidents" have occurred
- DCS has recognized red oil as an explosion event and has proposed a prevention strategy

Applicant's Approach

- 3 PSSCs, 5 safety functions, and distinction of open versus closed systems
  - Offgas: vent path (open) or evaporative cooling (closed)
  - Process Safety Control Subsystem: Steam temperature and residence time
  - Chemical Safety Control (Administrative): diluent selection and degraded organic compound quantity limit

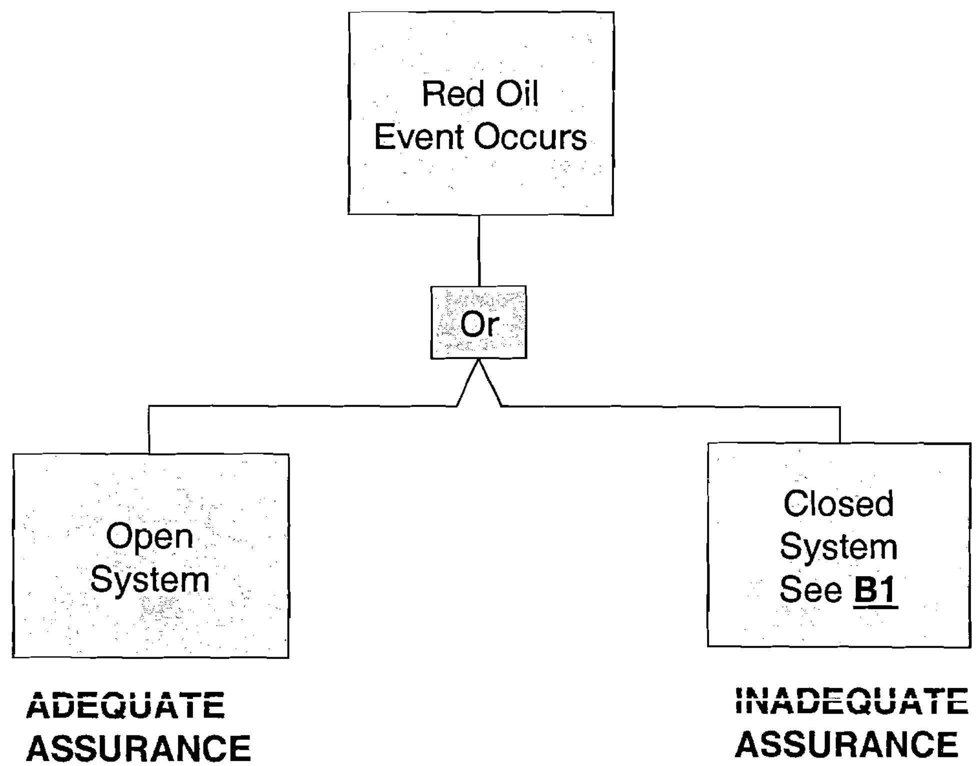
Open and Closed Systems

- Open system: capable of venting the full runaway reaction based upon experimental results (SRS), safety factor of 2.5, and assumption of 100% organics in system

- Closed system: Mass transfer to assure evaporative cooling at nitric acid/water azeotrope
  - vessel can have significant organics but not 100%
  - Safety factor 1.2 x [energy input + energy generation] with steam limited to 133°C
  - incapable of venting full runaway reaction (system would pressurize)

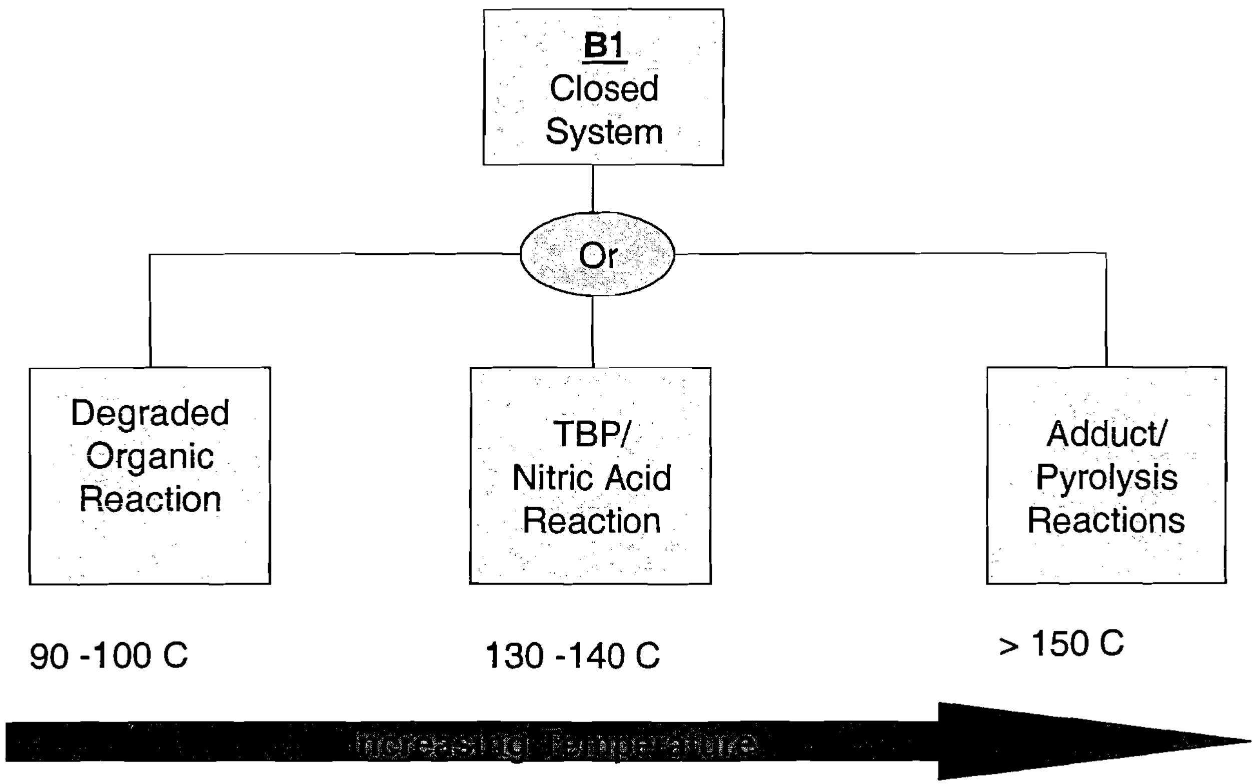
Top Level Fault Tree: Open Systems vs. Closed Systems

Initial Tree



Closed System Fault Tree - Top

FTA – Second Level



Conclusions

- Open system: approach capable of meeting "highly unlikely" and is acceptable
  - applies to most vessels (unheated)
- Closed system: approach not currently accepted by staff
  - likelihood of potential event not "highly unlikely"
  - some differences with approaches at existing facilities
  - limit solution temperature susceptibility to increases (steam pressure/temperature fluctuations, degraded/loss of venting capability)
- DCS to provide additional information to the NRC

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Seismic Review of the Mixed Oxide Fuel Fabrication Facility Construction Authorization Request

John Stamatakos
Rex Wescott
Herman Graves, III

DCS Approach to Seismic Design

- Seismic Design Based on Four-Part Seismic Hazard Assessment
  - Generic probabilistic seismic hazard assessment (PSHA) for the Savannah River Site.
    - Based on Lawrence Livermore National Laboratory (LLNL) and Electric Power Research Institute (EPRI) sesimic hazard studies for Central and Eastern United States.
    - DCS established design basis earthquake by implemented DOE Standard 1023 (parallels methodology in NRC Regulatory Guide 1.165).
    - Design basis earthquake based on DOE performance categories defined in DOE Standard 1020 with PC-3 and PC-4 (mean hazards at 5×10^{-4} and 1×10^{-4} annual exceedence probabilities).

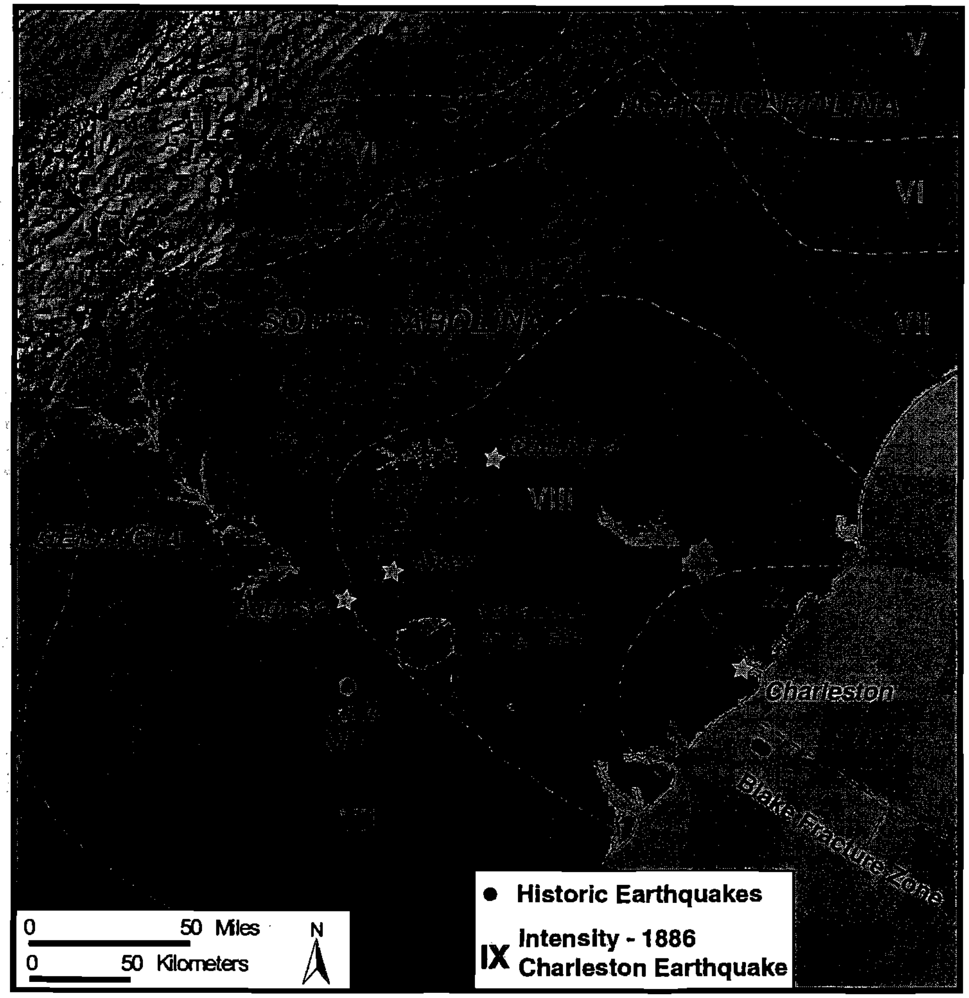
DCS Approach to Seismic Design (Continued)

- Facility design based on Regulatory Guide 1.60 spectra scaled to 0.20 g peak ground acceleration (Plant Vogtle NPP).
  - Target 1×10^{-4} mean annual exceedence probability for ground motions at frequencies of interest.
  - Design uses Regulatory Guide 1.60 horizontal soil surface spectrum scaled to 0.20 peak ground acceleration to meet this goal.
  - Vertical spectrum is also based on NRC Regulatory Guide 1.60 scaled to 0.20g peak ground acceleration.

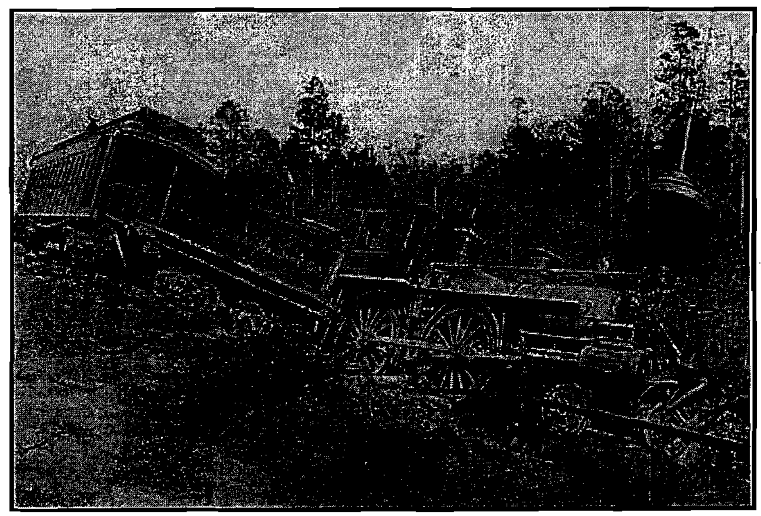
DCS Approach to Seismic Design (Continued)

- Soil stability analyses based on LLNL and EPRI seismic hazard results adjusted for site response.
  - Mean soil amplification factors were developed from site response model to scale the bedrock uniform hazard spectra to the soil surface.
  - DCS also developed alternative site-specific amplification functions (bedrock to soil surface) to validate response model.
  - For soil stability analyses, DCS used the bedrock PC-3 ground motions scaled so that when amplified throught the soil they produce surface ground motions with 0.20 peak ground acceleration.

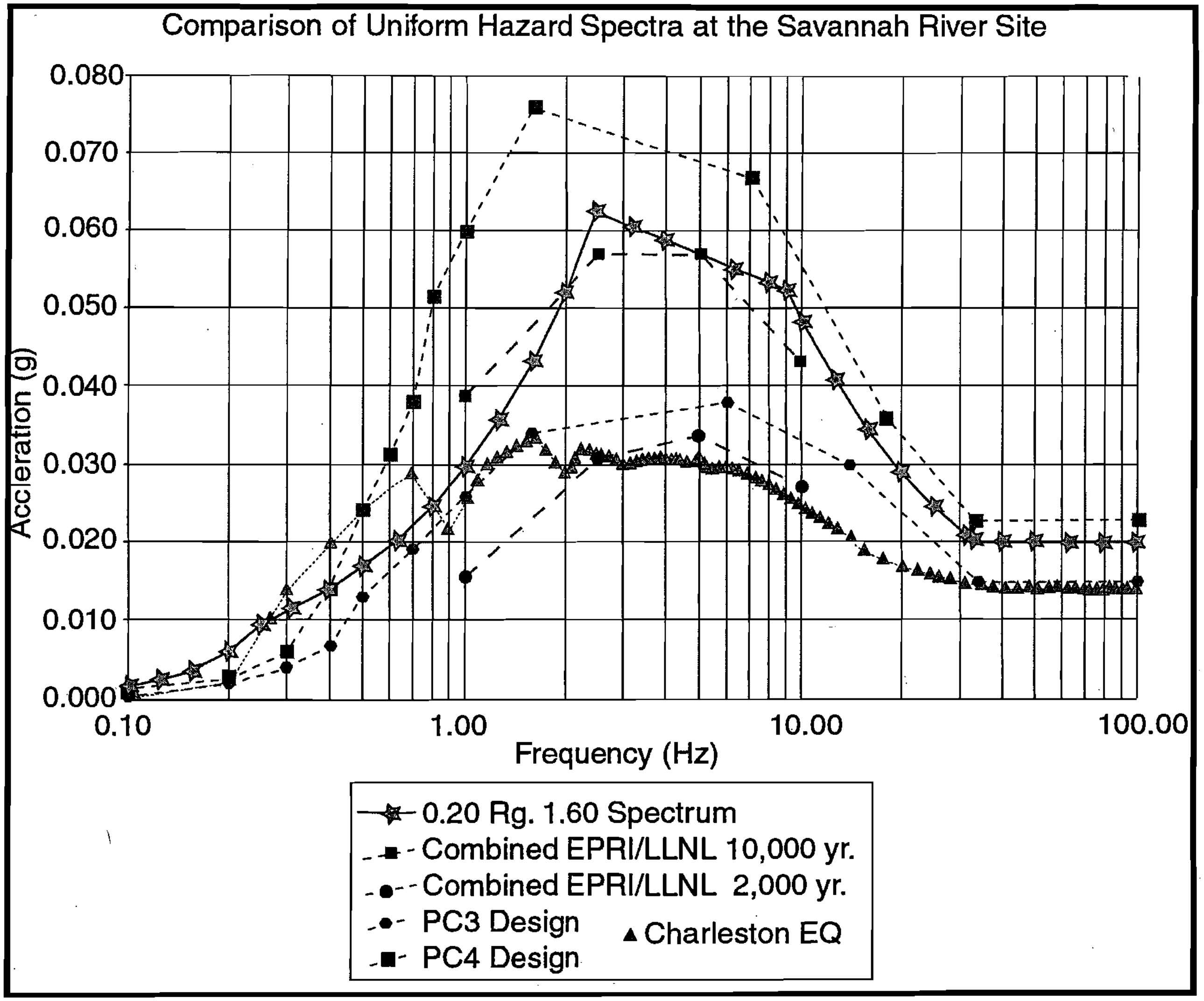
DCS Approach to Seismic Design (Continued)



- "Historic Check" using repeat of the 1886 Charleston Earthquake.
  - Magnitude =7.3 at Distance of 120 km from the site.



Summary of MOX Seismic Hazard and Design Spectra



Confirmatory Risk Assessment

- DCS performed limited probabilistic risk assessment on representative systems, structures, and components (e.g., offsite power, glove box, building structure).

- DCS showed seismic performance of these systems, structures, and components is 1×10^{-5}/yr or better, consistent with guidance in NUREG 1718.

Staff Evalaution

- Bedrock Seismic Hazard
  - Application of LLBL and EPRI hazard results is appropriate.
- Site reposnse
  - Site response models, based on site-specific soil data, are adequate.
- Sesimic Design
  - Regulatory Guide 1.60 spectra scaled to 0.20 g peak ground acceleration envlopes uniform hazard spectrum and Charleston Earthquake "historic check" at frequencies of interest.
  - Probabilistic risk assessment shows that critical systems, structures, and components consistent with performnce objectives in NUREG-1718.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Safety Assessment:
Radiological Consequences

Rex Wescott, Integrated Safety Analysis Specialist
NMSS/FCSS/SPIB

Overview

Safety Assessment: Radiological Consequences

- The staff's review of the applicant's radiological consequence calculations included review of:
  - Source term calculations
  - Facility worker dose estimates
  - Downwind consequence calculations
  - How the applicant's safety strategy reduces the risk to each receptor.

Safety Assessment: Radiological Consequences

Source Term using Five Factor Formula

- The staff reviewed:
  - material-at-risk (MAR)
  - damage ratio (DR)
  - atmospheric release fractions (ARFs)
  - respirable fractions (RFs)
  - leak path factors (LPFs) (i.e., HEPA filters)

Safety Assessment: Radiological Consequences

Source Terms: Applicant's proposed methodology

- ARFs and RFs were assigned for each type of material and each event.
- Material release forms
  - Solution
  - Powder
  - Pellet
  - Rod
  - Unencased filter
- Events
  - Explosive detonation
  - Explosive overpressurization
  - Fire/boil
  - Drop
  - Entrainment

Safety Assessment: Radiological Consequences

Source Terms: Applicant's proposed methodology (cont.)

