UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 14, 2004

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

SUBJECT: SUMMARY REPORT - 512th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, MAY 5-8, 2004, AND OTHER RELATED ACTIVITIES OF THE COMMITTEE

Dear Chairman Diaz:

During its 512th meeting, May 5-8, 2004, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following letters:

LETTERS:

Letters to William D. Travers, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Use of Mixed Oxide Lead Test Assemblies at the Catawba Nuclear Station, dated May 7, 2004
- Good Practices for Implementing Human Reliability Analysis, dated May 13, 2004
- Resolution of Certain Items Identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria," dated May 21, 2004

HIGHLIGHTS OF KEY ISSUES

1. Safeguards and Security Matters

The Committee heard presentations by and held discussions with representatives of the Office of Nuclear Regulatory Research regarding safeguards and security matters. This meeting was closed to protect information classified as national security information as well as unclassified safeguards information pursuant to 5.U.S.C. 552b(c)(1) and (3).

Committee Action

The Committee plans to hold additional discussions with the NRC staff and its contractors in August/September 2004 to discuss security issues related to reactors, fuel cycle facilities, spent fuel cask storage, and emergency response planning. The Committee plans to provide reports to the Commission on these topics in the future.

2. Use of Mixed Oxide (MOX) Lead Test Assemblies at the Catawba Nuclear Station

The Committee heard presentations by and held discussions with representatives of Duke Power, the Union of Concerned Scientists (UCS), the Nuclear Energy Institute (NEI), and the NRC staff regarding the Duke Power's application to irradiate four MOX fuel lead test assemblies (LTAs) in the core of one of the reactors at the Catawba Nuclear Station. Representative of Duke Power presented information about the experience base elsewhere in the world with the fabrication and use of MOX fuel in commercial reactors. The NRC staff presented its evaluation of the key safety issues, which centered on fuel assembly performance, and changes to the accident source term arising from the use of MOX fuel LTAs. The UCS representative expressed concerns related to the behavior of MOX fuel during design-basis and beyond-design-basis accidents, that the UCS believes has not been appropriately treated by Duke, or the staff. The representative from NEI commented in support of the application.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated May 7, 2004 concluding that, under the restricted circumstances considered in both the Duke Power application and the NRC staff's safety evaluation, the four MOX fuel LTAs can be irradiated in non-limiting locations in either of the cores of the Catawba reactors with no undue risk to the health and safety of the public.

3. Risk Management Technical Specifications

The Committee heard presentations by and held discussions with representatives of the Office of Nuclear Reactor Regulation (NRR), NEI, and the South Texas Project (STP) regarding the status of the Risk Management Technical Specifications (RMTS), Initiative 4b, Risk Informed Completion Times. The purpose of this project is to risk-inform the technical specifications. Initiative 4b is intended to extend the completion times from a current nominal value to a predetermined maximum using configuration risk management. The staff is currently reviewing a draft guidance document from NEI and pilot proposals from the STP and Fort Calhoun. Hope Creek Plant has also volunteered to be a pilot. RMTS is dependent upon a robust and quality PRA. Communication and training of headquarters and regional staff are essential. Some issues associated with this project are the extent of incorporation of risk monitors and assessment tools into the PRAs, QA/QC of the software and its updates, and the time necessary to calculate the risk.

Committee Action

This was an information briefing and no Committee action was taken.

4. Trial/Pilot Implementation of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"

The Committee met with representatives of the NRC staff and NEI to discuss the current activities and plans related to the five pilot applications of Regulatory Guide (RG) 1.200. NEI provided its perspective on the pilot applications of RG 1.200. In a September 22, 2003 report to the Commission, the Committee agreed with the staff's recommendation that RG 1.200 be issued for trial use with an appropriate sample of pilot plants. The staff and NEI discussed what they hoped to learn from the pilot applications. Both the staff and NEI said that applying RG 1.200 has been more intensive than anticipated.

Committee Action

This was an informational briefing. The Committee plans to review Appendix C to RG 1.200, which will endorse the American Nuclear Society (ANS) Standard on external events. Also, the Committee plans to review the proposed revision to RG 1.200, which will incorporate the lessons learned from the trial applications.

5. Good Practices for Implementing Human Reliability Analysis

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding the Draft Letter Report (JCN W6994), "Good Practices for Implementing Human Reliability Analysis (HRA)," dated April 6, 2004. The staff provided a broad overview of the HRA research program and discussed HRA good practices. The purpose of the guidance in HRA good practices document is to ensure some level of consistency and quality in HRA analyses and their review.