- For example:
  - Powder, fire: ARF x RF = $6 \times 10^{-3} \times 0.1 = 6 \times 10^{-4}$
  - Solution, explosion = $1.0 \times 0.01 = 10^{-2}$
  - Rods, dropped = $3 \times 10^{-5} \times 1.0 = 3 \times 10^{-5}$
    - Source: Table 9.1-5 of the April 30, 2003 DSER

Safety Assessment: Radiological Consequences

Source Terms: Staff's Evaluation

- DSER, Section 9.1.1.4.2:
  - Staff finds the values chosen by the applicant to be consistent with recommendations in NUREG/CR-6410, and find them acceptable for construction authorization.
- Staff will review all 5 factors again during review of the Integrated Safety Analysis

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS PRESENTATION

Review of the Mixed Oxide Fuel Fabrication Facility
Construction Authorization Request

Remaining Open Items

Andrew Persinko, Sr. Project Manager
NMSS/FCSS/SPIB

Remaining Open Items

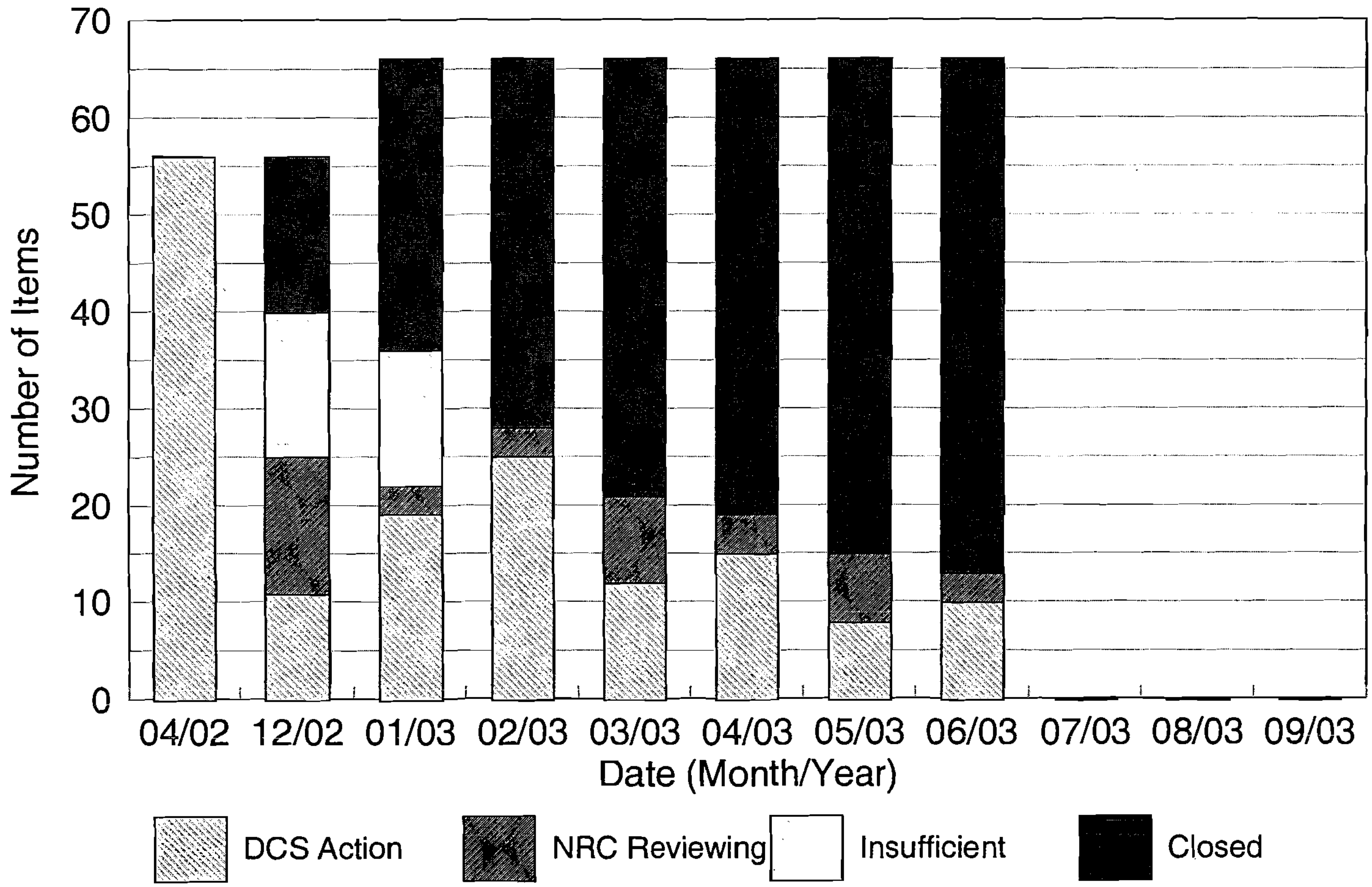
- Titanium fires (AP-3)
- UO_2 burnback (MP-1) Hydroxylamine nitrate (HAN)/hydrazine (CS-2)
- Design basis for flammable gases: lower flammabilty limit values 25% vs 50%
  - Solvent temperatures (CS-9, AP-9)
  - Electrolyzer (AP-2)
  - Offgas unit (AP-8)

Remaining Open Items

- Emergency control room habitability limits (CS-10)
- Use of Temporary Emergency Exposure Limits (TEELs) (CS-5b)

DSER Open Items

June 2003



DUKE COGEMA
STONE & WEBSTER

MFFF
General Facility Mission and Layout

Presentation to the
504th Advisory Committee on Reactor Safeguards
10 July 2003

Proposed Criteria for the Treatment of Individual Requirements in a Regulatory Analysis

Briefing to the ACRS

Brian J. Richter
Reactor Policy & Rulemaking Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
July 10, 2003

Objective of Presentation

- Obtain ACRS consent on the staff's approach to the treatment of the "bundling" issue.
  - Given the proposed approach is improved over existing guidance, obtaining ACRS consent would allow implementation, via the issuance of the Regulatory Analysis Guidelines, Revision 4, to occur in a timely fashion.
- Obtain comments from ACRS to improve the proposed approach and, if needed, adapt additional input for a new revision (5) to the Guidelines.

Issue

- The Commission and others have raised the issue of the “bundling” of individual requirements into a single regulatory analysis.

- The concern is that the net benefit from one regulatory requirement could potentially support other requirements that are not cost-justified.

Background

- SECY-00-0198, "Status Report on Study of Risk-informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)"
  - Staff recommended that the Commission not allow selective implementation of voluntary rules, as was suggested by some licensees.
  - Commission agreed, but challenged staff to establish a process solution.

Background (cont.)

- SECY-01-0134, “Final Rule Amending the Fitness-for-Duty Rule”
  - Stakeholders raised concerns with the NRC approach that “bundles” individual requirements into a single analysis.
  - SRM directed the staff to ensure that individual rule changes are integral to the purpose of the rule, cost-justified, or qualify as backfit exceptions.

Background (cont.)

- SECY-01-0162, "Staff Plans for Proceeding with the Risk-Informed Alternative to the Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors in 10 CFR 50.44"
  - Staff proposed to develop guidance to address how to assess new risk-informed requirements.
  - Commission agreed and directed the staff to ". . . [I]mplement a disciplined, meaningful, and scrutable methodology for evaluating the value-impact of any new requirements that could be added by a risk-informed alternative rule."

Staff Activity

- Working Group formed containing members from NRR, NMSS, RES, and OGC.
- Proposed revision to Regulatory Analysis Guidelines, NUREG/BR-0058, Rev. 3 to address the treatment of individual requirements.
- Published FRN February 13, 2002, which presented preliminary proposed criteria for the treatment of individual requirements in a regulatory analysis, and announced public meeting.
- Held public meeting on March 21, 2002, to discuss preliminary proposed criteria.
- Extensive comments were received at the meeting..
- Developed proposed guidance considering comments received.

Staff Activity (cont.)

- Obtained CRGR endorsement of proposed guidance.
- Staff obtained approval from the Commission to publish these proposed revisions to NUREG/BR-0058 for public comment in the *Federal Register*.
- SRM said staff should provide the final criteria to the Commission for review not only if the comments result in significant changes to the criteria, but also if there are significant adverse comments regarding the criteria.
- A request for comment on the proposed criteria was published in the *Federal Register* on April 18, 2003.

Published Proposed Criteria

- The proposed criteria were:
  - If an individual requirement is "necessary" (i.e., it is needed in order to meet the objectives of the rule or maintain consistency with Commission policies), it does not need to be analyzed separately.
  - If an individual requirement is supportive but not necessary, it should be included only if it makes the bundled initiative more cost-beneficial.

Published Proposed Criteria (cont.)

- If an individual requirement is unrelated to the overall regulatory action, it should be included only if it makes the bundled requirements more cost-beneficial *and* it passes the backfit test, if applicable.

- Disaggregation is only appropriate if it produces substantively different alternatives with potentially meaningful implications on the cost-benefit results.

- If an individual requirement in a voluntary rule is justifiable under backfit criteria, NRC should consider imposing this as a mandatory backfit.

Comments Received

- NEI made the only submittal of comments
  - Criteria necessary to evaluate the bundling of individual requirements into a single regulatory analysis.
    - Risk-informed voluntary alternatives should be cost-justified <u>and</u> integral, not cost-justified <u>or</u> integral.
    - Lack of scrutable guidance by the NRC.
  - Use of subjective judgment in making bundling decisions.
  - Request another public meeting on the issue.

Future Actions

- Working group to meet to resolve NEI comments and requests.
- Management is considering whether to hold another public meeting.
- Draft revised Guidelines input and submit to the Commission.
- Issue NUREG/BR-0058, Rev. 4.

5

ESBWR Pre-Application Review

Amy Cubbage, Project Manager
New Reactor Licensing Projects, NRR

ACRS
July 10, 2003

1

ESBWR Pre-Application Scope

- TRACG Application for ESBWR LOCA and Containment Analyses
- TRACG Qualification
- Test and Analysis Program Description (TAPD) and PIRT
- SBWR and ESBWR Testing Reports
- ESBWR Scaling Report

2

6

GE Nuclear Energy

ESBWR Design Overview and Technology Closure

Atam Rao
Project Manager ESBWR
GE Nuclear Energy, USA

ACRS Meeting
July 10, 2003,
Rockville, Maryland

Outline

- ESBWR Program Overview
  - stepwise program with Technology Closure as first step
- ESBWR Design Overview
  - simpler with more margin – by design
  - comprehensive testing and analysis program
- Technology program
  - comprehensive plan with complete implementation
  - testing and qualification
  - single integrated computer code for analysis with well defined application methodology
- Technology Closure Program – pre-application review
  - safety evaluation report for TRACG
- Summary and Conclusions

A simple design, extensive testing and analysis – Implementing an action plan to minimize regulatory risk

ESBWR Program overview

- Stepwise program for design development
  - Developed passive systems
  - Developed integrated plant design – SBWR
    - Completed extensive system and building design
    - Defined extensive test and analysis program
    - Completed extensive test and analysis program
  - Improved plant economics and design
    - Plant optimization and economies of scale
    - Incorporated utility requirements
    - Utilize ABWR experience – components, construction
- Stepwise program for regulatory approval
  - Simpler with more margin – by design
  - Technology Closure Program – pre-application review
    - safety evaluation report for TRACG
    - focus on safety systems & containment
    - use of single integrated computer code for analysis
    - comprehensive testing and analysis basis
  - Safety analysis report & design certification – after technology closure

GE is committed to develop and license the ESBWR

Goals for the technology closure

- Approval of the use of TRACG for analysis
  - vessel response to pipe break – loss of coolant accident (called ECCS/LOCA)
  - containment response to pipe break (called Containment/LOCA)
  - vessel response to anticipated operational occurrences (called AOO) - delayed
  - plant response to anticipated transients without scram (ATWS) and normal operation stability (ODYSY) – LATER
- Confirmation of the adequacy of TRACG
  - adequacy of the qualification base and approach

Is a 15+ year comprehensive technology program enough?

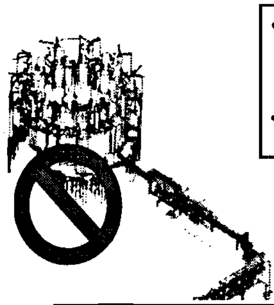
Comparison of Key ESBWR Parameters to Operating BWRs

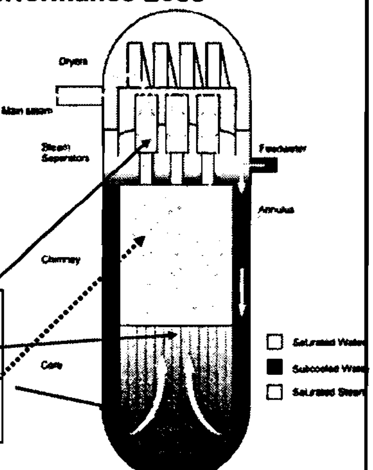
Parameter	BWR/4-Mk I (Browns Ferry 3)	BWR/6-Mk III (Grand Gulf)	ABWR	ESBWR
Power (MWt/MWe)	3293/1098	3900/1360	3926/1350	4000/1390
Vessel height/dia. (m)	21.9/6.4	21.8/6.4	21.1/7.1	27.7/7.1
Fuel Bundles (number)	764	800	872	1020
Active Fuel Height (m)	3.7	3.7	3.7	3.0
Power density (kw/l)	50	54.2	51	54
Recirculation pumps	2(large)	2(large)	10	zero
Number of CRDs/type	185/LP	193/LP	205/FM	121/FM
Safety system pumps	9	9	18	zero
Safety diesel generator	2	3	3	zero
Core damage freq./yr	1E-5	1E-6	1E-7	1E-7
Safety Bldg Vol (m^3/MWe)	115	150	160	70

Evolution Within a Small Range Minimizes Operational Risks

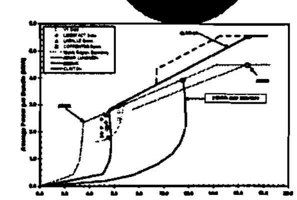
Natural Circulation – Simplification Without Performance Loss

- **Passive safety/natural circulation**
  - Put the water in the vessel – larger vessel
  - Increase driving head – larger vessel
- **Significant reduction in components**
  - Pumps, motors, controls, HXers
- **Load following with Control Rod Drives**
  - Minimal impact on maintenance



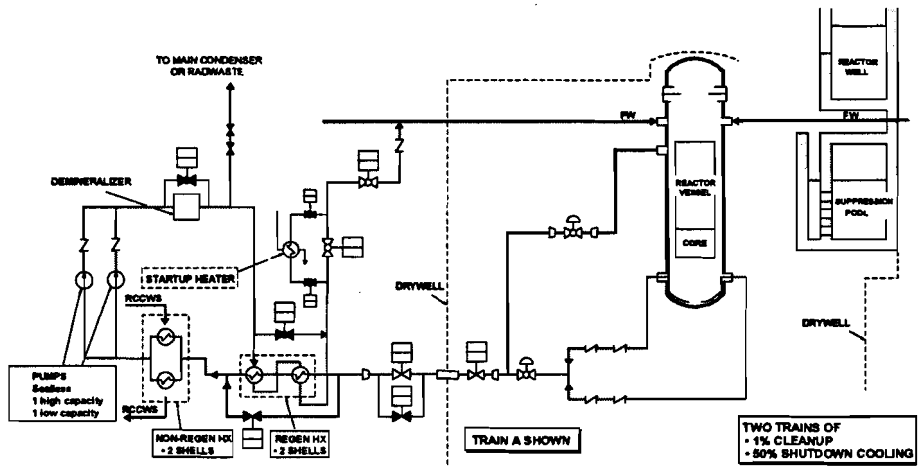


- Reduced flow restrictions
  - Improved separators
  - Shorter core
  - Increase downcomer area
- Higher driving head
  - Chimney/taller vessel



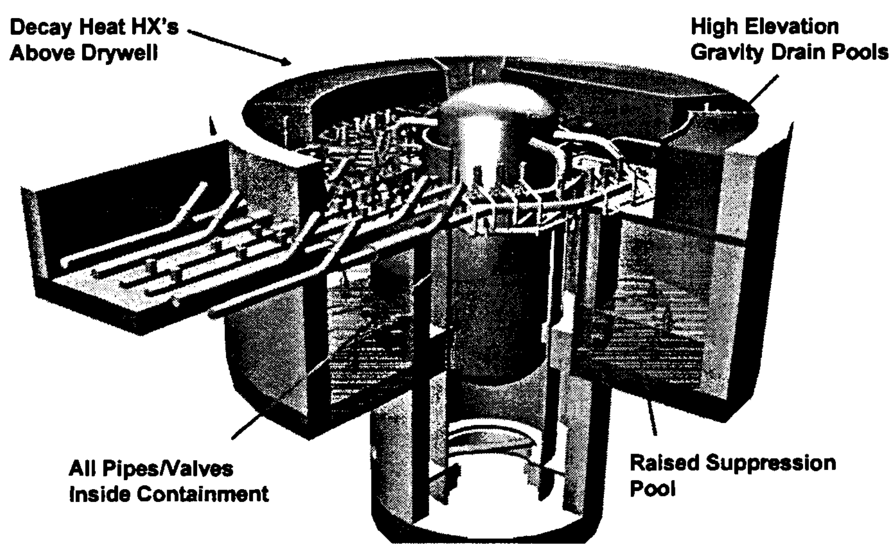
Enhanced Natural Circulation Compared to Standard BWR's

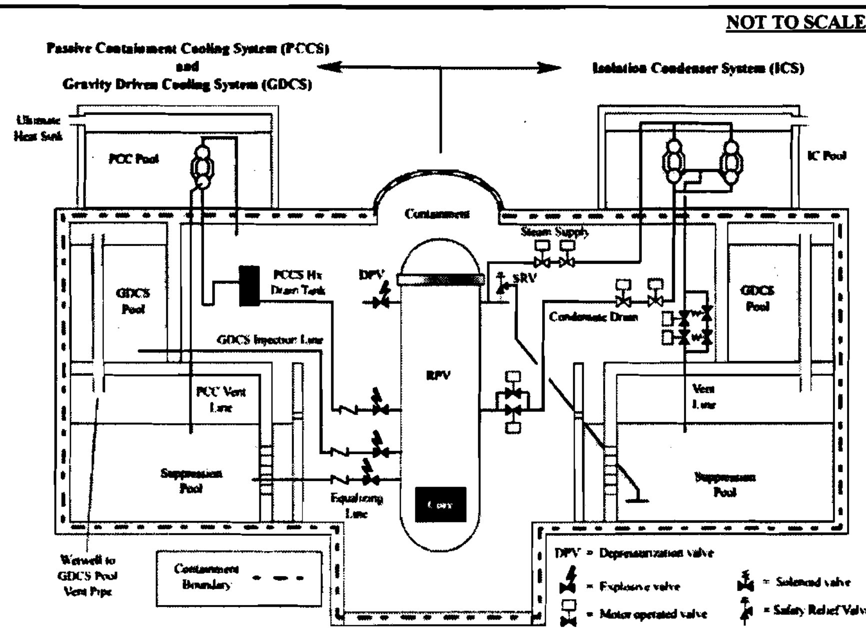
Reactor Water Cleanup / Shutdown Cooling System
(both systems combined for ESBWR)



Standard 2 x1% reactor water cleanup system with Unique full pressure shutdown cooling system

Passive Safety Systems Within Containment Envelope





Passive systems simplify the plant, reduce maintenance & increase security

AR-9

Design philosophy for core cooling

- **Increase inventory in the vessel**
  - Use taller vessel - NEW
  - Increase amount of subcooled water - NEW
- **Minimize inventory loss from the vessel**
  - Eliminate large pipes below the core and minimize other pipes - NEW
- **Keep core covered after initial blowdown**
  - Shorter core lower in the vessel - - NEW
- **Provide inventory makeup – low head using gravity**
  - Provide diverse depressurization system for high reliability - NEW
  - Required makeup rate is very low
    - Multiple pools rely on gravity to fill vessel - NEW
  - No high capacity systems needed
  - Fewer systems interactions
- **Utilize integrated BWR analyses tools**

Design features improved the plant response

AR-10

Design Philosophy for decay heat removal

- **Remove Decay Heat From Vessel**
  - Main Condenser
  - Full pressure normal shutdown cooling system - NEW
  - Isolation condensers - NEW
  - Remove vessel heat through relief valve opening
- **If Needed, Remove Heat From Containment**
  - Passive containment cooling (PCC) Hx (safety-grade) - NEW
    - Always available and drywell/wetwell pressure difference removes the non-condensables from the heat exchangers
    - Condensed steam returns to drywell/vessel, non-condensables collect in the wetwell airspace
    - No operator action needed for 72 hours
  - Suppression pool cooling (non-safety)

Several Diverse Means of Decay Heat Removal

AR-11

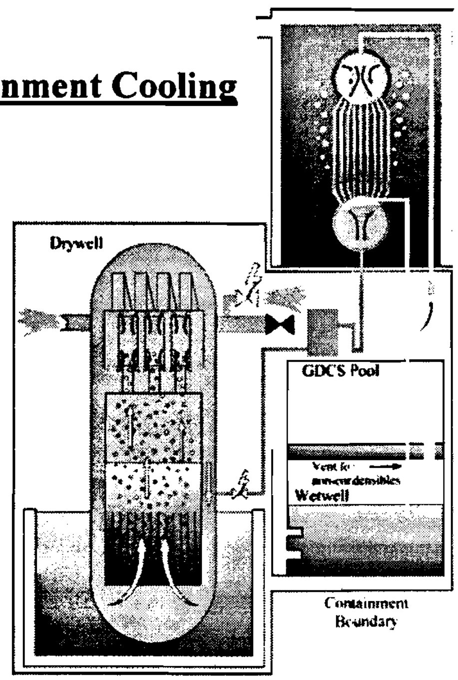
Decay Heat Removal from Containment - How it works

- **Initially steam (blowdown energy) flows to large heat sink in containment (suppression pool) and through heat exchangers**
- **Longer term (decay heat) steam flows to heat exchanger (similar to an isolation condenser) and heat is transferred outside containment**
  - vertical tube heat exchangers in a pool of water
  - non condensables removed by pressure difference between drywell and wetwell
- **Containment pressure determined by non-condensables in wetwell airspace and vapor pressure**

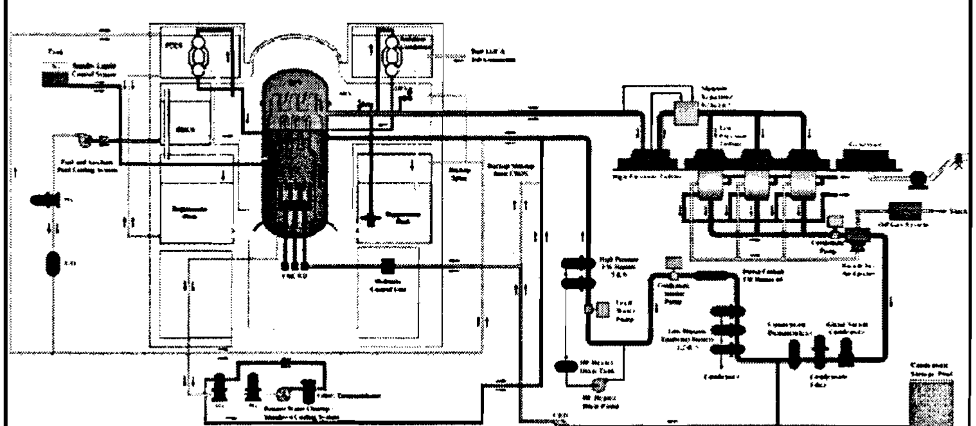
Concept is simple, reliable - extensive testing and analysis provide high confidence in the design margin

AR-12

Passive Containment Cooling

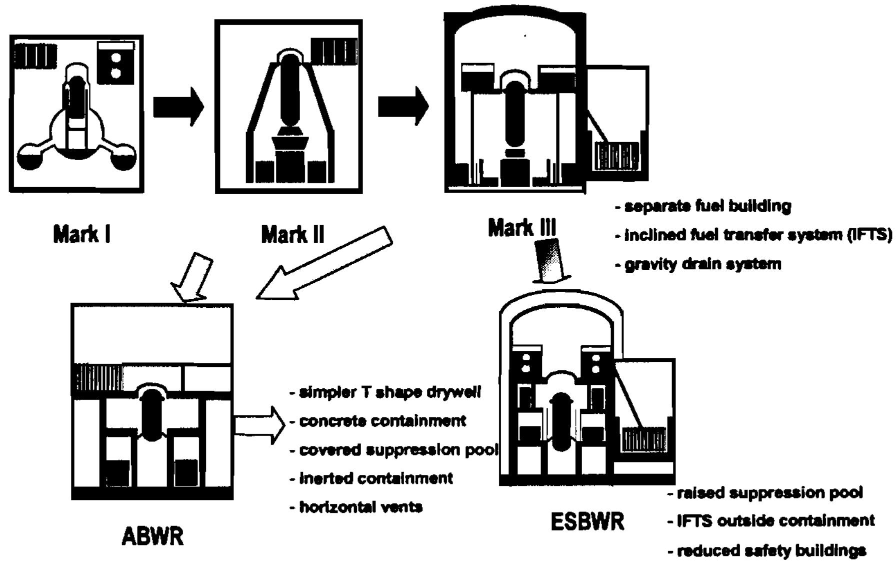


ESBWR Plant Schematic

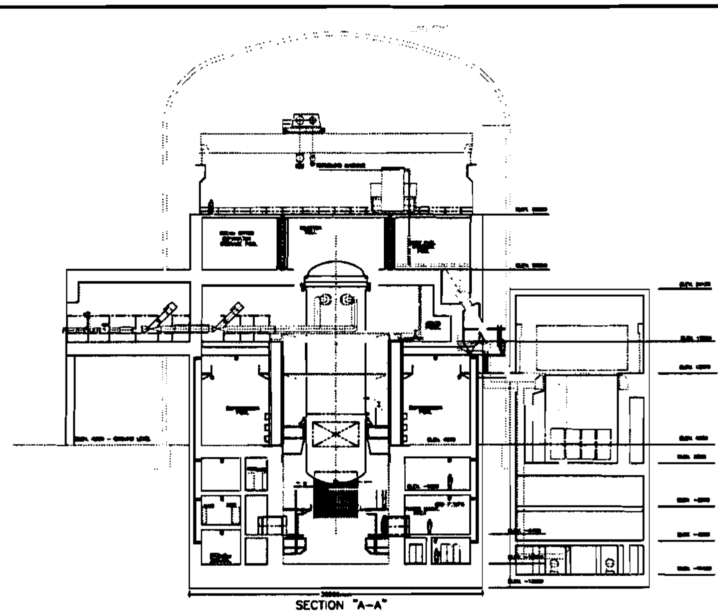


A Typical but Simpler Direct Cycle Plant

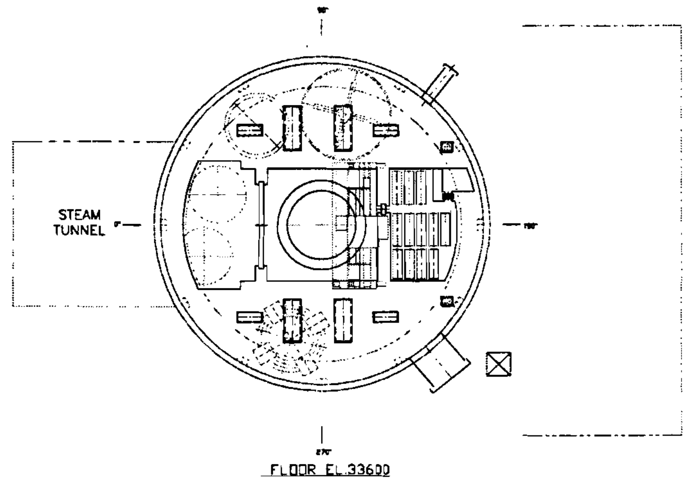
Evolution of BWR Containments and Reactor Buildings



Evolution and Innovation Towards Simplicity



Smaller structures – less equipment with good access
Separate fuel building and improved security



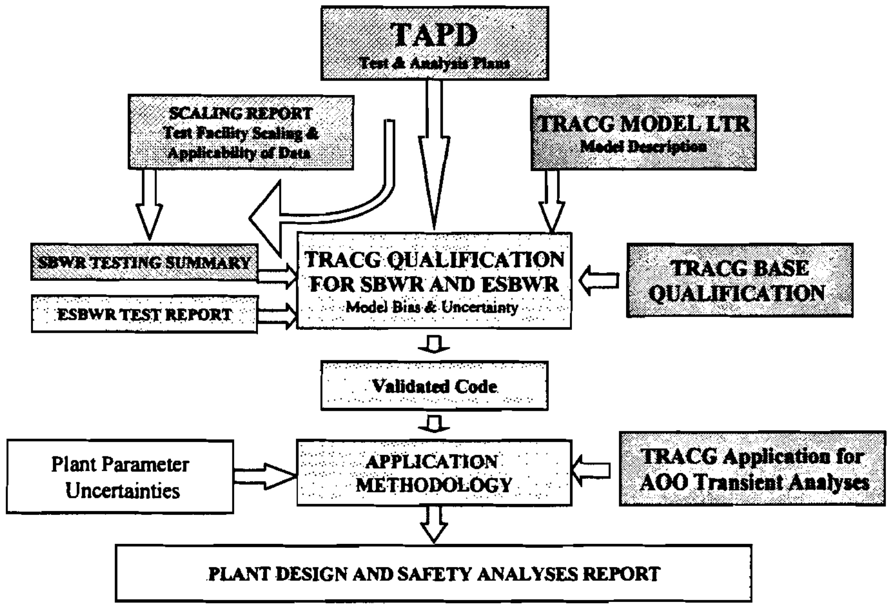
Refueling floor arrangement controls building size.
Improved fuel storage and transfer system

ESBWR and BWR Analysis Methods

Analysis Type	Analysis Method	
	BWR	***ESBWR***
Steady state	ISCOR	ISCOR
Transients		
· Pressurization	TRACG	TRACG
· Loss of feedwater heating	PANACEA	PANACEA
ATWS	ODYN/TASC	TRACG
Stability	ODYSY/TRACG	ODYSY/TRACG
LOCA/ECCS	SAFER	TRACG
LOCA/containment		
· Pressure/temperature response	M3CPT/SUPERHEX	TRACG
· Loads	Approved Methodology	Approved Methodology

TRACG – an integrated proven code - used for most analysis for ESBWR

ESBWR Technology Program Elements



Strategy for Determination of Test & Analysis Needs

- Develop list of governing phenomena and system interactions
- Top-Down Process based on plant accident/transient scenarios
  - Determine key phases of transients
  - List potentially important phenomena
  - Expert group ranking phenomena (PIRT)
- Bottom-Up process based on all unique ESBWR design features
  - Determine associated phenomena/system interactions
  - Evaluate and rank issues by importance
  - Supplements PIRT ranking approach to fill any gaps by focusing on ESBWR-unique features
- Consolidate highly ranked phenomena and system interactions
- Evaluate capability of analysis models & testing plans
  - Implement any needed models or bounding modeling procedures
  - Fill in testing gaps
  - Evaluate uncertainties to establish appropriate design margins

Rigorous process followed to define technology plan

Differences between ESBWR and SBWR

- Power Increased
  - Steamlines increased to 4 vs 2
  - 1020 bundles of 3 m height
    - F-lattice core design with wide span control blades
  - Isolation condensers (IC) capacity increased by 50%
  - Passive condenser (PCCS) capacity increased by 80%
    - 4 vs 3 units of 13.5MW/unit vs 10MW/per unit
- Plants systems and buildings
  - System sizes and capacities increased but not numbers
  - Utilize gravity drain system (GDCS) pool draindown space to provide increased wetwell volume (~15%)
  - Major building optimization incl. transfer of non safety systems and spent fuel storage

Differences do not affect governing phenomena (PIRT) for normal operation, transients and accidents

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Overview of Passive Plant Test Programs

- **Component Tests**

  PANTHERS/PCC (FULL SCALE)

  Full-scale prototype performance tests
  Steady state and transient tests with heavy (air) and light (helium) noncondensibles

  PANTHERS/IC

  One of two modules of a full scale unit
  Steady state, startup and transient tests

  PANDA PCC Tests (S Series) (1/50scale)

  10 steady state tests

  DPV Tests (FULL SCALE)

  Performance tests of prototype valve

  Vacuum Breaker Tests

  Performance tests of prototype valve

- **Integral System Tests**

  GIST (1/1000 scale)

  26 ECCS/LOCA integral tests with GDCS from wetwell pool

  GIRAFFE Step 3 (1/800 scale)

  Long term containment response

  GIRAFFE/Helium (1/800 scale)

  Long term containment response with light noncondensible gas

  GIRAFFE/SIT (1/800 scale)

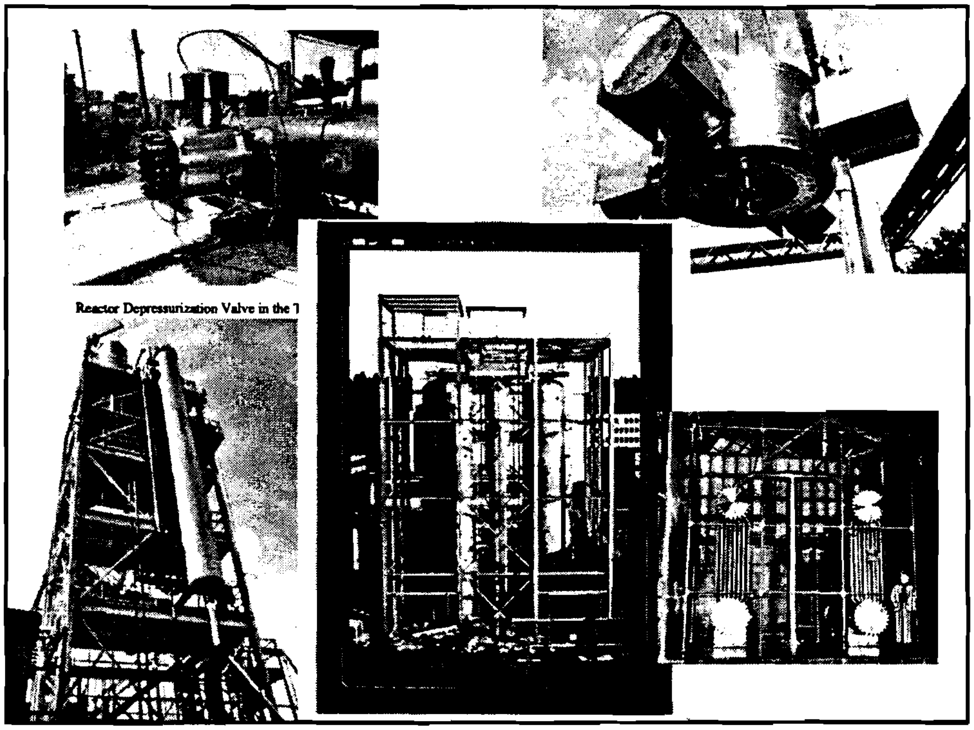
  ECCS/LOCA and containment integral tests with SBWR configuration (GDCS pool in drywell)

  PANDA (1/50 scale) M Series

  Long term containment response

Extensive tests at different scales in different facilities

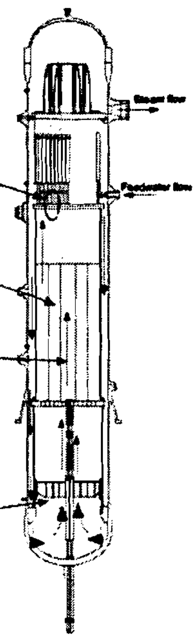
AR- 22



Other Applicable Test Programs

- CRIEPI Natural Circulation Stability Tests
  Thermal-hydraulic stability tests of natural circulation loop
- Dodewaard Plant Startup
  Natural circulation plant startup
- PSTF Mark III
  Containment Early Blowdown Response
  Suppression Pool Stratification
- Mark II 4T
  Containment Early Blowdown Response
- 1/6 Scale Boron Mixing Test
  Measurements of "boron" concentration in scaled regions of a BWR
- PANDA P-Series Program (performed after SBWR program was terminated)
  - Integral System LOCA Test with ESBWR configuration (~1/50 scale)
  - Tests of Long Term Containment Response and late GDCS period
  - Tests include release of lighter-than-steam gas (Helium)
- CRIEPI High Pressure Stability Tests (performed after SBWR program was terminated)
  - Thermal hydraulic stability tests with natural circulation loop

Natural Circulation Technology Program



Separator Performance
ATLAS Tests - AS2B
- smooth inlet geometry
- reduced pitch
  (305 mm -> 292 mm)

Chimney Void Fraction
Ontario Hydro Tests
- transient test (pump induced)
- round pipe (0.518 m ID)
- relatively flat void distribution

Startup Flow Oscillation
CRIEPI Tests
- single chimney
- SBWR conditions
- large margin to oscillation regime
Dodewaard Plant Start-up

Mixing
1/6 Scale Boron Injection
- low pressure
- salt simulant

Containment and Safety Systems Technology

SBWR

Condensation with N/C
MIT - external condensation
UCB - single tube tests
GIRAFFE - component tests
PANTHERS - component tests
PANDA - steady state tests

PCCS Performance
Steady-state: PANDA, GIRAFFE, PANTHERS
Start-up: PANDA, GIRAFFE
Secondary Side ht: PANDA, PANTHERS, GIRAFFE
N/C Buildup: PANDA, PANTHERS, GIRAFFE
Interactions: PANDA
Unit interactions PANDA

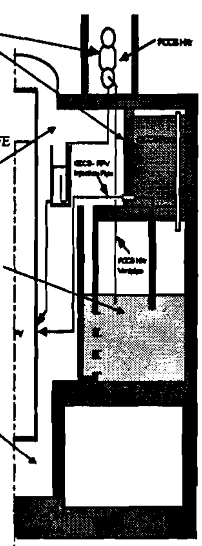
System Interactions
PANDA
GIRAFFE

SP Stratification
PANDA
SP Stratification

DW Stratification and Hideout
PANDA
GIRAFFE

Steam Quenching
Main Vent:
-Horiz. Vent Test/MK III tests (PSTF)
-Mark II 4T
PCC Vent:
- PANDA

Heat/Mass Leakage DW to WW
VB Testing



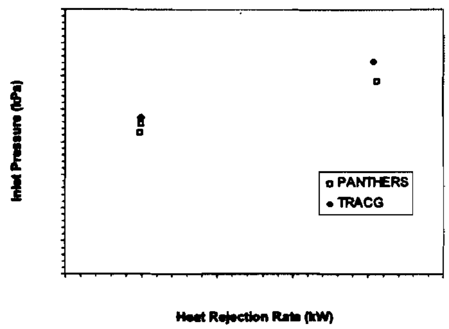
Confirmatory Tests

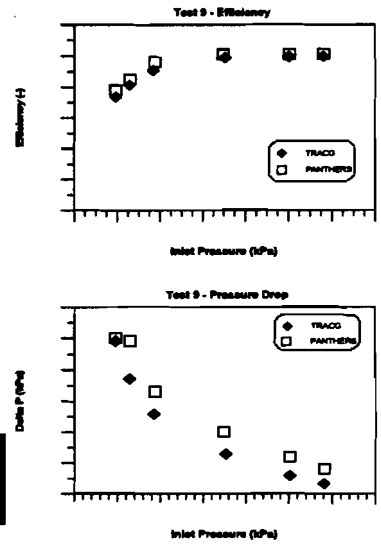
PCCS Performance
PANDA-P series
- startup
- interactions
- secondary side ht
- N/C Buildup
- Unit interactions

ESBWR Configuration
PANDA -P series
- reduced cont. volume
- GDCS in WW
- PCCS Condensate to RPV

DW Stratification and Hideout
PANDA-P series
- Asymmetric loading
- hydrogen

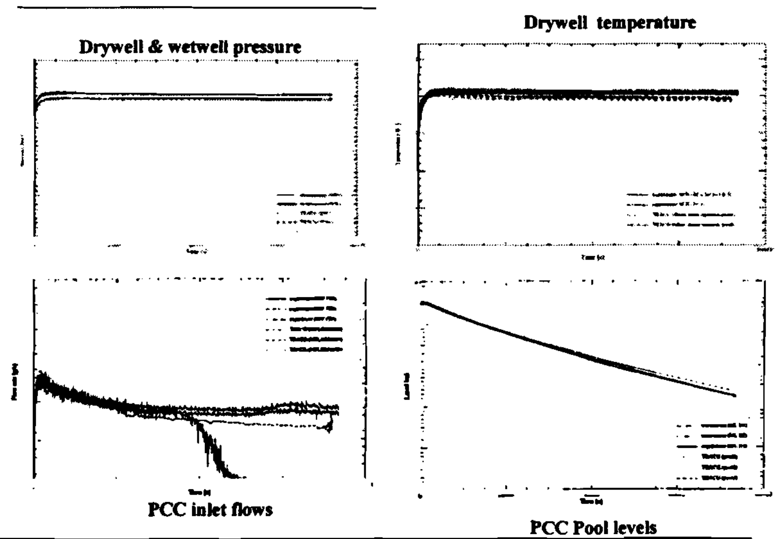
TRACG comparisons for PANTHERS PCC Tests - with & w/o non-condensibles





TRACG predicts component tests accurately for different test conditions

TRACG comparison to PANDA Test M3



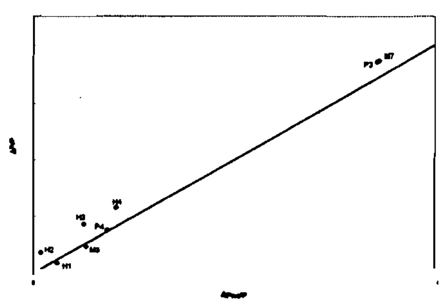
TRACG predicts figure of merit and details accurately

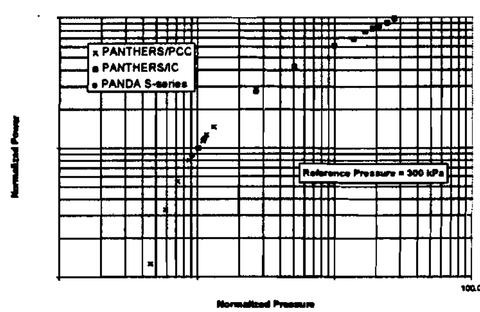
TRACG Qualification Summary

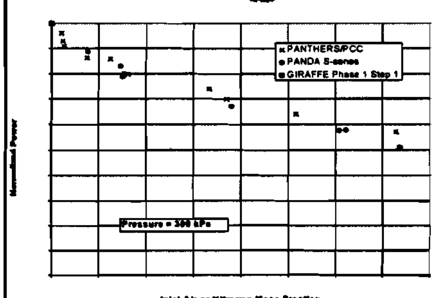
- All qualification activities identified in test and analysis plan have been satisfactorily completed
  - "generic qualification" studies have been reviewed and accepted by NRC for AOOs for operating plants
  - Significant additional qualification has been performed, particularly for long term containment response
  - Accuracy of models has been quantified for prediction of key parameters
- Model limitations have been identified and bounding approaches developed to treat these limitations
- TRACG is qualified for passive BWR (SBWR/ESBWR) analysis with appropriate application procedures

A comprehensive qualification program has been completed

Effect of Scale on test results







- PCC/IC's are Readily Scalable
  - PANDA IC/PCC is a section of PANTHERS IC/PCC
  - PANTHERS PCC is a slice of ESBWR PCC
  - GIRAFFE PCC has different header configuration
- Containment pressure varies with non-condensable gas quantity in wetwell for different scales and different gases

Tests at different scale show similar results

ESBWR Technology Program Summary

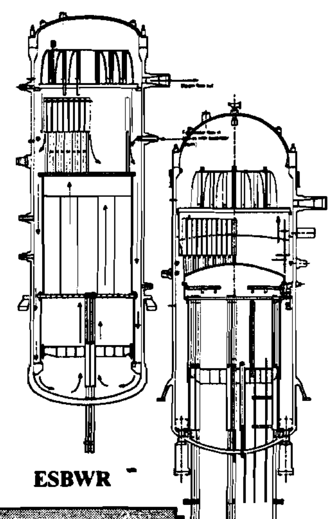
- Passive safety systems have simplified the plant design
- Plant evaluations are simpler
  - Less complex analyses
    - Realistic calc. with uncertainties defined as for operating plants - AOO
    - Realistic calc. with simplified accounting of uncertainties – ECCS/LOCA
    - Conservative calculation for containment/LOCA
  - Low parameter uncertainty
  - Substantial margins exist in the design
  - Improved integrated code shows better performance
  - Defense in depth systems provide additional back-up
- Extensive qualification of TRACG
- Technology issues extensively studied

Performance improved by design features
Improved performance measured by qualified methods

AR-31

Design Features Affecting Plant Response

	ESBWR	ABWR	BWR5	BWR4
Large pipes below core	No	No	Yes	Yes
Core height, m	3.05	3.66	~3.66	~3.66
TAF above RPV bottom	~ 1/4	~ 1/2	~1/2	~1/2
Separator standpipes	Long	Short	Short	Short
Vessel height, m	27.7	21.1	~21.9	~21.8
Water volume outside shroud (above TAF), m^3	222	88	94	92