Committee Action

The Committee issued a letter to the NRC Executive Director for Operations on this matter dated May 13, 2004, recommending that the draft letter report be issued for public comment and also peer-reviewed by domestic and international experts. The Committee plans to review the draft final letter report after the public comment period and peer review.

6. Potential Adverse Effects from Power Uprates

The Committee heard presentations by and held discussions with representatives of the NRC staff regarding potential adverse effects from power uprates. The staff discussed the issue of steam dryers cracking at certain boiling water reactor (BWR) plants. In some cases, fractured metal parts from the steam dryer have entered the reactor coolant system and steam lines. The staff presented its actions and the industry activities for resolving this issue.

The members were critical of the staff and the industry response to this issue and questioned whether the staff and the industry really understood the causes of steam dryer cracking at several BWRs over the past two years and how extended power uprates affected this

equipment. The members were concerned about the apparent lack of risk analyses conducted at plants with steam dryer problems and about the staff's plans to continue granting uprates without first resolving the associated technical issues.

Committee Action

This was an information briefing and no Committee action was taken. However, the Committee will continue to be involved in the staff's plans and activities to resolve this issue.

7. Subcommittee Report on Fire Protection Issues

The Chairman of the ACRS Subcommittee on Fire Protection provided a report to the Committee regarding the matters discussed at the April 23, 2004 Subcommittee meeting. He stated that representatives of the NRC staff and the industry discussed three of the many ongoing NRC fire protection initiatives. The items discussed included resolution of post-fire circuit analysis issues, the revised Fire Significance Determination Process (SDP), and the RES-EPRI Fire Risk Requantification Study. The staff also provided status updates on rulemaking to allow operator manual actions to satisfy fire protection requirements and the voluntary adoption of the National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants."

Committee Action

The Committee plans to review the draft final rule on operator manual actions.

8. Resolution of Certain Items Identified by the ACRS In NUREG-1740 Related to the Differing Professional Opinion on Steam Generator Tube Integrity

The Committee completed its review of the NRC staff's resolution of certain items identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria," related to the differing professional opinion (DPO) on steam generator tube integrity. During the 509th ACRS meeting on February 5-7, 2004, the Committee heard presentations by and held discussions with representatives of the NRC staff and their contractors regarding the staff's resolution of several items identified by the ACRS in NUREG-1740 as well as the status of activities associated with the resolution of the remaining ACRS issues. The staff presented the resolution of certain items, which included steam generator tube integrity during main steamline break, correlation between voltage and leakrate for 7/8" steam generator tubes, and use of appropriate iodine spiking factor in the dose calculations for the design-basis accident.

Committee Action

The Committee issued a letter to the Executive Director for Operations on this matter dated May 21, 2004, which included several recommendations regarding the staff's resolution of certain items identified by the ACRS in NUREG-1740. The Committee plans to continue its discussion of this matter during future meetings.

9. Reconciliation of ACRS Comments and Recommendations

There were no EDO responses for discussion during this meeting.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from April 15, 2004 through May 5, 2004, the following Subcommittee meetings were held:

- Reactor Fuels - April 21, 2004

  The Subcommittee reviewed the proposed license amendment requesting authorization to use MOX fuel Lead Test Assemblies at Catawba.

- Human Factors/Reliability and Probabilistic Risk Assessment - April 22, 2004

  The Subcommittees discussed the proposed staff's guidance regarding Good Practices for Implementing Human Reliability Analysis and data development for Human Reliability Analysis.

- Fire Protection - April 23, 2004

  The Subcommittee discussed the resolution of post-fire safe shutdown circuit analysis revisions to the Reactor Oversight Process (ROP) fire SDP, and the preliminary results of the staff's Fire Risk Requantification Study.

- Planning and Procedures - May 5, 2004

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The ACRS Subcommittees on Plant Operations and on Thermal-Hydraulic Phenomena plans to hold meetings, as needed, to discuss the progress made by the staff in resolving the issues of potential adverse effects resulting from power uprates.

- The Committee plans to review Appendix C to RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," which will endorse the ANS Standard on external events. In addition, the Committee plans to review the proposed revision to RG 1.200 once the lessons learned from the trial applications have been incorporated.

- The Committee plans to meet with the staff and its contractors in August/September 2004 to discuss security issues related to reactors, fuel cycle facilities, spent fuel cask storage, and emergency response planning.

- The Committee plans to review the draft final report (JCN W6994), "Good Practices for Implementing Human Reliability Analysis (HRA)," after the public comment period and peer review.

- The Committee plans to continue its discussion of the staff's resolution of the remaining issues identified by the ACRS in NUREG-1740, "Voltage-Based Alternative Repair Criteria."