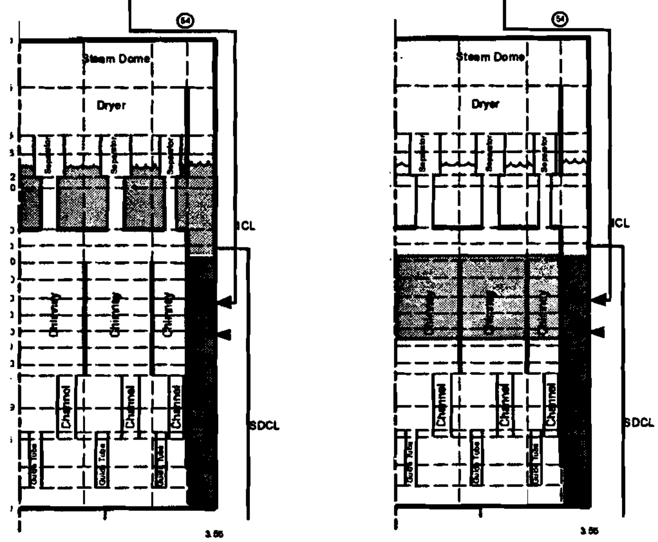


ESBWR

ABWR

Greater water inventory - improved plant LOCA performance
Larger steam volume – improved transient performance

Substantial Initial Water Inventory inside RPV



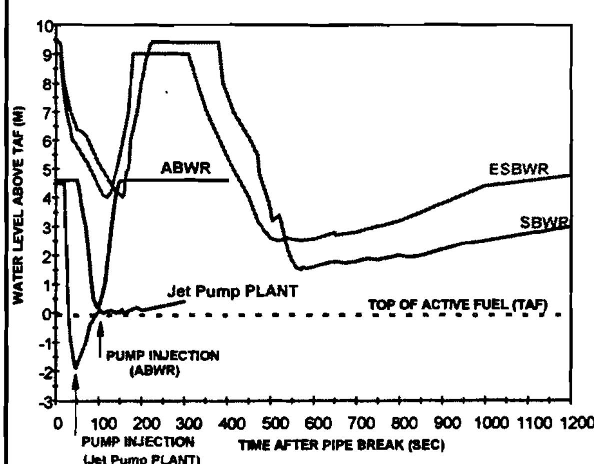
Collasped Levels at time = 0.0 sec

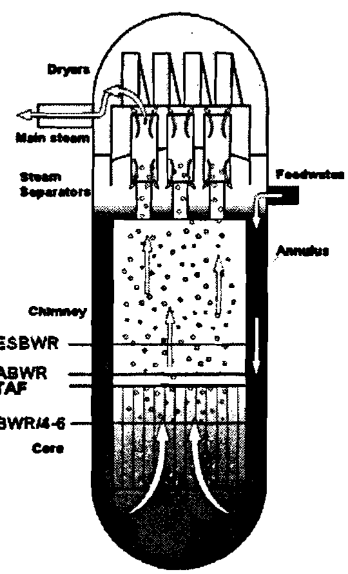
Collasped Levels at time ~ 20.0 sec

RPV Inventory Distribution Immediately following a LOCA

Water Level in Shroud Following a Typical Pipe Break

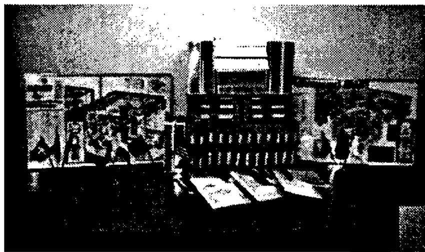
(values are intended to show typical trends for limiting breaks)





ESBWR Margins are Increased Over Past Plants

ESBWR Design/Technology based on SBWR and ABWR



Extensive new submittals

Extensive SBWR submittals and reviews, new test data and reports, coupled with design changes to add margin

Current status and near term deadlines

- **Extensive submittals made to NRC in 2002**
  - several meetings, conference calls and interactions
  - Extensive and thorough review by staff w 300+ RAI's
  - Final GE responses due by August 15, 2003
  - DSER for TRACG application due mid October
  - Additional application submittals covering AOO, ATWS and stability due over the next 6+ months
- **Preparation of SAR and certification**
  - Complete plant design optimization and SAR
  - Expected FDA 24 months after submittal
- **Challenges for the coming months**
  - Ensure GE responses are timely and complete
  - Complete technology closure (TRACG SER) with no open items

ESBWR Program Summary and Conclusion

- **15+ year technology and design program**
  - a BWR with less components
- **Simplification and margins by design**
  - large vessel results in benign response
  - analysis is simplified
- **Challenges for the coming months**
  - need closure and confirmation that regulatory risk is manageable

Expert Elicitation in Support of Risk-Informing 10 CFR 50.46

Robert L. Tregoning
Lee Abramson
US Nuclear Regulatory Commission

504th Advisory Committee on Reactor Safeguards Meeting
July 10, 2003

7

Previous ACRS Briefings and Program Milestones Since Last Briefing

- Previous ACRS briefings
  - May, 2002: Combined M&M, THP, R&PRA Subcommittee briefing on interim LOCA frequency elicitation and LOCA break size redefinition plans.
  - June, July, November, 2001: Overviews of LOCA frequency and break size redefinition effort provided to outline its importance within 10 CFR 50.46 revision framework.
  - March, 2001: Technical issues necessitating LOCA reevaluation.
- Program milestones Since May 2002
  - Selected expert panel and facilitation team.
  - Conducted kick-off meeting.
  - SRM Issued on SECY-02-0057 (Option III plan for risk-informing 10 CFR 50.46, Appendix K and GDC-35).
  - Conducted base case review meeting.
  - Held Public Meeting to discuss 10 CFR 50.46 effort: June 2003.

Expert Elicitation: Executive Summary

- Elicitation objective and approach are consistent with SRM guidance for development of near-term LOCA frequencies.
- Elicitation will develop LOCA frequencies as a function of leak rate and operating time considering both piping and non-piping contributions for all modes of plant operation.
- The conditional LOCA probabilities of larger, "emergency faulted" loadings are being estimated.
- Elicitation to combine aspects of group and individual elicitation approaches as appropriate to achieve objectives.
- Plans are in place to provide confirmatory analysis for the elicitation as well as develop a methodology for continually assessing LOCA challenges.

SRM Guidance

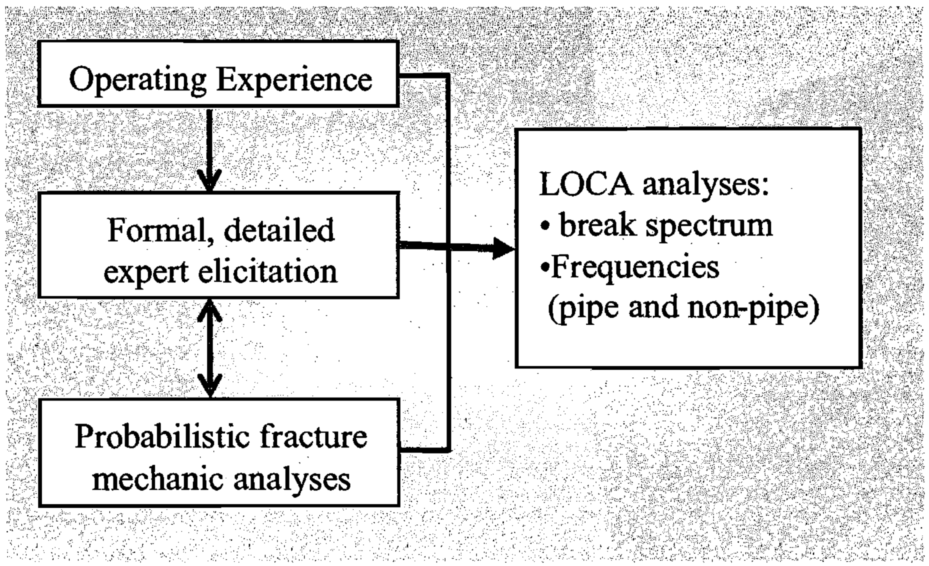
- The staff should conduct a practical reconciliation of LOCA frequency distributions by the 1) expert use of service-data, 2) Probabilistic Fracture Mechanics (PFM) and 3) expert elicitation to converge the results.
- Provide a comprehensive LOCA failure analysis and frequency estimation.
- Develop realistically conservative estimates, with appropriate margin for uncertainty.
- The staff should credit leak-before-break considerations only in conjunction with the establishment by a licensee of reliable and comprehensive means to detect primary system leaks of the relevant size.
- Use a 10-year period for the estimation of LOCA frequency distributions, with re-estimation every 10 years and review of new type of failures every 5 years.

General NRC Approach

1. **Operating Experience Assessment**
2. **Expert Elicitation.**
   - Re-evaluate LOCA frequencies.
   - Develop relationship between leak rate/break size and expected frequency for LOCA events.
   - Provide input to probabilistic LOCA computer code development.
3. **Probabilistic LOCA Code Development**
   - More rigorously combine operating experience and PFM insights.
   - Explicitly consider contributions from piping and non-piping components, and the evolution of new degradation mechanisms.
4. **Continual LOCA Assessment.**
   - Develop and maintain LOCA precursor database through expansion of existing pipe failure database.
   - Identify emerging degradation mechanisms and conduct anticipatory research to assess LOCA significance.

LB-LOCA Frequency Reevaluation

"The staff should conduct … expert elicitation to converge the results"



Value of Expert Elicitation

- Limitations of traditional approaches for LOCA frequency assessment.
  - Operating experience alone not sufficient to make future projections.
  - Probabilistic fracture mechanics (PFM) not sufficiently mature to model all system variables.
- Expert opinion (elicitation) is a formal process for providing quantitative estimates for the frequency of physical phenomena when the required data is sparse or when the subject is too complex to adequately model.
  - Data sparseness: no LOCAs have occurred.
  - Complexity is evident.
    - Many non-piping failure mechanisms.
    - Tremendous piping system variability.
- Elicitation has been used for similar problems.
  - Development of seismic hazard curves.
  - Performance assessments for high-level radioactive waste repository.
  - Determination of reactor pressure vessel flaw distributions.

General Elicitation Philosophy

- Use group assessment to develop base case conditions and frequencies. Conduct individual elicitations with respect to base case conditions and frequencies.
  - Assure common understanding of the base case.
  - Preserve independence and diversity of opinion.
- Conduct elicitation training.
  - Identify sources of motivational and cognitive biases.
  - Demonstrate value of elicitation process.
- Delay quantitative assessment until after panel discussion and issue analysis.
  - Ensure common understanding and clarity of issues.
  - Allow experts time to analyze issues.
- Require quantitative answers and rationale for mid, high, and low estimates.
  - Assess uncertainty
  - Ensure consistency among quantitative estimates.
  - Enhance understanding of LOCA frequencies contributing issues.
- Use facilitation team to guide elicitations.
  - Minimize motivational and cognitive biases.
  - Probe deeper into important issues.
  - Ensure comparable results among panel members.

Formal Elicitation Approach

- Select panel and facilitation team.
- Develop technical issues.
  - Define scope and objectives of elicitation.
  - Construct approach for determining LOCA frequencies.
  - Determine significant issues affecting LOCA frequencies.
- Quantify base case estimates.
  - Develop quantitative estimates for well-defined piping conditions.
  - Two estimates using PFM and two estimates from service history analysis.
- Formulate elicitation questions.
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

Panel Selection Process

- Expert Panel.
  - Panel solicited from industry, academia, national laboratories, contracting agencies, other government agencies, and international agencies.
  - Panel members chosen to represent a range of relevant technical specialties: PFM, piping design, piping fabrication, operating experience, materials, degradation mechanisms, thermo hydraulics, operating mitigation practices, stress analysis, nondestructive evaluation, etc.
  - Initially started with a pool of 55 nominally qualified people supplied by a number of knowledgeable sources (incl. future panel members).
  - Solicited resumes from 25 people and asked them to judge their relevant technical areas of expertise.
  - Final panel of 12 chosen based on broad relevant expertise, and to ensure a diversity of opinion, expertise, and backgrounds.
- Facilitation Panel.
  - Comprised of normative and substantive experts.
  - Substantive experts chosen to provide relevant background knowledge.

Formal Elicitation Approach

- Select panel and facilitation team.
- **Develop technical issues.**
  - **Define scope and objectives of elicitation.**
  - **Construct approach for determining LOCA frequencies.**
  - **Determine significant issues affecting LOCA frequencies.**
- Quantify base case estimates.
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  - Two estimates using PFM and two estimates from service history analysis.
- Formulate elicitation questions.
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

Elicitation Scope and Objectives

- Develop piping and non-piping passive system LOCA frequencies as a function of leak rate and operating time up to the end of the license extension period.
- Determine LOCA frequency distributions for typical plant operational cycle and history including all modes of operation.
- Estimate condition LOCA probability distributions for rarer, emergency faulted load conditions.
  - Seismic loading.
  - Other large, unexpected internal and external loads.

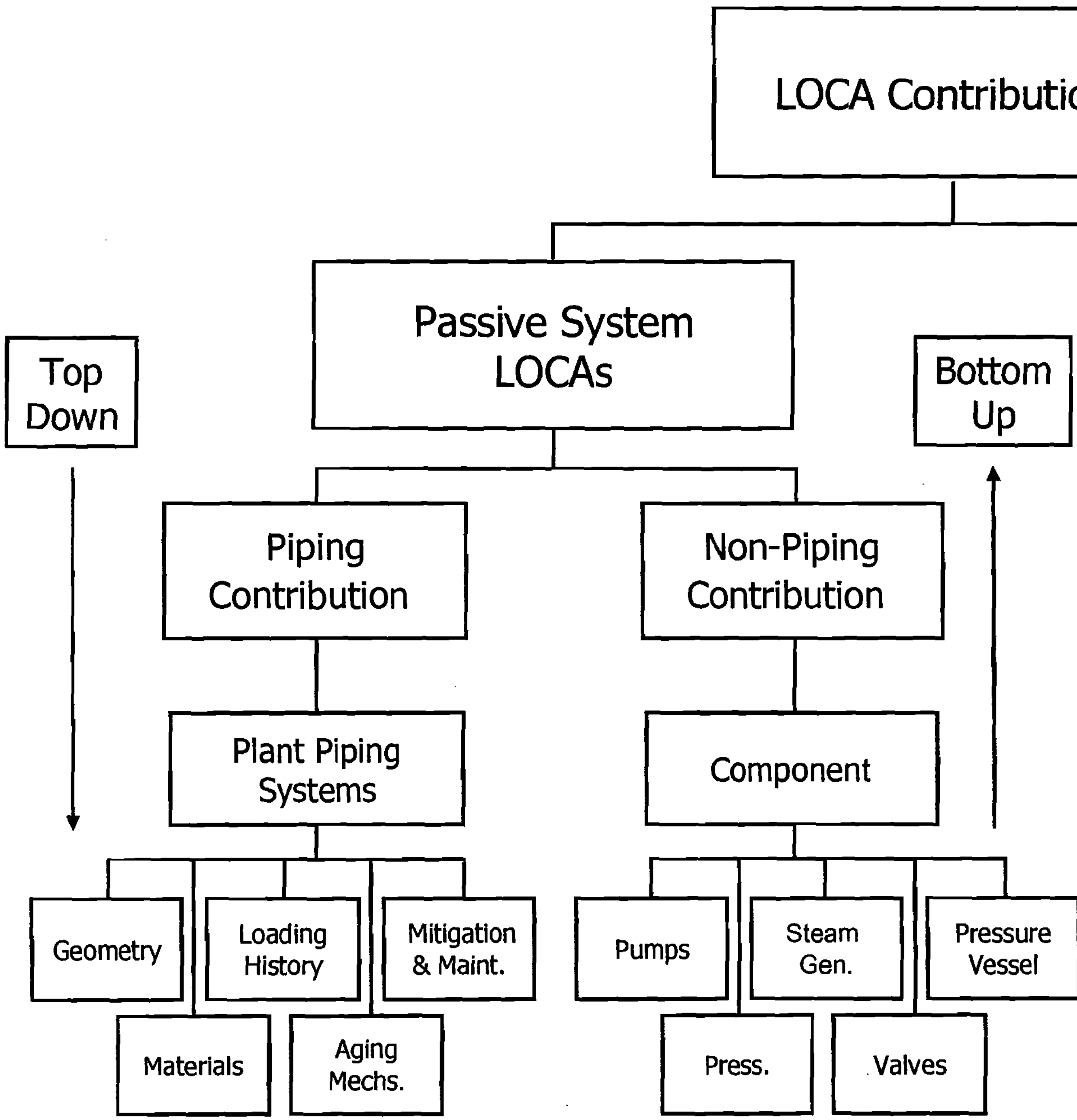
LOCA Sizes and Operating Time Periods Evaluated

- LOCA sizes based on leak rate to group plant system response characteristics.
- First three categories encompassed traditional definitions utilized in NUREG-1150 and NUREG/CR-5750.
- Three more LBLOCA categories added to examine trends with larger break sizes.
- Correlation between leak rate and break size developed for relevant BWR and PWR systems.

Category	Leak Rate Threshold (gpm)	LOCA Size
1	> 100	SB
2	> 1500	MB
3	> 5000	LB
4	> 25,000	LB a
5	> 100,000	LB b
6	> 500,000	LB c

- Three time periods evaluated.
  - Current (average 25 years of operating experience.
  - End of design life (40 years of operation).
  - End of life extension (60 years of operation).

General Issue Classification



- Elicitation will focus on passive system LOCAs.
- Relevant active system LOCAs will be added.
- Important piping and non-piping variables identified.
- Elicitation structure will support top down and bottom up analysis.

Piping Issue Classification

- Brainstormed variables in all relevant categories which influence the LOCA frequencies: materials, geometry, loading, mitigation & maintenance, degradation mechanisms.
- Agreed that variables and their effects are a function of the piping system.
- Individual categories developed for experts to summarize pool of applicable variables for a given piping system.
- Determined LOCA sensitive piping systems for BWR and PWR plants.
- Determined which of the individual variables in each of the five categories were relevant for each piping system considering individual plant variability.
- Developed master tables for BWR and PWR plants for the expert elicitation panel.

Non-Piping Issue Classification

- Identified approximately 25 different sections of primary components (i.e. pressurizer, reactor, steam generator, pumps, valves) where passive system failures could lead to a LOCA.
- Discussed failure mechanisms which could lead to LOCAs in these components.
- Identified components that may have existing failure data.
- Determined which of the individual variables in each of the five categories were relevant for each non-piping system.
- Developed master tables for non-piping LOCA contributors for the expert elicitation panel.

Formal Elicitation Approach

- Select panel and facilitation team.
- Develop technical issues.
  - Define scope and objectives of elicitation.
  - Construct approach for determining LOCA frequencies.
  - Determine significant issues affecting LOCA frequencies.
- **Quantify base case estimates.**
  - **Develop quantitative estimates for well-defined piping conditions.**
  - **Two estimates using PFM and two estimates from service history analysis.**
- Formulate elicitation questions.
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

Piping Base Case Development

- The base cases will be used to anchor the elicitation responses.
- Base case conditions specify the piping system, piping size, material, loading, degradation mechanism(s), and mitigation procedures.
- Five Base Cases Defined.
  - BWR
    - Recirculation System
    - Feedwater System
  - PWR
    - Hot Leg
    - Surge Line
    - High Pressure Injection makeup.
- The LOCA frequency contribution (per year) of each set of base case conditions will be calculated as a function of leak rate and operating time.
- Four panel members chosen to perform calculations: two using operating experience and two using probabilistic fracture mechanics.

Piping Base Case Approach

- Iterative process involving facilitation team and expert panel.
- Evaluate LOCA frequencies at 25 (current), 40 (end-of-license), and 60 years (end-of-license extension) after plant startup.
- Each base case member should benchmark results using service experience for leaking cracks.
- All base case calculations should capture as closely as possible the conditions established by the expert panel.
- Sensitivity analyses of PFM results conducted to evaluate:
  - Seismic loading
  - Effect of ISI
  - Loading history variability
  - Effectiveness of mitigation

Non-Piping Base Case Development

- The non-piping base cases could have been developed in a similar manner to the piping base cases.
  - Choose several representative systems.
  - Examine and extrapolate operating experience through modeling
- However, the variety and complexity of the non-piping failure mechanisms makes this assessment intractable and of limited value.
- Philosophy here is to conduct database searches for each non-piping failure mechanism listed to develop leaking and cracked component frequencies.
- These frequencies will be used to anchor the non-piping responses for each expert.
- Each expert must determine how to translate the leaking and crack frequency information into meaningful LOCA estimates.