PROPOSED SCHEDULE FOR THE 513th ACRS MEETING

The Committee considered the following topics during the 513th ACRS meeting, held on June 2-4, 2004:

- Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors"
- Revised License Renewal Review Process
- Digital Instrumentation and Control System Research Activities
- NRC Staff Response to the ACRS Report on the AP1000 Design
- Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP
- Metrics for Evaluating the Quality of the NRC Research Programs

Sincerely,

/RA/

Mario V. Bonaca
Chairman

CERTIFIED

Date Issued: 5/28/2004
Date Certified: 6/29/2004

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I. *Federal Register Notice*
II. Meeting Schedule and Outline
III. Attendees
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MINUTES OF THE 512th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
May 5-8, 2004
ROCKVILLE, MARYLAND

The 512th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on May 5-9, 2004. Notice of this meeting was published in the *Federal Register* on April 28, 2004, (65 FR 23230) (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), 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. Dr. Graham B. Wallis did not attend this meeting. 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 11:00 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 Matters (Closed)

[Note: Dr. Richard P. Savio was the Designated Federal Official and Mr. Richard K. Major was the cognizant staff engineer for this portion of the meeting.]

This session was closed to protect information classified as national security information as well as unclassified safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).

III. Used of Mixed Oxide (MOX) Lead Test Assemblies at the Catawba Nuclear Station (Open)

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

Dr. Powers opened the session with a description of the issue. It relates to the national policy program to dispose of weapons-grade Pu in nuclear power plants. There is some significant experience with Pu/MOX in European reactors, but it uses reactor-grade Pu, not the weapons-grade material that will be used here. The subcommittee met with stakeholders on April 21, 2004. He noted that there is still an outstanding issue involving the NGF lead test assemblies (LTAs), and the Committee might have to delay issuance of its recommendations to the Commission.

Mr. Steve Nesbit, Duke Power, presented the program that has been proposed for Pu disposition. The first part of the program includes the insertion of 4 LTAs at Catawba. This program is the focus of the Pu-disposition program. He described the LTA program, which includes preparation of Pu at Los Alamos National Laboratory (LANL), fabrication of pellets and rods at Caderache, assembly fabrication at MELOX, irradiation at Catawba, and hot cell PIE at Oak Ridge National Laboratory (ORNL). Dr. Apostolakis asked about transport of the material. Mr. Nesbit explained that US transport will be by the Department of Energy (DOE) secure transport, by PNTL ship transport across the Atlantic, and by truck in France. Mr. Nesbit described the irradiation and examination program. The peak burnup is expected to be about 48GWD/MTHM at the end of the second cycle. The Cycle 3 burnup is expected to reach almost 60 GWD/MTHM. He described the planned core design, including the use of the NGF LTAs, and the required regulatory approvals.

The fuel will be fabricated using the MIMAS process, which has decades of experience in Belgium and France. The pellets have a uniform distribution of Pu at a macroscopic scale, with a heterogeneous microstructure at the micronic scale. He described the MIMAS process, and how it uses a master-blend process to achieve a homogeneous product. Mr. Sieber asked why the blending process used tails versus natural U. Mr. Nesbit explained that the use of tails is more representative of the current European process, and they wanted to be as close to that experience base as possible. Mr. Sieber asked about the grain size and use of previous experience at Hanford, and Mr. Nesbit provided micrographs of the MOX fuel microstructure, to show the degree of agglomeration of the Pu. The fuel is not quite as heterogenesous as might be thought, and he presented distribution charts of the amount of Pu in the different phases in the material that showed relatively low agglomeration.

Mr. Nesbit presented several comparisons of material properties of MOX versus LEU, to show how close the two fuel types are. One significant difference involves decay heat - MOX fuel decay heat is less than LEU up to about 3 days, and then it is greater than LEU. He also presented comparisons of some of the core nuclear physics parameters. Mr. Sieber noted that these physic differences will have to be taken into account for the batch loading, and Mr. Nesbit agreed, but noted that they are not significant for the 4 LTAs.

Mr. Nesbit described the fuel element design, which is based on the Advanced Mark BW design, and will use M5 cladding. The MOX design is almost identical to the LEU, but has a slightly longer fuel rod length, and a lower planned batch burnup. The current Catawba fuel supplier is Westinghouse (W), with RFA fuel. He also described the MOX fuel experience base in Europe. This experience includes hot cell examinations and power ramp testing and instrumentation to high burnups. The results have demonstrated the same behavior as LEU fuel in terms of various fuel rod phenomena. There has been somewhat higher fission gas release at higher burnup, but better pellet-cladding mechanical interaction. Overall, the physical characteristics of LEU and MOX are similar, and the experience base has proven this.