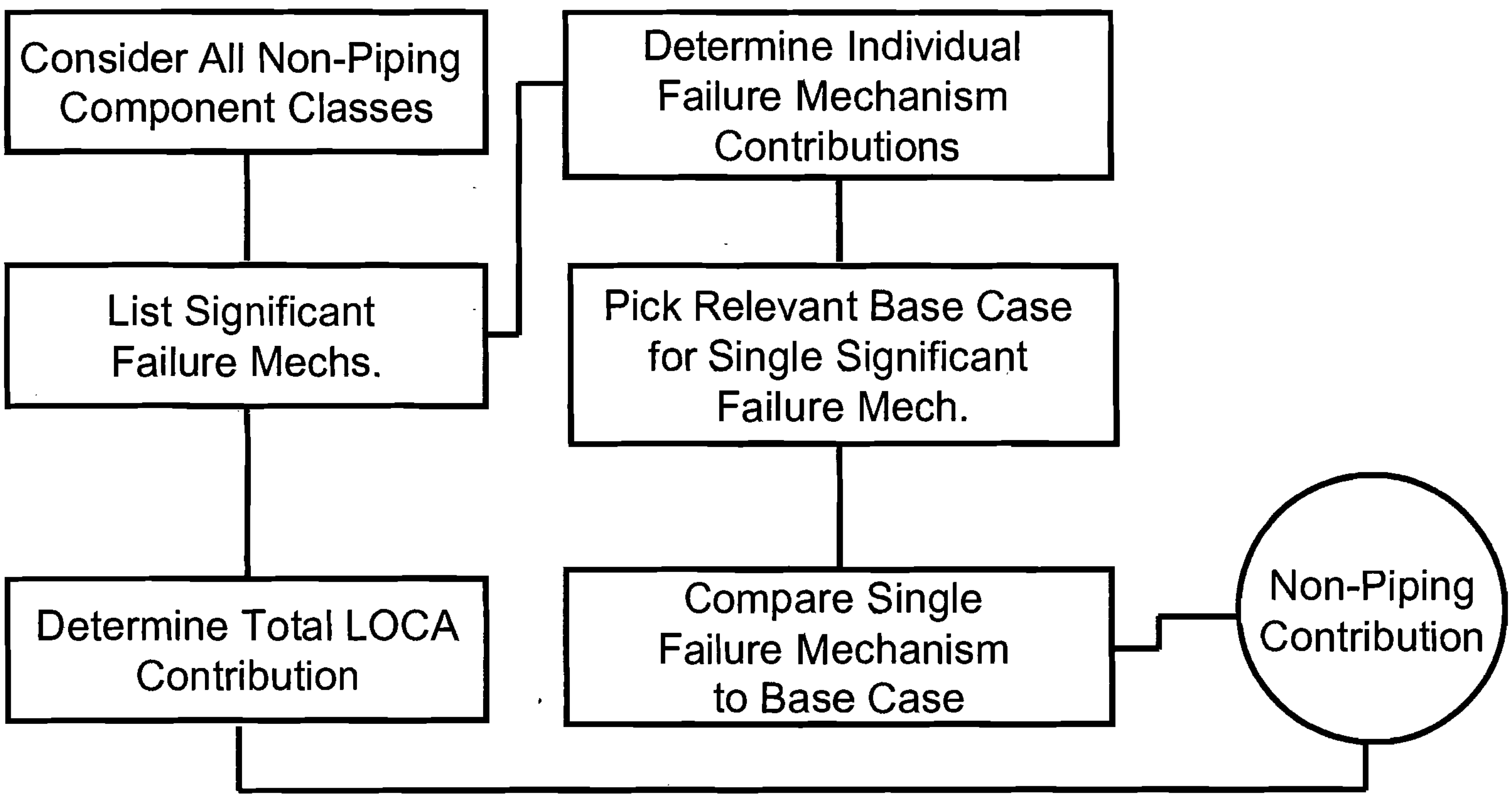
Formal Elicitation Approach

- Select panel and facilitation team.
- Develop technical issues.
  - Define scope and objectives of elicitation.
  - Construct approach for determining LOCA frequencies.
  - Determine significant issues affecting LOCA frequencies.
- Quantify base case estimates.
  - Develop quantitative estimates for well-defined piping conditions.
  - Two estimates using PFM and two estimates from service history analysis.
- **Formulate elicitation questions.**
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

Elicitation Question Development

- Questions will be focused on the following topic areas.
  - Base Case Evaluation.
  - Regulatory and Utility Safety Culture.
  - LOCA frequencies of Piping Components.
  - LOCA frequencies of Non-Piping Components.
  - LOCA Frequencies of Piping Components under Emergency Faulted Loading Conditions.
  - LOCA Frequencies of Non-Piping Components under Emergency Faulted Loading Conditions.
- Questions will be asked relevant to a set of conditions and quantitatively linked to the base case results.
- Each question will ask for mid, low, and high values for each question as well as appropriate rationale or comments.
- Questions can be answered using a top-down or bottom-up approach.
- Rationale will be discussed for important and some unimportant issues for each expert.

Non-Piping Components: Top-Down Approach



Ongoing and Future Elicitation Work

- **Conduct individual elicitations.**
  - Provide answers to questions and rationale for answers.
  - Discuss significant issues which impact LOCA frequency estimation.
  - First two scheduled for July 15th and 16th. Remaining ten to be completed between mid-August and end of September.
- **Analyze quantitative results and qualitative rationale.**
  - Calculate results for each expert if appropriate.
  - Combine answers for individual questions and calculate results.
  - Propagate uncertainties.
- **Conduct wrap-up meeting.**
  - Summarize quantitative and qualitative results.
  - Summarize analysis methodology and LOCA results.
  - Obtain feedback from the expert panel.
- **Summarize and document results.**

SRM Guidance

- The staff should conduct a practical reconciliation of LOCA frequency distributions by the 1) expert use of service-data, 2) Probabilistic Fracture Mechanics (PFM) and 3) expert elicitation to converge the results.
- **Provide a comprehensive LOCA failure analysis and frequency estimation.**
- **Develop realistically conservative estimates, with appropriate margin for uncertainty.**
- **The staff should credit leak-before-break considerations only in conjunction with the establishment by a licensee of reliable and comprehensive means to detect primary system leaks of the relevant size.**
- Use a 10-year period for the estimation of LOCA frequency distributions, with re-estimation every 10 years and review of new type of failures every 5 years.

Addressing Specific SRM Guidance

- **"Provide a comprehensive LOCA failure analysis and frequency estimation."**
  - Include piping and non-piping LOCA contributions.
  - Consider LOCA contributions from all modes of operation.
  - Examine conditional LOCA probabilities under extreme loading conditions.
- **"Develop realistically conservative estimates, with appropriate margin for uncertainty."**
  - Develop best estimate values for all LOCA distributions.
  - Evaluate uncertainty among all experts as well as for each expert's responses. Combine answers for individual questions and calculate results.
  - Propagate uncertainties to understand impact and appropriateness of subsequent regulatory action.
- **"The staff should credit leak-before-break considerations …".**
  - Credit leak detection based on technical specification limits when evaluating both the service history and in making future projections of operating performance
  - Identify and quantify LOCA contributions of non-leaking cracks.

SRM Guidance

- The staff should conduct a practical reconciliation of LOCA frequency distributions by the 1) expert use of service-data, 2) Probabilistic Fracture Mechanics (PFM) and 3) expert elicitation to converge the results.
- Provide a comprehensive LOCA failure analysis and frequency estimation.
- Develop realistically conservative estimates, with appropriate margin for uncertainty.
- The staff should credit leak-before-break considerations only in conjunction with the establishment by a licensee of reliable and comprehensive means to detect primary system leaks of the relevant size.
- **Use a 10-year period for the estimation of LOCA frequency distributions, with re-estimation every 10 years and review of new type of failures every 5 years.**

LOCA Frequency Reevaluation: Passive LOCA Code Development

- **Objectives:**
  - Determine the relationship between break size and expected event frequency for large primary system pipes (>150 mm diameter).
  - Provide confirmatory analysis of elicitation results.
  - Develop tool which can be used for subsequent frequency re-evaluation.
- **Approach**
  - Construct separate modules to consider piping, non-piping contributions, and future surprise mechanisms.
  - Modules will couple state-of-the-art PFM modeling with understanding of operating experience historical, recent, and potential degradation mechanisms to determine frequency partitioning.
  - Scale modeling frequencies using expert judgment to determine the LBLOCA frequency.
  - Use insights from elicitation to initially focus on the most important systems and mechanisms.
  - Monitor and interact with the NURBIM program which has similar objectives.

LOCA Frequency Reevaluation: Continuous Assessment of LOCA Challenges

- **Objectives:**
  - Develop framework for evaluating operating experience for LOCA precursor events in piping and non-piping components.
  - Evaluate mechanisms or trends which could be detrimental affect future LOCA frequencies.
- **Approach**
  - **Participate in the CSNI-sponsored OECD Piping Database Exchange (OPDE) project to expand international operating experience.**
    - Twelve participating countries: Belgium, Canada, Czech. Republic, France, Finland, Germany, Japan, Korea, Spain, Sweden, Switzerland, U.S.
    - The SKI-pipe SLAP database serves as the baseline.
    - Year 1 of 3 year effort is focusing on events between 1998 and 2001. Year 2 on events from 1995 – 1998 and 2002.
  - **Continue to evaluate international operating experience and research results to identify trends and mechanisms which warrant further research.**

SOUTH TEXAS PROJECT UNIT 1
BOTTOM MOUNTED INSTRUMENTATION NOZZLE LEAKAGE ISSUE

Matthew A. Mitchell, Senior Materials Engineer
Materials and Chemical Engineering Branch
Office of Nuclear Reactor Regulation

Advisory Committee on Reactor Safeguards Full Committee Meeting
July 11, 2003

BACKGROUND

- April 12, 2003 - Licensee performed boric acid corrosion control (BACC) walkdowns as part of GL 88-05 program. Inspections included a bare metal visual examination of the reactor pressure vessel (RPV) bottom head.

- The licensee's access to the South Texas Project Unit 1 (STP Unit 1) RPV lower head is very conducive to these inspections. Plant design includes an insulating "box" around the lower head with panels that can be opened to permit direct viewing of the bare metal.

- Licensee had performed similar inspections of the lower heads of both STP Unit 1 and Unit 2 previously. The most recent inspection of Unit 1 had been conducted in November 2002 with no evidence of deposits noted.

BACKGROUND

- In April 2003, the licensee discovered deposits characterized as, in total, “about the size of one half of an aspirin tablet” around bottom mounted instrumentation (BMI) penetrations #1 and #46.

- Chemical analysis showed evidence of boron and lithium, indicating the reactor coolant system (RCS) to be the most likely source of the deposits.

- Radiochemical analysis based on cesium isotope dating indicated that the deposits were approximately four years old.

NONDESTRUCTIVE EXAMINATION - SCOPE

- The licensee has conducted extensive nondestructive examination (NDE) on all 58 STP Unit 1 BMI nozzles. Framatome Technologies was chosen as the vendor for the inspections, using a tooling system which had been used previously for BMI inspections in France.

- Performed ultrasonic testing (UT) using axial, circumferential, and zero degree probes from the tube inside diameter (ID) on all nozzles.

- Performed enhanced visual testing (EVT-1) examinations of the J-groove weld surfaces of all nozzles.

- Performed ID eddy current testing (ECT) on some nozzles to confirm UT data.

- Performed “ECT-on-a-stick” examination of the J-groove weld surface of eight penetrations, including #1 and #46.

NONDESTRUCTIVE EVALUATION - RESULTS

- The licensee's NDE results showed:

  - Three axially-oriented indications in nozzle #1. One indication characterized as having a length of ~1.38 inches, extending from above to below the J-groove weld and penetrating the ID of the tube. The other two indications were much smaller and near the root of the weld.

  - Two axially-oriented indications in nozzle #46. One indication characterized as having a length of ~0.98 inches and extending from above to below the J-groove weld. The other indication characterized as having a length of ~0.95 inches and not surface connected.

  - EVT-1 examinations showed signs of extensive grinding on the nozzle and J-groove weld surfaces of many penetrations.

NONDESTRUCTIVE EVALUATION - ADDITIONAL

- The licensee performed additional NDE tests on penetrations #1 and #46, including:

  (1) ECT profilometry on nozzles #1 and #46 to compare as-found nozzle distortions with that predicted from weld finite element modeling to validate predicted weld residual stresses. Preliminary results suggest that the profilometry measurements were consistent with finite element modeling predications.

  (2) Helium pressurization tests on nozzles #1 and #46 to further investigate potential leakage paths in these penetrations. At 150 psi, bubbles were observed on nozzle #1, but not on nozzle #46.

  (3) Phased-array UT from the RPV head outside surface to look for evidence of wastage of the ferritic base material of the head. No evidence of wastage was found.

PRELIMINARY ROOT CAUSE ANALYSIS

- Based on the information currently available, two principal root cause theories are under consideration by the licensee.

(1) The cracking was caused by primary water stress corrosion cracking (PWSCC) which initiated in the nozzle at the toe of the J-groove weld.

- PWSCC of Inconel 82/182/600 observed in other applications
- Consistent with expectations in 1991 Westinghouse report for Sequoyah which assessed potential for BMI cracking
- Inconsistent with the fact that no cracking was observed in other penetrations

(2) The cracking initiated at “discontinuities” (weld lack of fusion, etc.) at the tube/weld interface and propagated to the tube surface.

- Consistent with observed discontinuities in #1 and #46
- Consistent with understanding of general fabrication practices/issues
- Inconsistent with the fact that discontinuities were evident in other penetrations
- No specific mechanism to explain subcritical crack growth

PRELIMINARY ROOT CAUSE ANALYSIS

- The licensee is taking material samples from nozzles #1 and #46 for evaluation. Information from these samples is expected to confirm the degradation mechanism(s) and, potentially, the initiation sites for the observed indications.

- Information from these material samples is expected to clarify whether either of the two principle preliminary root causes is substantiated. Some combination of mechanisms may also be indicated by the information from the material samples.

- Information from the licensee's evaluation of the materials samples will be included in the final root cause report which is currently projected to be completed in late September/early October, 2003.

STP UNIT 1 BMI NOZZLE REPAIRS

- The licensee has repaired STP Unit 1 nozzles #1 and #46 using a "half nozzle repair" similar in design to those used to repair other Alloy 600 penetrations.

- The repair was made using Alloy 690 nozzle material and Alloy 52/152 weld material, including the installation of a temper bead weld pad on the outside of the RPV lower head. The RCS pressure boundary weld was moved to the outside surface of the RPV.

- Questions regarding future inspections of the repair, along with inspections of the ferritic base material which will be left exposed to the reactor coolant, are being addressed by the licensee in support of NRC staff review and approval of the repair.

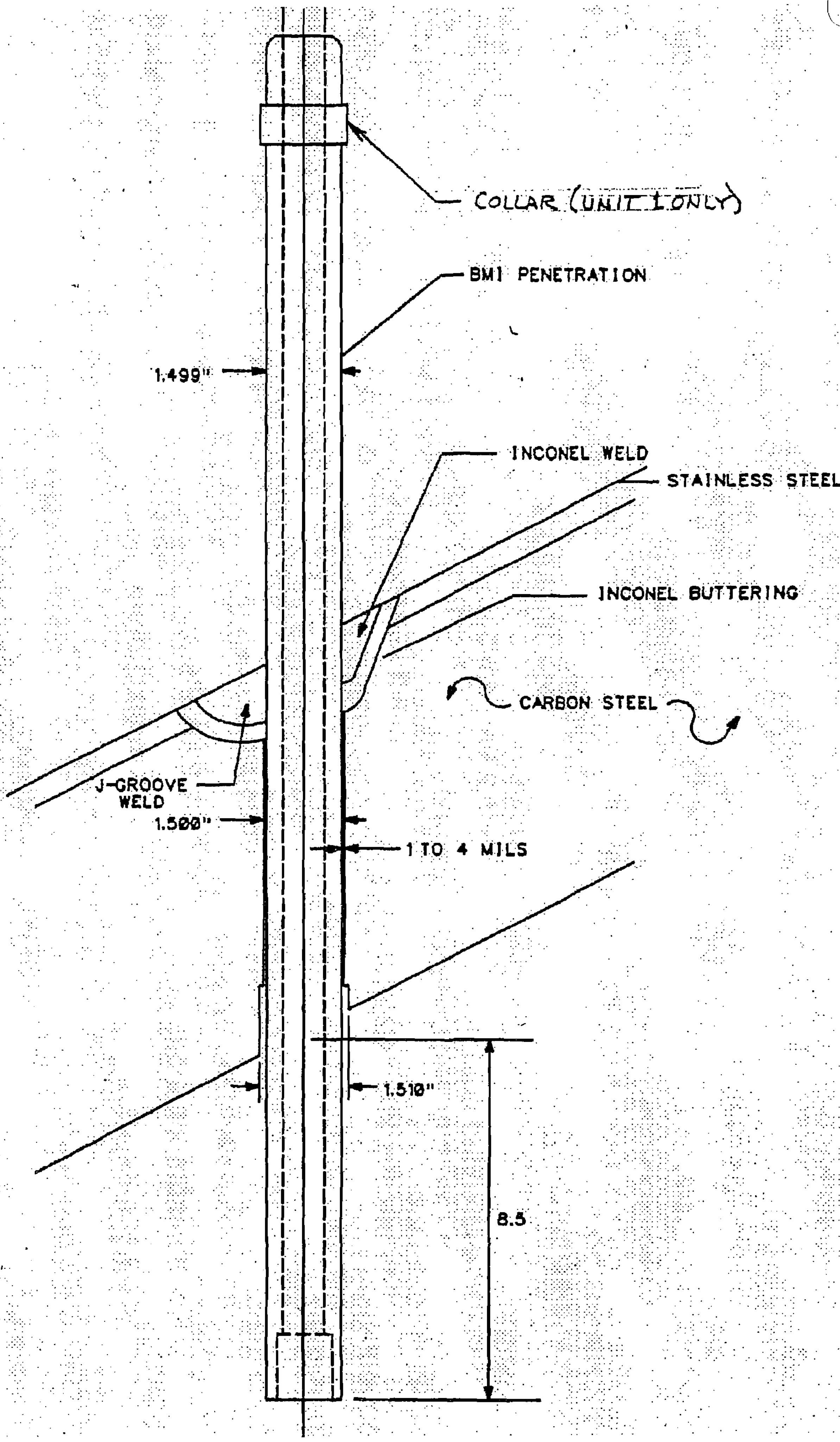
POTENTIAL GENERIC IMPLICATIONS

- None of the available information suggests that STP Unit 1 is unique with regard to being susceptible to lower head penetration cracking.

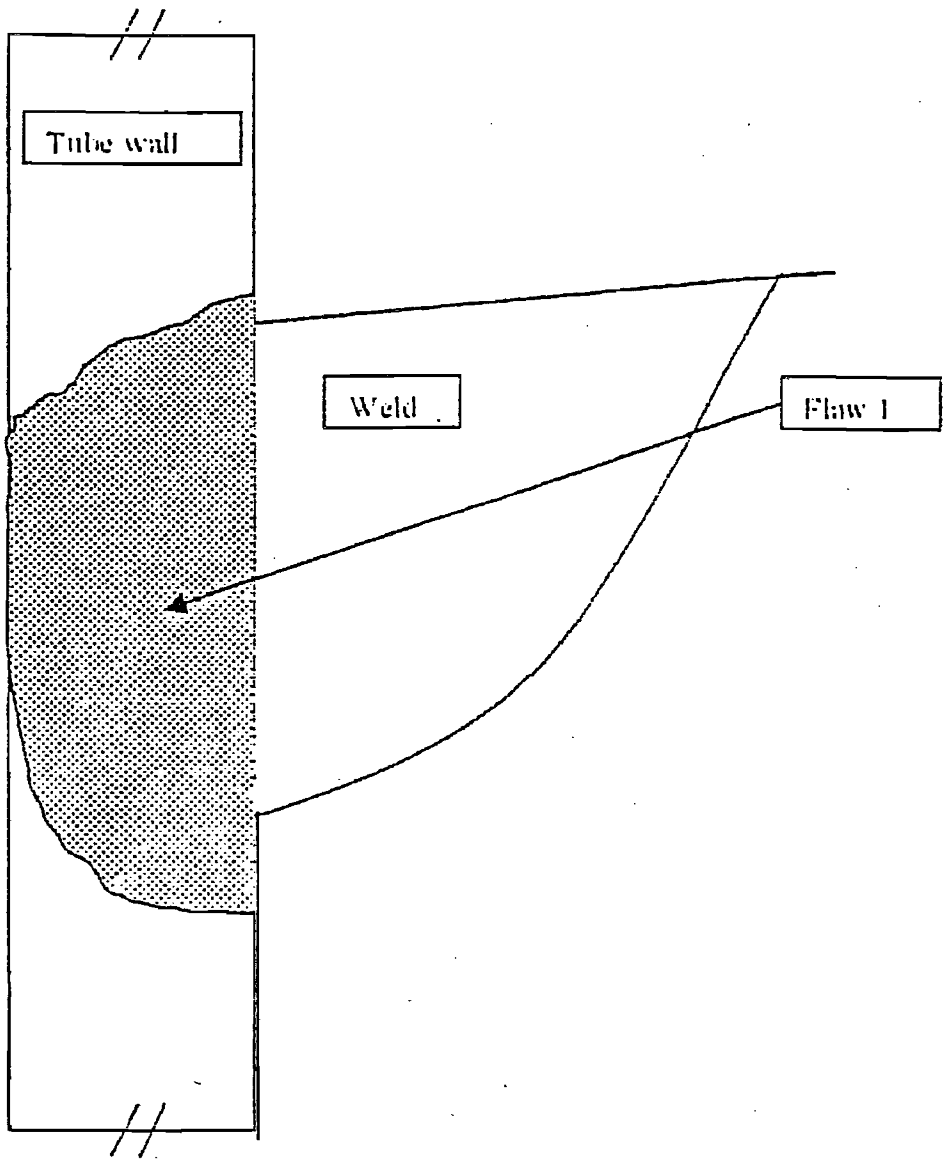
- Based on the "as found" condition of the STP Unit 1 BMI penetration nozzles, the NRC staff has concluded that the risk significance of the situation at STP Unit 1 was minimal.

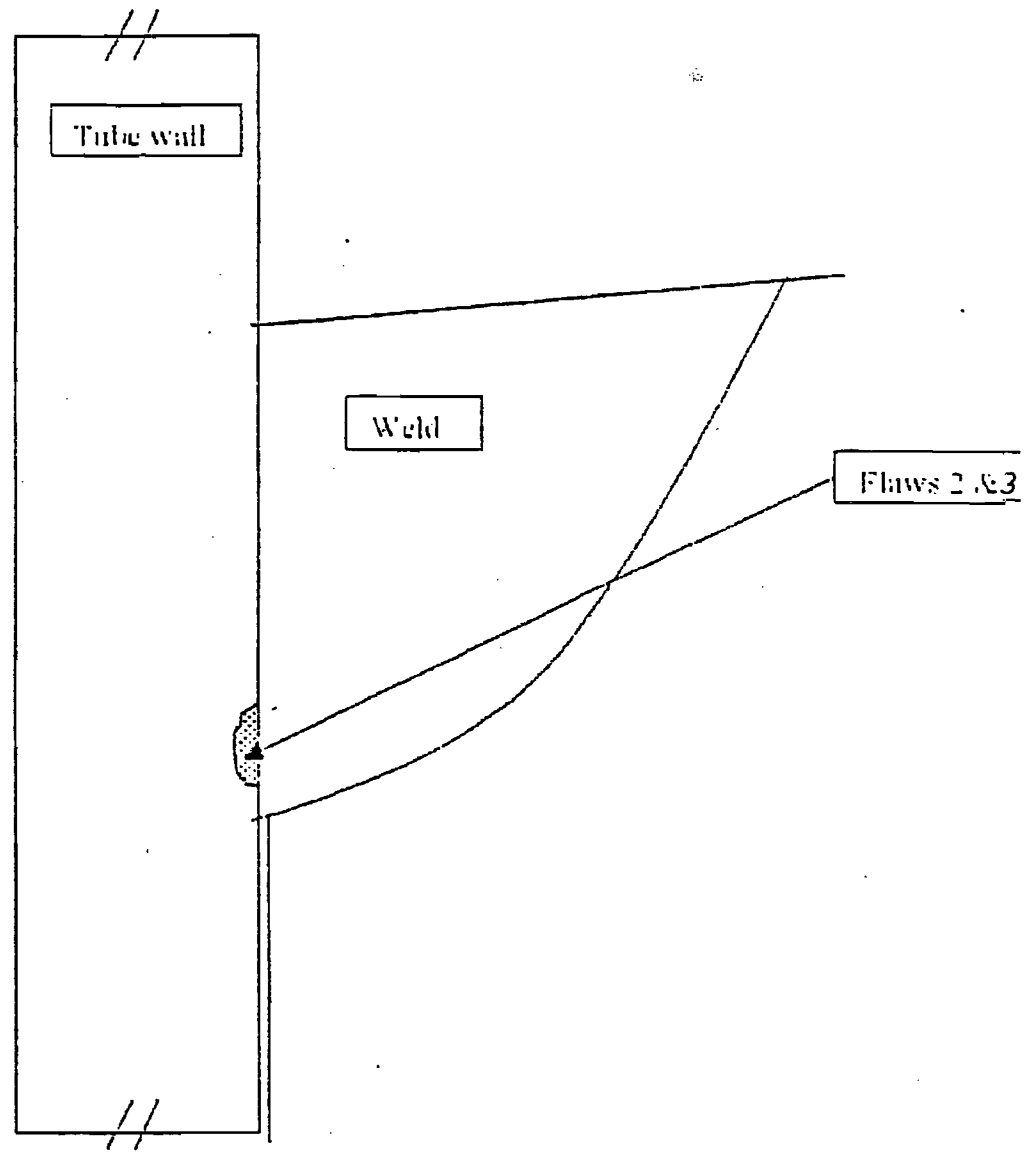
- However, should the operative degradation mechanism(s) at STP Unit 1 be directly or indirectly capable of inducing large, circumferentially-oriented flaws in RPV lower head penetrations, the risk implications for the U.S. PWR fleet could be significant.

- The NRC staff is in the advanced stages of determining what path we intend to follow with regard to developing generic communication(s) concerning PWR RPV lower head inspections given the information coming out of the STP Unit 1 event.

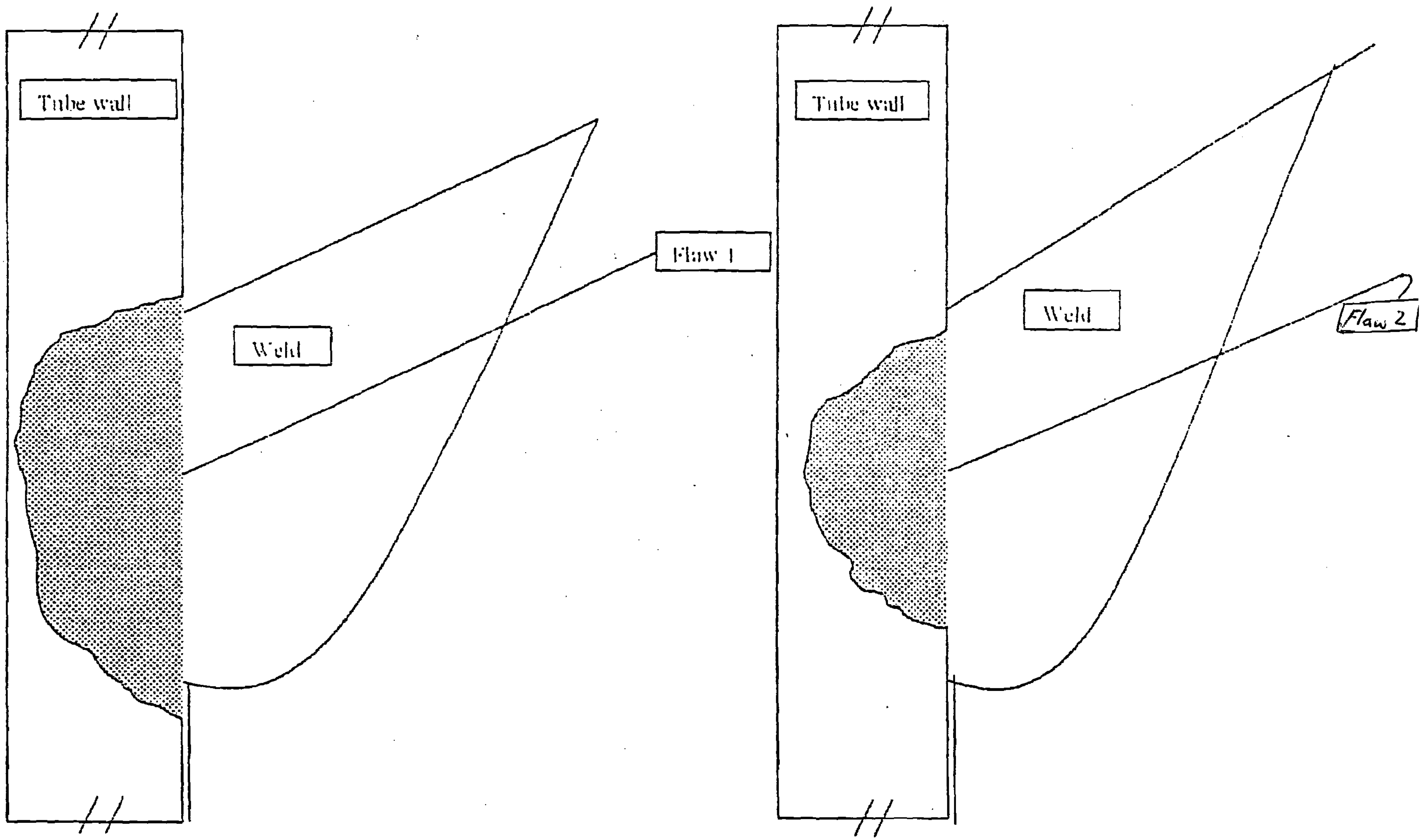

COLLAR (UNIT 1 ONLY)
BMI PENETRATION
1.499"
INCONEL WELD
STAINLESS STEEL
INCONEL BUTTERING
CARBON STEEL
J-GROOVE WELD
1.500"
1 TO 4 MILS
1.510"
8.5

Penetration #1





Penetration #46



(10)

OPERATING EVENTS
APRIL - JUNE 2003

HIGHLIGHTS

STP1	BMI indications - Separate discussion
Quad Cities 2	Stuck Open Relief Valve Fuel Problems Dryer Cracking
BWR fuel	I'm hearing several reports of fuel leaks in both GE and Framatome fuel. Limerick, Brunswick, Pilgrim, Peach Bottom, Quad Cities, others. What's happening here? I think we need a presentation on this.
Automatic Scrams	In 3 months 8 of 13 Automatic Scrams were related to generator, main transformer or switchyard. This seems to be a continuing trend. Of the remaining 5 Automatic Scrams, 3, or perhaps 4, appear due to electronic component failures which alone can cause a scram. I think we need to hear an analysis of this data.

NOTE: On July 7, 2003, all units in country (with exception of Davis Besse and South Texas) were at nominal 100%.

G. Leitch
Jul 10, 2003

OPERATING EVENTS
APRIL - JUNE 2003

Siren & ENS

Sirens & ENS	Many problems with sirens and emergency notification systems to numerous to list. Almost all weather related or a few caused by traffic accidents. Two occasions at Indian Point 5/15/03 and 6/17/03	

Fires

Robinson 2	4/26/03	Unusual Event - Fire in Circ Water Pump Motor lasting more than 10 minutes. No flames. No offsite assistance.
Indian Point 3	4/29/03	Unusual Event - Turbine and Reactor Manually tripped due to fire in Turbine. Fire out in 47 minutes - no offsite help.
Monticello 1	5/2/03	Small fire in Control Room Alarm Panel Immediately extinguished by hand held extinguisher.
Seabrook 1	6/30/03	Fire in Unit 2 Containment - Extinguished in ½ hour by contractor doing disassembly work.
TMI 2	7/2/03	Unusual event - Fire in Unit 2 - Transformer inn Mcc. Out in ½ hour by station fire brigade.

Scrams

<u>Main Generator/Main Transformer/Switchyard Related</u>

Grand Gulf 1	4/24/03	Auto Scram from 100% power. High winds caused minor damage to switchyard. Wind blew disconnect switch closed. Partial L.O.O.P.
Indian Pt. 2	4/28/03	Auto Scram from 100% power due to trip of generator breakers. Turbine overspeed trip.
Summer 1	5/12/03	Auto Scram from 100% power. Exciter Breaker opened for reasons unknown. Exciter tripped, then generator, then turbine, then reactor.
Comanche Peak Both Units	5/15/03	Auto Scram on both units from 100% power due to loss of 345Kv grid.

OPERATING EVENTS
APRIL - JUNE 2003

Pilgrim	6/1/03	Auto Scram from 100% power. Main Generator lockout relay actuation caused both 345Kv Breakers to open.
North Anna 1	6/11/03	Auto Scram from 100% power due to main transformer relay activation.
Indian Pt. 3	6/22/03	Auto Scram from 100% power. White closing one 345Kv Breaker after maintenance a fault occurred. Tripped generator then Reactor and Turbine.

Other Automatic Scrams

Peach Bottom 2	4/12/03	Auto Scram from 100% power - One MSIV Closed due to instrument air line failure.
Sequoyah 2	4/12/03	Auto Scram from 100% power due to false Main Turbine high vibration trip.
Pilgrim	5/19/03	Auto Scram from 3% while starting up. Main Turbine by pas valves failed open. Five rods failed to fully insert. Later reported that rods were beyond "full in".
Harris 1	5/17/03	Auto Scram from 28% power due to unexplained RPS actuation.
Calvert Cliffs 2	5/28/03	Auto Scram from 100% power. Turbine Trip Troubleshooting was in progress on main turbine governor.
Columbia	6/30/03	Turbine trip from 79% power - followed by Rx Scram. Reason for turbine trip unknown.

Manual Scrams

Clinton 1	4/11/03	Manual Scram from 30% -Turbine Vibration
Quad Cities 2	4/16/03	Manual Scram from 100% - Spontaneous opening of M.S. Relief Valve - Stuck Open- Alert declared - Torus Temperature greater than 95°F
North Anna 1	4/19/03	Manual Scram from 74% - Main Turbine lube oil leak

OPERATING EVENTS
APRIL - JUNE 2003

Limerick 1	4/23/03	Manual Scram on Low Reactor Level due to unexplained isolation of condensate deep bed demineralizes
Oyster Creek	5/20/03	Shutting down due to trip of 4160 Volt Bus resulting in loss of one train of safety equipment. D/G will not start because of fault of bus. Manually scrammed from 60%.
Harris 1	5/20/03	Manual Scram from 20% power due to loss of main feedwater pump.
	6/14/03	Repeat of above
Cooper 1	5/26/03	Manual Scram from 100% due to main turbine vibration
St. Lucie	6/11/03	During start up following refuel @ 30% power low power feed reg valve stem and disc separated cutting off flow. Low S/G level main F.W. Valve ByPass opened - High S/G level - Turbine Trip - Reactor Scram
Surry 1	6/13/03	Reactor manually Scrammed from 1% power on S/U following refueling due to indicators of misaligned control rod.

Other Interesting Things

D.C. Cook Both Units	4/24/03	Alert declared - Intake cooling water blocked by intake of fish Special Inspection Team dispatched.
Oconee 3	4/21/03	While shutting down for refuel one ADS isolation valve had bonnet leak and another could not be opened. Chronic problem with these valves on all 3 Oconee Units. Special Inspection Team dispatched.
Perry 1	4/24/03	Alert Declared - High Radiation Alarm in Fuel Handling Building. Bubble was seen floating to surface during handling operations.
Harris 1	4/28/03	Loss of S/D Cooling during period of reduced inventory and high decay heat Special Inspection Team dispatched.

OPERATING EVENTS
APRIL - JUNE 2003

North Anna	5/15/03	Old Reactor Head on its way to Utah involved in traffic accident in Kansas. Side swiped by drunk driver going 89 mph. Minor damage to covering. No contact with head. No radiation release.
River Bend	6/17/03	Operator removed wrong circuit breaker. Intended to work on Standby Service Water Pump Incorrectly removed breaker from H.P. Core Spray.
Perry	6/30/03	Unusual Event - Earthquake - Minor-no impact on equipment or operation - Plant continues at 100%
Brunswick 1	6/30/03	Tech Spec Shutdown - Unidentified leakage greater than 2 gpm (2.63 gpm)
MIT	6/30/03	Operator asleep at control

SECURITY

Columbia	5/6/03	20 Security Officers establish information picket line 10 miles from plant - Mediation in progress. No impact on operation (in refuel).
Oyster Creek	5/22/03	Work stoppage - Management manning workstations.
Pt. Beach	6/11/03	Security Officer - Discovered to have committed a criminal act.
Hatch	Current	Potential strike
St. Lucie	6/27/03	Inadvertent discharge of firearm
Sequoyah	7/2/03	Unaccounted for security weapon

METALLURGICAL ISSUES

South Texas 1	4/13/03	Reactor Vessel Bottom Head Degradation Indication of boron deposit @ Bottom Mounted Instrumentation (BMI) penetrations #1 and #46 - Further investigation
	5/20/03	Update - Axial indications in the nozzle wall near, but not in, welds of two BMI penetrations - Exam continues.

OPERATING EVENTS
APRIL - JUNE 2003

St. Lucie 2	4/30	VHP - unacceptable flaw detected. To be repaired prior to startup.
Oconee 3	5/2/03	Reactor Head being replaced. Indications of leakage at 2 CRDM penetrations on old head.
Cook 2	5/18/03	VHP - Crack indications on I.D. Not through wall. To be repaired prior to start up
Quad Cities 1	5/20/03	While shutting down for refueling outage, discovered leak in head vent line up stream of isolation valves
Quad Cities 2	6/12/03	Due to high moisture carry over in late May 2003, unit was shutdown, dryer inspected, and cracks and damage were found. Extensive repairs in progress. Expect unit back in early July 2003.

(11)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
504	9	9.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors

JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

9

Instructions to Preparer

1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person

JOHN T. LARKINS/

G:PlanPro(ACRS):ppmins.504
July 10, 2003

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
July 8, 2003

The ACRS Subcommittee on Planning and Procedures held a meeting on July 8, 2003, in Room T 2 B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 11:00 a.m. and adjourned at 12:25 p.m.

ATTENDEES

MEMBERS

M. Bonaca
G. Wallis
S. Rosen

ACRS STAFF

J. T. Larkins
S. Bahadur
H. Larson
S. Duraiswamy
R. P. Savio
J. Gallo
S. Meador
M. Snodderly
M. Weston
H. Nourbakhsh

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the July ACRS meeting

Member assignments and priorities for ACRS reports and letters for the July ACRS meeting are attached (pp. 7-10). Reports and letters that would benefit from additional consideration at a future ACRS meeting will be discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the July ACRS meeting be as shown in the attachment (pp. 7-10).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through October 2003 is attached (pp. 7-10). The objectives are to

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

During this session, the Subcommittee also discussed and developed recommendations on items included in Section II of the Future Activities List (pp. 11-14).

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate. The Committee should decide on the Subcommittee's recommendations on items in Section II of the Future Activities List.

3) Meeting with the NRC Commissioners

The ACRS is scheduled to meet with the NRC Commissioners between 9:30 and 11:30 a.m. on Wednesday, October 1, 2003, to discuss items of mutual interest. Topics proposed by the Planning and Procedures Subcommittee for this meeting are as follows:

I. Overview (MVB)

A) Differences in regulatory requirements between U.S. and other countries -- status report

B) Safeguards and security

C) Future ACRS activities

D) Risk-informing 10 CFR 50.46 and proposed 10 CFR 50.69

E) License renewal activities

F) Preapplication review of ESBWR design

G) Power uprate review standard

2

II Advancement of PRA technology in risk-informed decisionmaking ~~(GEA)~~ (TSK)

III Safety Culture ~~(GEA)~~/(GML)

IV Mixed Oxide Fuel Fabrication Facility (DAP)

The Planning and Procedures Subcommittee thought that these last two items could possibly be eliminated:

~~V Interim review of AP1000 design (TSK/GBW)~~

~~VI Reactor oversight process (JDS)~~

The Committee should select the topics during the July ACRS meeting. Subsequently, they will be sent to the Commission for approval. Since there is no ACRS meeting in August, the Committee should start preparing for the meeting with the Commissioners during its September meeting. To support this, cognizant members should complete the presentation slides in August.

The October ACRS meeting was previously scheduled for Thursday, October 2 through Saturday, October 4. Since the meeting now starts on Wednesday, October 1, the Committee should decide whether to have a four-day meeting (October 1-4, 2003).

RECOMMENDATION

The Subcommittee recommends that the Committee approve a list of topics for meeting with the Commissioners. Cognizant members should prepare slides and send them to the ACRS staff engineers by August 8, 2003. If needed, the October meeting should be scheduled through Saturday, October 4, 2003.

4) A Critical Review of the PIRT Process

The phenomena identification and ranking table (PIRT) process was originally formulated, as a major step in the code scaling, applicability and uncertainty (CSAU) evaluation methodology, to support a revised emergency core cooling system (ECCS) rule for light water reactors. This revised ECCS rule (10CFR 50.46) was issued in September 1988 and allows, as an option, the use of best estimate plus uncertainty methods in safety analysis. The CSAU evaluation methodology was developed to demonstrate the feasibility of the best estimate plus uncertainty approach. The objective of the PIRT process was to define plant behavior in the context of identifying the relative importance of systems, components, processes, and phenomena.

The PIRT process, with some variations, has been used in many more applications than was originally envisioned. These applications include development of experimental programs and safety analysis requirements for proposed advanced light water reactors, identification of thermal-hydraulic phenomena of importance to pressurized thermal shock (PTS) evaluation, assessment of the adequacy of the planned research programs in addressing the high burnup and new cladding alloy issues, support to resolution of Generic Safety Issues (GSIs) and providing technical guidance in allowing burnup credit (BUC) in the criticality safety analysis of spent fuel in transport and storage configurations. The NRC Office of Nuclear Regulatory Research also plans to use the PIRT process for identifying and prioritizing the research needs to develop regulatory infrastructure including data, codes and standards, and analytical tools in support of regulatory review of advanced reactor applications.

In view of widespread use of PIRT process and its role in prioritization of research needs to address reactor safety technical issues, it is important to provide lessons learned from the past several years of experience with the PIRT process and to identify potential improvements for future PIRT development. Dr. Nourbakhsh planned to provide a presentation to the Committee at the July 2003 ACRS meeting on this matter. The purpose of this presentation was to review the PIRT process and its prior applications and to provide some suggestions for enhancement of the process. Use of system dynamics techniques, such as influence diagrams, offers an attractive alternative for developing a phenomena identification and ranking table, which is the principal product of the PIRT process. The use of influence diagrams as a comprehensive framework to identify and prioritize the physical processes which need to be addressed for resolving a technical issue will also be discussed.

During its June meeting, the Committee agreed to hear a presentation by Dr. Nourbakhsh during the July meeting. Owing to lack of time, this presentation has been postponed to the September meeting. Dr. Nourbakhsh plans to send a report on this matter to the members prior to the September meeting.

RECOMMENDATION

The Subcommittee recommends that Dr. Nourbakhsh provide a presentation to the full Committee during the September 2003 ACRS meeting.

5) Comments on NUREG/CR-6813, Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decisionmaking

We recently published NUREG/CR-6813 prepared by Mr. Fleming under a contract with the ACRS/NRC. Mr. Lochbaum, Union of Concerned Scientists, has sent some comments on this report to the NRC Office of Public Affairs (OPA). Mr. Fleming prepared a response to Mr. Lochbaum, addressing every comment made by Mr. Lochbaum and sent it to Dr. Nourbakhsh. Mr. Lochbaum's comments and Mr. Fleming's response were e-mailed to all members by Dr. Nourbakhsh on May 5, 2003. The ACRS Executive Director e-mailed Mr. Fleming's response to OPA, NRR, and Mr. Lochbaum on May 5, 2003.

On July 7, 2003, the EDO has submitted comments on Mr. Fleming's report (pp. 15-17). In summary, the EDO states that:

The author of NUREG/CR-6813 identified some key issues that should be addressed to enhance the use of PRA for risk-informed decisionmaking. While the NRC staff is aware of and is addressing these key issues, the staff would like to note that it is not in full agreement with all of the characterizations of the current state of PRA technology and its use. In particular, the staff is not in full agreement with some of the author's views expressed in the report regarding the Davis-Besse vessel head degradation issue.

RECOMMENDATION

The Subcommittee recommends that Dr. Apostolakis propose a course of action with regard to responding to the EDO comments.

6) Meeting with the Executive Director for Operations

The members of the Planning and Procedures Subcommittee were previously scheduled to meet with the EDO and his deputies during lunch on Friday, June 13 to discuss items of mutual interest, including the following. This meeting is now scheduled to be held at noon on Friday, July 11, 2003:

- Differing views between the ACRS and the NRC staff on Reactor Oversight Process
- NRC staff process for tracking commitments made by the EDO/staff in response to ACRS comments and recommendations
- Timely submittal of documents for ACRS review
- Staff Requirements Memorandum on risk-Informing 10 CFR 50.46
- Safeguards and Security matters
- Comments on NUREG/CR-6813, Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decisionmaking
- Safety Culture

RECOMMENDATION

The Subcommittee recommends that the ACRS Chairman and other members of the Subcommittee provide a report to the Committee on the results of this meeting.

7) Member Issues

- Duration of July meeting (3 vs. 4 days)

  RECOMMENDATION

  The Committee should decide on Thursday, July 10 whether to extend the meeting through Saturday.

- Letter from Union of Concerned Scientists to Chairman Diaz

  Mr. Lochbaum, UCS, asked that the attached letter be forwarded to the ACRS regarding resolution of GSI-191 for Committee information and comment (pp. 18-22).

  RECOMMENDATION

  During its review of the proposed resolution of GSI-191, the Thermal-Hydraulic Phenomena Subcommittee should consider the issues raised by Mr. Lochbaum along with the staff's response.

- NRC System Simulation Capability

  Attached is a memo from V. Ransom regarding NRC System Simulation Capability (pp. 23-26).

  RECOMMENDATION

  Dr. Ransom should provide a brief report to the Committee during the July ACRS meeting outlining his concerns about the NRC System Simulation Capability.

ANTICIPATED WORKLOAD
JULY 9-11, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Weston	Safety Culture Report	A	To provide committee's views to the Commission	**Draft**
		Savio	Safeguards and Security	A	To provide early feedback to the Commission	**Draft**
Kress	--	Snodderly/ Duraiswamy	Proposed Criteria for Treatment of Individual Requirements in a Regulatory Analysis	A	To provide feedback to the staff	**Draft**
	Wallis	El-Zeftawy	ESBWR pre-application review	Report as needed	--	--
Leitch	--	Weston/ Caruso	Significant recent Operating events (South Texas)	--	--	--
Powers	--	Weston	Mixed Oxide Fuel Fabrication Facility	A	To provide early feedback to the Commission	--
Shack	Wallis	Snodderly	Expert Elicitation as directed by the Commission in the March 31, 2003 SRM related to risk-informing 10CFR 50.46.	Report as needed	--	--

ANTICIPATED WORKLOAD
SEPTEMBER 11-13, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly	Draft final Regulatory Guide DG-1122 on PRA quality	Report as needed	--	--
		Nourbakhsh	A critical Review of PIRT Process (Briefing by Dr. Nourbakhsh)	--	--	--
		Savio	Interim Report on Safeguards and Security **[Spent Fuel Fire, etc.]**	A	To identify issues of concern to the ACRS	--
Bonaca	All Members	Larkins	Preparation for meeting with the Commissioners (October 1)	--	--	--
	Leitch	Jain/Duraiswamy	Final review of St. Lucie license renewal application	A	To meet the CTM schedule	--
Kress		El-Zeftawy	Draft 10 CFR Part 52 Construction Inspection Program Framework	A	To provide early feedback	--
		El-Zeftawy	Interim review of the AP 1000 design	A	To identify issues of concern to the ACRS	--
		El-Zeftawy/ Snodderly	Framework for future nuclear power plant licensing **[Information Briefing]**	--	--	--
Powers	--	Sykes/Duraiswamy	Draft Guide 1105, Seismic Soil Liquefaction	--	--	--
Ransom	Wallis	Caruso	Draft final NRC review standard for review of core power uprate requests	A	To meet the CTM schedule	--
Rosen	--	Snodderly	Subcommittee Report on Fire Protection Issues	--	--	--

ANTICIPATED WORKLOAD
SEPTEMBER 11-13, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Sieber	--	Weston	Proposed Resolution of GSI-186, Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	A	To support staff schedule	--
Wallis	Ransom	Caruso	Draft final Reg. Guide DG-1107, Water Sources for Long-Term Recirculation Cooling Following a LOCA and Draft final Generic Letter 2003-xx, Potential Impact of Debris Blockage on Emergency Recirculation Design-Basis Accident at PWRs	A	To support staff schedule	--

ANTICIPATED WORKLOAD
OCTOBER 1-4, 2003

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Bonaca	All Members	Larkins	Meeting with the NRC Commissioners.	--	--	--
Kress	--	El-Zeftawy	Early Site Permit Review Standard.	A	To support the staff schedule	--
Powers		Jain/Duraiswamy	Draft Final Reg. Guide DG - 1099, Concrete Anchorage	A	To support the staff schedule.	--
		Sykes/Snodderly	Draft Guide DG-1109, Soil and Rocks.	--	--	--
		El-Zeftawy	Proposed 10 CFR Part 26, Fitness for Duty	--	--	--
Rosen	--	El-Zeftawy	Draft Final Regulatory Guide 1.168, Verification and Validation of Digital I&C Systems	A	To support the staff schedule	--
Ford	--	Jain/Caruso	Materials Degradation Programs	Report as needed	--	--
Leitch	--	Weston	Recent Operating Events	--	--	--
		Sykes/Duraiswamy	Subcommittee Report - H.B. Robinson License Renewal Application	--	--	--
Shack	--	Snodderly	Proposed Revision to Reg. Guide 1.53, Single Failure Criteria	Report as Needed	--	--

II. ITEMS REQUIRING COMMITTEE ACTION

1. Proposed Revision to Section 9.5.1, "Fire Protection Program," of the Standard Review Plan (Open) (SR/MRS)

Purpose : Determine a Course of Action

Review requested by the NRC staff [D. Frumkin, NRR]. In a May 13, 2003 memorandum, NRR requested agreement that re-review of a proposed revision to Section 9.5.1, "Fire Protection Program," of the Standard Review Plan (SRP) and Branch Technical Position SPLB 9.5-1 is not necessary since these documents are based on previously reviewed NRC practices and are consistent with Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants." Section 9.5.1 was last revised in July 1981. The revision gathers existing review guidance into one document for shutdown or decommissioned reactors, advanced reactors, license renewal, review criteria for PRA, and power uprates.

Mr. Rosen has previously recommended that the Fire Protection Subcommittee review this matter during its next meeting. Mr. Rosen participated in a teleconference with the staff to discuss a proposed agenda for the next meeting of the Fire Protection Subcommittee which has been tentatively scheduled for September 9, 2003. Mr. Rosen decided to reassess the need to review the proposed revision to Section 9.5.1 relative to higher priority fire protection issues, such as, an upcoming revision to 10 CFR 50.48, circuit analysis resolution, a revision of the fire protection significance determination process, crediting of manual actions, and fire dynamics spreadsheets.

The Planning and Procedures Subcommittee recommends that Mr. Rosen propose a course of action regarding this matter taking into account the competing priorities in the fire protection area.

2. Review of Draft NUREG Report on Updated SPAR HRA Methodology (Open) (DAP/MRS)

Purpose : Determine a Course of Action

Review requested by ACRS [P. O'Reilly, RES]. In a May 1, 2003 memorandum, RES provided a draft NUREG report on the updated human reliability analysis (HRA) methodology used in the Standardized Plant Analysis Risk (SPAR) models. This draft NUREG presents a simplified HRA method for estimating the human error probabilities associated with operator and crew actions and decisions in response to initiating events at nuclear power plants. This methodology is not meant to be a substitute for more detailed HRA approaches, such as those used in ATHENA. RES has requested NRR's and the Region's comments on the draft NUREG by July 1, 2003. They have not requested ACRS review but they did provide copies of the draft NUREG which have been distributed to the Members.

Dr. Powers proposes that the Reliability and PRA Subcommittee hold a meeting to discuss this document. Dr. Powers' note is attached.

3. TRACE Thermal-Hydraulic System Code (Open) (VR/RC)

Purpose: Determine a course of action

Review requested by V. Ransom. The NRC staff has recently released a new version of the TRACE thermal-hydraulic(T/H) system code to the T/H user community. This code, which was formerly known as the TRAC-M code, was developed as a successor to the RELAP5, TRAC-P, TRAC-M, and RAMONA computer models. It is intended to be used to evaluate the T/H system behavior of light water reactors during transients, and accidents, including especially, LOCAs. The code was developed to take advantage of new advances in computer hardware and software, and to ensure portability among platforms, flexibility to incorporate new technical insights, and maintainability, compared to the codes that it replaces. Now that the first version has been released to the user community, it would be opportune for the ACRS to review some of the technical innovations, and listen to the reaction of the user community.

In an email sent to the members on June 17, 2003, Dr. Ransom expressed concern that the NRC staff has not made a technical presentation to the Committee describing the TRACE code development activities. He noted that the staff had not used an NRC-developed tool to support licensing reviews of the SRELAP5 code, or the AP1000 review, and he was especially concerned that the ongoing code consolidation program was focused on maintained existing analytical capability, with little or no fundamental improvements to the technical capability. Dr. Ransom has agreed to take the lead in organizing this review.

It is envisioned that this review would include several subcommittee meetings and a full Committee meeting, and would result in the preparation of a letter describing the Committee's assessment of the state of the agency's T/H analytical capability.

The Planning and Procedures Subcommittee recommends that Dr. Ransom develop a plan, identifying issues and schedule for review and submit to the Planning and Procedures Subcommittee for review during its September 2003 meeting.

4. Review of Draft Final DG-1105, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites" (Open) (DAP/MRS)

Purpose: Determine a course of action

Review requested by the NRC staff [Y. Li, RES]. DG-1105, "Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites," provides guidance to license applicants on acceptable methods for evaluating the potential for earthquake-induced instability of soils resulting from liquefaction and strength degradation. It discusses conditions under which the potential for such response should be addressed in safety analysis reports. The guidance includes procedures and criteria to assess the liquefaction potential of soils ranging from gravel to clays. This final draft addressed public comments received, including those from Brigham Young University and Clemson University. NRR has reviewed this draft and agreed with this final version. OGC also reviewed the draft guide and had no legal objection. The Subcommittee may consider forwarding DG-1105 to the ACNW for their consideration.

The Planning and Procedures Subcommittee recommends that Dr. Powers propose a course of action.

7/2/03

Mike,

I believe the PRA subcommittee should review this report with the aim of presenting the mat'l, eventually, to the full committee. The issues to address

- comparison of the method to other methods of HRA and quantification

- validity of the method by comparison to experimental or experiential data

- applicability recommendations of the method.

- justifications for the various performance shaping factors – especially those that are different than those selected by other methods, those that are quite different than expected range of effects, and those not substantiated by a supporting data base.

It will, of course, be of interest to understand why a new HRA methodology had to be developed rather than adopting an existing, familiar methodology.

Dana

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 7, 2003

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: NUREG/CR-6813 "ISSUES AND RECOMMENDATIONS FOR ADVANCEMENT OF PRA TECHNOLOGY IN RISK-INFORMED DECISION-MAKING"

Dear Dr. Bonaca:

Thank you for your letter, dated April 29, 2003, which forwarded NUREG/CR-6813, "Issues and Recommendations for Advancement of PRA Technology in Risk-Informed Decision Making." We also received your letter dated May 16, 2003, which provides the ACRS's perspectives and recommendations on improving the quality of risk information for regulatory decision making including a consideration of your contractor's report. The May 16th letter addresses several important issues and we look forward to interacting with ACRS on them. We appreciate the ACRS's insights and will be providing a response to that letter in separate correspondence.

The author of NUREG/CR-6813 identified some key issues that should be addressed to enhance the use of PRA for risk-informed decision making. While the NRC staff is aware of and is addressing these key issues, the staff would like to note that it is not in full agreement with all of the characterizations of the current state of PRA technology and its use. In particular, the staff is not in full agreement with some of the author's views expressed in the report regarding the Davis-Besse vessel head degradation issue. The comments relative to this issue are included in the enclosure to this letter.

We value input on our programs and will be considering this report, as appropriate, in our ongoing efforts to risk-inform NRC regulations. We expect to identify and discuss specific areas of concern, as they become relevant, in future interactions with the ACRS.

Sincerely,

William Travers

William D. Travers
Executive Director
for Operations

Enclosure: Staff Comments on Davis Besse Vessel Head Degradation Discussion in NUREG/CR-6813

cc: Chairman Diaz

Commissioner McGaffigan
Commissioner Merrifield
SECY

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Staff Comments on Davis-Besse Vessel Head Degradation Discussion in NUREG/CR-6813

In April 2003, the Advisory Committee on Reactor Safeguards (ACRS) issued NUREG/CR-6813 which presents the views of the author, Karl Fleming, including specific insights he drew from the risk-informed evaluation of the Davis Besse vessel head degradation. Section 3.1 of the report describes the staff's decision to allow the licensee to delay the vessel inspection from the December 31, 2001, goal until a planned outage to commence by March 31, 2002. Mr. Fleming uses the evaluation conducted by the licensee to illustrate his views on the implications of knowledge and modeling uncertainties. While we agree that it is important to consider uncertainties in safety decisions, we disagree with the characterization of the Davis-Besse decision. This characterization does not accurately describe the basis for the staff's decision to permit continued plant operation.

At the time of the Davis-Besse decision, the staff was well aware of the corrosive effects of boric acid on carbon steel. The Oconee plant first reported through-wall leakage from a control rod drive mechanism (CRDM) in early 2001, and the related accumulation of boron crystals on the vessel head. No corrosion was found on the Oconee vessel head. The Davis-Besse licensee stated that they had cleaned the vessel head prior to entering into the last operating cycle. Based on this experience, the staff concluded that, because the vessel head is operated at very high temperatures, coolant leakage through any CRDM cracks would evaporate, leaving dry boron crystals on the vessel head. Other plants have found similar evidence of CRDM leakage and dry boron crystals. There has been no evidence that the presence of the dry boron crystals has caused any corrosive wastage. These licensees informed us that, where the dry boron crystals have been found, the vessel heads had been cleaned prior to entering the next operating cycle.

In the fall of 2001 the licensee for Davis-Besse responded to Bulletin 2001-01 and informed the staff of their intent to operate beyond the December 31, 2001, targeted completion date for vessel inspections. In that submittal, the licensee initially presented a variety of technical information to support their plans to perform the inspection during an outage scheduled to begin on March 31, 2002. During subsequent questioning by the staff, the licensee informed the NRC that they had revised the outage schedule to begin on February 16, 2002. In their response to the bulletin, the licensee stated that "[t]he RPV head area was cleaned with demineralized water to the greatest extent possible while maintaining the principles of As-Low-As-Reasonably-Achievable (ALARA) regarding the dose."

On this basis, the staff evaluation focused on the potential for CRDM cracking and the increased likelihood that such cracking would result in a loss-of-coolant accident during the period between December 2001 and February 2002. As described in the detailed safety evaluation dated November 5, 2002, the staff concluded that the licensee provided sufficient justification to support the extended period of operation, as follows:

> "The staff concluded that the small increase in LOCA probability, the low conditional core damage probability and the low conditional containment failure probability meant that defense in depth was preserved, although leakage from the reactor coolant pressure boundary was likely. The staff further concluded that, while the structural margin of some CRDM nozzles was believed to be

Enclosure

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significantly reduced, sufficient margin remained to maintain safety and prevent a LOCA. The staff concluded that the change in core damage frequency due to the potential for CRDM nozzle ejection was consistent with the guidelines of R.G. 1.174. On December 4, 2001, the staff issued a letter to the licensee approving the proposed nozzle inspection plan submitted in their response to Bulletin 2001-01."

When the licensee shutdown on February 16, 2002 and examined the vessel head for CRDM cracking, their findings were consistent with the staff's evaluation of the safety margins associated with the cracking. The licensee's discovery of the extensive corrosion on the vessel head, resulting from accumulated boron on the vessel head over a period of years, was unexpected. The staff evaluation specifically points out that ... "[h]ad the NRC been aware of this degradation, we would have reached a very different conclusion in November."

The Fleming report characterizes the basis for the staff's decisions in terms of "the same naive modeling assumptions" that were used in the previous evaluations of the susceptibility to CRDM cracking, and goes on to state that "[t]he validity of all previous evaluations ... of Alloy 600 nozzles is now open to question." For the reasons stated above, we do not believe that this is a fair representation of the Davis-Besse decision or the staff's deterministic and probabilistic evaluations of CRDM cracking. In fact, Mr. Fleming points out a risk evaluation that included the potential for vessel head corrosion would likely reach the same conclusion regarding the risk of continued operation. On this basis, we respectfully disagree with the characterization of the Davis-Besse decision making process in NUREG/CR-6813.

From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: "Carol Rowe" <CAR1@nrc.gov>
Date: 7/2/03 4:49PM
Subject: Fw: UCS letter to NRC regarding Bulletin 2003-01

Carol,

add for discussion at P&P
----- Original Message -----
From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: <wjshack@anl.gov>; "Vic Ransom" <ransom@ecn.purdue.edu>; <TSKress@aol.com>; "Steven Rosen" <historyart@computron.net>; <JDSieber@aol.com>; "Graham Wallis" <Graham.B.Wallis.@dartmouth.edu>; "Graham Leitch" <gmleitch@aol.com>; <FPCTFord@aol.com>; "Dana Powers" <dapower@sandia.gov>; <apostola@MIT.EDU>
Cc: <SXD1@nrc.gov>; <SXB@nrc.gov>; "John Larkins" <JTL@nrc.gov>
Sent: Wednesday, July 02, 2003 4:51 PM
Subject: Fw: UCS letter to NRC regarding Bulletin 2003-01

> John,
>
> Let's discuss this issue at P&P
>
> Mario
> ----- Original Message -----
> From: "Paul Blanch" <pmblanch@attbi.com>
> To: "Mario Bonaca" <mvbonaca@snet.net>
> Cc: "Dave Lochbaum" <dlochbaum@ucsusa.org>
> Sent: Tuesday, July 01, 2003 9:56 AM
> Subject: FW: UCS letter to NRC regarding Bulletin 2003-01
>
>
> > Mario:
> >
> > Dave asked me to forward this to you as Chairman of the ACRS.
> >
> > Paul M. Blanch
> > 135 Hyde Rd.
> > West Hartford, CT 06117
> > Cell 860-881-6011
> > Office 860-236-0326
> > FAX 801-991-9562
> >
> >
> > -----Original Message-----
> > From: Dave Lochbaum [mailto:dlochbaum@ucsusa.org]
> > Sent: Tuesday, July 01, 2003 9:45 AM
> > To: opa@nrc.gov
> > Cc: jgl1@nrc.gov; JIZ@nrc.gov; jxl4@nrc.gov; rea@nrc.gov; SRB3@nrc.gov
> > Subject: UCS letter to NRC regarding Bulletin 2003-01
> >
> > Good Day:
> >
> > UCS mailed the attached letter to NRC Chairman Diaz this morning.
> >

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> > We are concerned that the NRC's interim fix to the PWR containment
> > sump screen issue is ill-advised and dangerous. We advocate stopping
> > all work on Bulletin 2003-01 and redirecting that effort to closure
> > of GSI-191, which is the right way to solve the problem once and
> > for all.
> >
> > Thanks,
> >
> > Dave Lochbaum
> > Nuclear Safety Engineer
> > Union of Concerned Scientists
> > 1707 H Street NW Suite 600
> > Washington, DC 20006-3962
> > (202) 223-6133 x113
> > (202) 223-6162 fax
> >
> >
>

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

July 1, 2003

Dr. Nils J. Diaz, Chairman
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REGULATORY PRUDENCE AND BULLETIN 2003-01

Dear Chairman Diaz:

The Union of Concerned Scientists (UCS) finds itself in the unaccustomed position of asking you to curb your staff's efforts on a nuclear safety issue. They have, as we have, the very best intentions on this matter. But we are extremely concerned that the NRC staff is "jumping the gun" on this matter and are pursuing actions that may be adverse to safety. We ask you to redirect them towards the prudent and proper resolution of this matter.

The issue is the containment sump screens for pressurized water reactors (PWRs). Generic Safety Issue (GSI-191) was opened years ago to address this issue. The matter was accelerated to the fast track recently by the discoveries at Davis-Besse that (a) there was ample debris inside containment to clog the containment sump screen following design basis loss of coolant accidents, (b) that sufficient debris would remain inside containment even after extensive foreign material exclusion (FME) efforts, and (c) gaps existed in the screen to allow passage of debris large enough to challenge functioning of the containment spray nozzles and high pressure injection pumps. The containment sump screen at Davis-Besse was made larger by more than an order of magnitude.

For the express purpose of reducing risk until formal containment sump screen evaluations could be completed for other PWRs, the NRC staff issued Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors," on June 9, 2003. The bulletin made PWR owners aware of the sump blockage issue and requested them to consider taking interim measures to reduce the risk while the necessary formal containment sump screen evaluations were completed and any applicable corrective actions implemented. The NRC staff met with industry representatives and other stakeholders on June 30, 2003, to discuss the bulletin.

I attended the June 30th meeting and was extremely troubled by the approach outlined by the NRC staff for the proposed interim measures. The staff is urging plant owners to implement the interim measures without doing the "homework" necessary to ensure that the net effect will not be increased risk. It is my considered opinion that many of the proposed interim measures can actually increase risk quite substantially. Even if the proposed interim measures reduce risk, the NRC staff is not requiring the "homework" needed to verify this desired outcome.

Washington Office: 1707 H Street NW Suite 600 • Washington DC 20006-3919 • 202-223-6133 • FAX: 202-223-6162
Cambridge Headquarters: Two Brattle Square • Cambridge MA 02238-9105 • 617-547-5552 • FAX: 617-864-9405
California Office: 2397 Shattuck Avenue Suite 203 • Berkeley CA 94704-1567 • 510-843-1872 • FAX: 510-843-3785

For example, the bulletin urges plant owners to consider providing alternative water sources for the reactor core cooling and containment spray pumps in event the containment sump becomes unavailable. In written responses to industry questions provided during the June 30th meeting, the NRC staff indicated that "non-qualified components and non-Regulatory Guide 1.97 instrumentation may be used." In oral answers to the same questions, the NRC staff indicated that plant owners need not apply the single failure criterion when selecting alternative water sources. In its written response to industry question #38, the NRC staff stated "Many PWRs have margin between the current maximum pool depth [inside containment] and the depth which could result in a loss of integrity or structural failure of the containment. Therefore, although containment overfill should be a concern, many plants would likely be able to provide some quantity of additional injection from alternative sources without jeopardizing containment integrity." The NRC staff is implicitly conceding that containment failure could result from overfilling the containment using alternative water sources. Yet, the NRC staff is not applying the protection necessary to prevent inadvertent overfilling. By using non-qualified components not subject to the single failure criterion, it is credible that injection using an alternative water source could be initiated and not stopped when necessary.

The bulletin also invites plant owners to adopt interim measures to preclude debris accumulation on things inside containment, like mesh doors, which could prevent adequate water inventory in the containment sump. In oral answers to questions at the June 30th meeting, the NRC staff suggested it might be advantageous to leave doors to high radiation areas open with the plant at power, contrary to many lessons learned from radiation over-exposure events in the past. As the staff indicated in its written response to industry question #16, "Given that the response period for Bulletin 2003-01 is 60 days, the staff is not expecting lengthy, detailed analysis." In other words, neither the plant owner nor the NRC inspectors will have sufficient grounds to determine if leaving radiation protection doors unlocked and open is necessary, yet along worth the risk of personnel overexposure.

Perhaps the worst part of the bulletin is its suggestion that plant owners deliberately turn off one train of reactor core cooling and/or containment spray so as to minimize the potential for clogging the containment sump screens with debris. But as the staff stated in its written response to industry question #66, "these calculations [of sump screen clogging] are plant-specific and the actual results may be counterintuitive. As an example, for certain plants with partially submerged sump screens during a small-break LOCA, this accident may present the greatest NPSH margin challenge." It is dangerous and imprudent to direct operators to turn off a safety train without good reason. Absent the plant-specific calculation, such direction is wrong. It only becomes right when a plant-specific calculation indicates that it is necessary. When one licensee at the June 30th meeting reminded the NRC staff that his plant got approval for the alternate source term on the basis of having both containment spray loops operating, the NRC staff's oral response was, in essence, "it's only an interim measure, wing it."

The interim compensatory measures sought by the NRC staff in Bulletin 2003-01 will set up the operators at PWRs more than the "solid pressurizer avoidance" mindset prior to TMI if they are not backed by formal evaluations to verify overall risk is being reduced.

There are many more reasons why the proposed interim measures may have unintended adverse safety implications. UCS is troubled that the NRC staff is urging plant owners to take steps that may undermine safety.

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UCS is also concerned that the NRC staff and industry resources devoted to Bulletin 2003-01 are themselves adverse to safety. If those efforts delay the proper resolution of GSI-191 by even a second, they will have been counter-productive. The prudent and proper way to address the containment sump screen issue is to devote full attention to the resolution of GSI-191. Bulletin 2003-01 is therefore a needless and counter-productive distraction for both the NRC staff and industry (and also for UCS, but we'll manage.)

UCS requests that the NRC staff terminate Bulletin 2003-01 now and redirect its efforts towards expeditious resolution of GSI-191. Resolution of GSI-191 entails the rigor and discipline necessary to prevent the correction of one problem from inadvertently creating another larger problem. It is the right way to handle the containment sump screen issue.

The NRC has four strategic goals: (1) maintain safety, (2) increase efficiency and effectiveness, (3) improve public confidence, and (4) reduce unnecessary regulatory burden. Continuing forward with Bulletin 2003-01 meets none of these goals. Stopping Bulletin 2003-01 now and accelerating resolution of GSI-191 meet all four. Please help the NRC staff to do the right thing.

Sincerely,

<ORIGINAL SIGNED BY>

David Lochbaum
Nuclear Safety Engineer
Washington Office

From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: "Carol Rowe" <CAR1@nrc.gov>
Date: 7/2/03 4:49PM
Subject: Fw: Concern about NRC System Simulation

For discussion at P&P
----- Original Message -----
From: "Mario V. Bonaca" <mvbonaca@snet.net>
To: <Graham.B.Wallis@Dartmouth.EDU>; <SXD1@nrc.gov>; <SXB@nrc.gov>; "Steven Rosen" <historyart@computron.net>; "John Larkins" <JTL@nrc.gov>
Sent: Wednesday, July 02, 2003 5:04 PM
Subject: Fw: Concern about NRC System Simulation

> Gentlemen,
>
> Vic Ransom wrote this thoughtfull expression of concern with the NRC T-H
> capabilities. Some of us are left with the question, is this concern
> justified? I would like to discuss it at the P&P, and get Graham's
> perspective on this issue. As a minimum we need to open it up for
Committee
> discussion. This is a concern raised by a knowledgeable member of the T-H
> subcommittee, thus has a degree of credibility.
>
> Mario
>
>
> ----- Original Message -----
> From: "Victor H. Ransom" <ransom@ecn.purdue.edu>
> To: <mvbonaca@snet.net>; <FPCTFord@aol.com>; <apostola@mit.edu>;
> <TSKress@aol.com>; <gmleitch@aol.com>; <dapower@sandia.gov>;
> <historyart@computron.net>; <wjshack@anl.gov>; <jdsieber@aol.com>;
> <graham.b.wallis@dartmouth.edu>; <ransom@ecn.purdue.edu>
> Cc: <jtl@nrc.gov>; <rxc@nrc.gov>
> Sent: Tuesday, June 17, 2003 3:12 PM
> Subject: Concern about NRC System Simulation
>
>
> > See the attachment for my comments on NRC System Simulation Capability
> >
> > Vic
> >
>
> --
> ---
> > Victor H. Ransom, Professor Emeritus
> > Member USNRC Advisory Committee on Reactor Safeguards
> > School of Nuclear Engineering
> > Purdue University
> > West Lafayette, IN 47907-1290
> > (currently located at 1454 S. Woodruff Ave., Idaho Falls, ID 83404)
>
> --
> ---
> > TEL (208) 528 7934 Idaho Falls, ID "Empty your purse into your
head

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To: ACRS Members
Fr: V. H. Ransom
Re: NRC System Simulation Capability
Cc: Dr. John Larkins, Dr. Ralph Caruso

Introduction.

I am concerned about the state of the NRC LWR system simulation capability for reasons that I will elaborate on. Deterministic simulation of LWR system behavior for postulated scenarios is a key part of both probabilistic risk analysis and conservative deterministic analysis. In either case we depend on analytical models to predict whether core damage will occur for any assumed system transient. Historically, at least in the US, the NRC system simulation capability has been the standard, or audit, against which the acceptability of industry analyses have been judged. In the not too distant past the NRC LWR system simulation capability was the world standard. This is no longer the situation and at present I have concerns that the current NRC Research effort to improve system simulation capability may be faltering. Since I have joined the ACRS, no substantive technical information has been provided to or, to my knowledge, requested by the ACRS or the ACRS Thermal-Hydraulics Subcommittee concerning either the maintenance of the RELAP5 code or the development of the TRAC-M code (now called TRACE). This is disturbing, especially in view of the rigorous scrutiny to which we subject vendor and utility methods for licensing application, yet we have made no comparable scrutiny of the NRC system simulation capability. With the move to risk informed regulation it is even more important to have simulation audit capability with quantified and as low as reasonably achievable uncertainty.

BACKGROUND

Evidence of the poor state of NRC system simulation was apparent in several ACRS reviews during this past year. During the NRC review of the Framatome ANP SRELAP5 code no NRC audit calculations were performed which could be used to judge the adequacy of SRELAP5 code for regulatory applications. In spite of this lack of a standard for comparison, and the apparent concerns with system models, the SRELAP5 code was approved for use in large break LOCA analysis. I did not disagree with this approval, mainly because of my familiarity with the code progenitor, but it would have been reassuring to have seen a comparison of results to an NRC standard. In the ACRS review of the NRC work on the PTS rule evaluation, several deficiencies in the RELAP5/MOD3 assessment for applicability to PTS transients were noted, but no effort was made by NRC to identify the source of these deficiencies nor were any plans presented for how these deficiencies would be resolved. Recently Westinghouse presented system simulation results for the AP1000 using rather simplistic methods and again no NRC audit results were available that could be used to judge the adequacy of these methods. Presumably, these same methods will be used by Westinghouse in its licensing application for the AP1000.

The NRC RES plan for remedying this situation is the development of the TRAC-M or TRACE code which is to combine the features of RELAP5, TRAC-P, and TRAC-B. Planning for this effort was initiated in 1995 by formation of a team of consultants, including the author, each of whom was asked to prepare a summary report on recommendations for how the NRC might improve system simulation capability. This activity was culminated with a meeting in which each consultant presented his views. Thereafter the consultants, at least as a group, played no role in the final plan. I do not know if any of the group recommended the approach taken, but I did not. To my knowledge, this group was never convened to review the plan or for peer review of the project as it progressed. The code consolidation project was initiated sometime after 1995 and plans for RELAP5 maintenance as well as plans for the code consolidation effort were presented at the NRC sponsored RELAP5 Users Meeting held June 16-17, 1997 at Annapolis, Maryland. At that time the transition to the new code was scheduled for 2002 and to be concluded in 2003. A year ago, June 26, 2002, the Thermal-Hydraulic and Severe Accident ACRS Subcommittees heard a presentation by Joseph M. Kelly of RES in which plans and schedules were presented for the TRAC-M development during the 2002 – 2007 time frame. No technical detail or results were presented. More recently, the NRC EDO announced on March 17, 2003 in a letter to the commissioners that the code consolidation effort was closed in FY2002. However, we have yet to see any calculations made with TRAC-M or any discussion of the details of the model formulation, e.g. the momentum treatment. To my knowledge, no peer review by personnel outside the NRC or the project team has been conducted for this project.

It is no surprise to me that this project is taking longer than intended, but it is certainly time that the ACRS be apprised of the technical status of the project. In the past the formulation of models for system simulation has been a contentious issue and the NRC RES has used periodic peer reviews by knowledgeable individuals from national laboratories, universities, and industry to assure that such work was on the right track, no pun intended.

The bits and pieces of information that I have encountered, indicate that most of the consolidation effort has gone into restructuring and reprogramming of TRAC-P and adding components to TRAC-P to make the TRAC topology consistent with the RELAP5 topology for system representation. The motivation for this is the fact that the TRAC-P code was not widely used and the RELAP5 topological representation has a long history of development dating from FLASH and early RELAP codes of the 1960s. The TRAC codes have not been widely used in part due to the use of rather crude input schemes. The input uses the FORTRAN NAMELIST routines with little or no checking of input data for consistency. However, because of the heavy industry and NRC investment in RELAP5 plant models, it was a necessity that the consolidated code be able to use RELAP5 plant models. The same crude TRAC input methods have been retained in TRAC-M , however, a second code called SNAP has been developed by the NRC to provide a graphical user interface (GUI) that can be used with RELAP5 or TRAC-M and the topological models that were unique to RELAP5 have been added to TRAC-M. With these changes the SNAP code has or will have the ability to translate existing RELAP5 models to TRAC-M. While SNAP does not embody any physical models, it may be of interest to the ACRS, because it is the basic tool for creating and modifying system models. It would be interesting to know if SNAP embodies input checking for consistency, e.g. a loop elevation closure check.

Perhaps the most disturbing aspect of the code consolidation effort is that most of the effort has been devoted to recreating capability that existed, albeit in separate codes. As far as I know, there has been little or no fundamental improvement in the NRC simulation capability. The RELAP5 maintenance effort has been minimally supported and at present is one FTE. In spite of this several improvements have been made to RELAP5 that were subsequently installed in TRAC-M. There are fundamental modeling deficiencies in all the thermal hydraulic codes that need resolution in order to produce more robust simulation capability. A long term plan should address resolution of these modeling issues.

Since being in Idaho Falls, I have visited the INEEL code development group. They have not been involved in the NRC system code development or maintenance since 1997 when the work was transferred to Scientech, apparently because of Bechtel conflict of interest. The INEEL have continued the development of RELAP5 and ATHENA under DOE support. The ATHENA code was produced as an off shoot of the RELAP5 code beginning prior to1985 under DOE sponsorship for fusion safety applications. Its main distinction is that different fluids can be used in each separate loop, but with thermal interaction. This code has been used in many liquid metal and gaseous fluid applications. The RELAP5-3D code development now has the capability to model multi-dimensional components in either cylindrical or Cartesian coordinates, either 2D or 3D. It also can use multiple 3D components in a system model. These codes are planned for use by DOE in the Gen IV projects. It is unfortunate that the NRC and the DOE efforts have proceeded with little or no interaction.

CONCERNS

My fundamental concern is that the NRC system simulation capability has been more or less stagnant at what it was six years ago judging from applications that we have seen this past year. As I understand the situation, even when TRAC-M is ready for application, there will be little or no new or improved capability, only another suite of codes that combine the capability of two or three existing codes into two codes, TRAC-M and SNAP. This situation is especially disconcerting when the NRC and ACRS are advocating risk informed regulation. This move will require that the uncertainty associated with system simulations be determined. NRC research needs to address this aspect of system simulation and how this capability is to be incorporated into the TRAC-M code. At the same time we have seen applications in which the current methods have unexplained anomalies compared to data and these may carry over to TRAC-M. I do not know of any effort to address these inadequacies. Further, there are several modeling issues, e.g. Tees or branches, that have never been resolved in an entirely satisfactory manner in any of the thermal hydraulic modeling codes. Even with scarce resources some effort needs to be devoted to resolution of such issues. I believe the ACRS needs to take a more proactive approach to assessing the direction and adequacy of the NRC system simulation development efforts. If for no other reason than to be aware of what is being done and what can be expected of these efforts.

G:EDOMTG:7/11/03

PLANNING AND PROCEDURES SUBCOMMITTEE MEETING WITH THE EDO
FRIDAY, JULY 11, 2003 - NOON

1. Differing Views Between the ACRS and the NRC Staff on Reactor Oversight Process

- December 20, 2001 SRM stated that:

  The staff, with ACRS input, should provide recommendations for resolving, in a transparent manner, apparent conflicts and discrepancies between aspects of the ROP that are risk informed (e.g., SDP) and those that are performance based (e.g., Pis).

- February 13, 2002 ACRS Letter to the EDO:

  ACRS continues to believe that some of the threshold values for risk-based PIs are not meaningful. It is important that the thresholds adequately reflect the levels at which NRC will take action and the urgency with which this action will be taken.

- March 13, 2003 ACRS Report to NRC Chairman Meserve:

  Made 8 recommendations. The staff agreed with all but 2 noted below.

  - It is incorrect to base thresholds for PIs on risk metrics such as ΔCDF and ΔLERF.

  - The thresholds separating all performance levels (colors) should be performance based and determined by expert judgment similar to the selection of the current green/white thresholds.

  April 29, 2003 EDO response stated that based on experience to date, the staff believes that the PI thresholds are providing the necessary information to enable the staff to make informed decisions and take appropriate actions. Therefore, the staff does not plan to make any near-term changes to the PI thresholds, but will continue to monitor and assess these thresholds as part of the ongoing self-assessment activities.

- **Obvious disagreement between the ACRS and the NRC staff. Where do we go from here? How do we resolve this long-standing difference?**

2. NRC Staff Process for Tracking Commitments Made by the EDO/Staff in Response to ACRS Comments and Recommendations

- Concern raised by some Commissioners regarding lack of a transparent process to track EDO/Staff commitments.

- When the staff agrees with the ACRS comments and recommendations or agrees to revise a document in response to ACRS recommendations, how is it ensured that it has been done?

- **EDO should have a mechanism to keep track of the resolution of commitments.**

3. Timely Submittal of Documents for ACRS Review

- Timely submittal of documents for ACRS review has been slowly improving. Still there are some problems in this area that needs to be dealt with to enhance the process.

- Examples of documents not provided in a timely manner include the following:

  - Draft final Regulatory Guide DG-1122, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." [Even though the staff promised to provide the ACRS with relevant documents on March 18 for review by the Committee during its April 10-12, 2003 meeting, the draft final guide was not provided even during the April meeting. The Committee ended up reviewing the public comments, associated staff resolution, and proposed changes to the Regulatory guide. The Committee wrote a letter on the resolution of public comments and recommended that the staff submit the draft final regulatory guide to the ACRS prior to issuing for trial use. Reviewing one document twice is resource intensive to the ACRS and the staff.]

The following documents were provided to the Committee 10 days prior to the meeting:

- Draft final Regulatory Guide DG-1119, "Guidelines for Evaluating Electromagnetic and Radio-frequency Interference in Safety-Related Instrumentation and Control Systems."

  - Draft Regulatory Guide DG-1115, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors."

  - Generic Letter 2003-XX, "Control Room Habitability."

  - NEI 99-03, Rev. 1, "Control Room Habitability Guidance."

- Draft Final Regulatory Guide DG-1077, "Guidelines for Environmental Qualification of Microprocessor-Based Equipment Important to Safety."

Providing very voluminous documents to the Committee 10 days before the meeting will impede the Committee's ability to perform a meaningful review. The EDO coordinator should take efforts to minimize the recurrence of this problem.

Typically the NRC staff factors ACRS review in their schedule, however, as schedules change, ACRS staff should be told.

4. <u>Staff Requirements Memorandum on Risk-Informing 10 CFR 50.46</u>

- March 31, 2003 -- SRM directed the staff to prepare a proposed change to 10 CFR Part 50 that allows for a risk-informed alternative to the present maximum LOCA break size. The technical basis supporting the LB-LOCA size redefinition, supported by a 10-year estimation of LOCA frequencies, is to be provided to the Commission by March 31, 2004.

- June 9, 2003 - Staff held public workshop to solicit the perspectives of external stakeholders to assist in its response to the Commission's directives. The objective of the meeting was to identify areas where a common understanding of the Commission's SRM is needed in order to develop a proposed rule. NEI identified the following five areas as needing further discussion: (1) rule attributes, (2) PRA scope, (3) definition of "best estimate," (4) nature of new design basis analysis, and (5) scope of allowed changes, including reversibility due to 10 year re-estimation of LOCA frequency. The staff presented a more detailed list of issues and implications that, in general, covered the five areas identified by NEI.

- July 2003 - Follow-up public workshop to discuss draft positions of the areas identified from the June 9, 2003 public workshop. The objective of the meeting is develop a common understanding of the Commission's March 31, 2003 SRM (The ACRS is interested in the draft positions taken by NEI and the staff on the identified areas.]

- July 9, 2003 - ACRS to be briefed on Expert Elicitation being conducted by the staff in support of the LB-LOCA size redefinition.

- **ACRS is interested in early interactions with the staff on the development of the draft Commission paper in response to the Commission's March 31, 2003 SRM.**

5. <u>Safeguards and Security Matters</u>

- The Committee met with the NSIR and RES staffs in May 2002 to obtain an initial information briefing on the NSIR and RES programs. The Safeguards and Security Subcommittee subsequently met with the staff in October 2002, April

Executive Director e-mailed Mr. Fleming's response to OPA, NRR, and Mr. Lochbaum on May 5, 2003.

The EDO plans to submit comments on Mr. Fleming's report. RES has the lead in gathering the comments, including those from NRR and provide them to the EDO for transmittal to Dr. Bonaca.

7) <u>ACRS held a Workshop on Safety Culture during its June 2003 full Committee meeting</u>.

– The purpose of the Workshop was to discuss initiatives, methodologies, guidelines, and adopted approaches for safety culture. The workshop was organized into two panels: (1) to develop a collective understanding of safety culture, and (2) to discuss the attributed of safety culture.

– The panel had NRC staff and management, Vice-President of NEI, Tom Murley, Howard Whitcomb, William Kiesling, Dave Collins, Alan Poise (Vice-President Dominion Nuclear Connecticut, George Felgate, INPO, Lew Meyers (COO of 1st Energy Nuclear), William O'Connors (Vice-President Detroit Edison) and Songor Haber.

– ACRS in the process of finalizing a report that among other things notes the existing regulations allowed for the monitoring of aspects of safety culture that are appropriate for the NRC.

– NRC approach to safety culture is appropriately performance based

– Other issues still being debated