The LOCA analyses for MOX followed the current approved Appendix K model, with modifications to account for potential MOX effects, and a MOX to LEU comparison calculation was performed. Mr. Nesbit described some of the potential MOX effects, and how they were accounted for. The resulting difference in PCT was less than 40°F. Mr. Sieber commented that the Pu agglomerations produced hot spots in the pellet, which would produce cladding hot spots, and he asked how this was accounted for. Mr. Nesbit replied that they had not performed a local cladding analysis, but the temperatures are averaged over node sizes of about 6-12 inches, so micro-agglomeration effects are not visible. He also noted that they performed sensitivity studies and established peaking criteria to ensure that the MOX fuel remains within the LOCA acceptance criteria.

Non-LOCA events were considered, and the impact of 4 MOX LTAs was not appreciable. A few events with potential local effects were evaluated in more detail. Mr. Rosen asked how this experience would be factored into the batch loading analyses. Mr. Nesbit replied that Duke would come back to the staff for approval for the batch loading, and this would include consideration of their experience with the LTAs.

Mr. Nesbit also described the radiological consequences of using MOX versus LEU fuel, and he explained that the maximum impact would be seen in postulated accidents involving one or just a few assemblies, such as a fuel handling accident. In these cases, the offsite and control room doses are approximately 60% higher than LEU, but are still well within regulatory limits.

Mr. Nesbit described the environmental evaluation that was submitted, which determined that there is no impact on effluents, but a slight increase in fuel handling occupational doses. Environmental impacts due to accidents are addressed in the safety analyses and the radiological consequence analyses. By scaling the results of analyses performed by DOE and Mr. Lyman assuming 40% batch loading, they concluded that the maximum adverse impact of the 4 LTAs is about 1.6% greater than LEU fuel, including an assumption of a worst case

actinide release fraction. Overall, the severe accident behavior is driven by the LEU fuel, and the impact of the 4 MOX LTAs would be negligible. He compared this effect to the effect of a major power uprate, and noted that there is no consideration of severe accident environmental impact on power uprates.

Mr. Nesbit summarized his presentation with the observation that all nuclear power reactors are already using Pu fuel, as the power at end of cycle is about 50% due to Pu fissioning. He noted that the primary questions that have been raised relate to uncertainty of understanding fuel behavior, and he thought that the experience base was sufficient to show that the insertion of the LTAs is safe. Mr. Rosen asked what sort of dose increase would be involved. Mr. Nesbit explained that the MOX contact does is about 5 times higher than LEU.

Mr. Martin, Office of Nuclear Reactor Regulation (NRR), described the staff review process, including the submittals and the status of the NGF LTA issue. This work is ongoing, and the staff does not have a resolution path, yet. Mr. Leitch asked when the NGF LTAs would be inserted, and Mr. Martin stated that they are already in the core.

Ms. Shoop, NRR, described the staff review of the MOX LTA thermal-mechanical design. She began with a discussion of the LTA program, and the purpose of the LTA program, which is to generate data from a limited number of fuel assemblies, to support eventual batch application. She described the MOX fuel design report (BAW-10238), which provides the detailed design evaluation of the Advanced Mark BW fuel for MOX fuel. The assembly design differences include a longer fuel rod, use of European dish and chamfer dimensions for the pellets, 95% theoretical density, and the use of MOX instead of LEU. This staff approval of BAW-10238 is applicable to only the LTAs, and not to the batch loading of MOX. It will have to be re-reviewed when the batch application is made.

She explained that the Pu loading in the rods would be determined in order to provide an amount of reactivity that is similar to that provided by an equivalent LEU fuel bundle. The Pu has been polished to remove Ga down to a level of 300ppb, in order to prevent migration of Ga from the fuel to the cladding. The LTA data collection program will provide neutronics data about the fuel performance from startup physics testing, and about the fuel behavior from the PIE. Two of the LTAs will be located in core locations that are directly measured by in-core detectors. She described the poolside PIE examinations that will be performed to determine that the assembly geometry has not changed in an unexpected way, as well as the hot-cell examination that will be performed to evaluate fuel pellet behavior.

Ms. Shoop described the core nuclear design issues, including the key core physics parameters as a function of burnup. She noted that they do not change significantly as a result of the use of the MOX LTAs. Analyses of the non-LOCA transients confirmed that all physics parameters fall within the reference values previously calculated. For the control rod ejection event, the peak MOX enthalpy will be 30 cal/g, compared to a peak LEU enthalpy of 54 cal/g. for LEU. This is because none of the MOX LTAs will be loaded into rodded locations.

Dr. Landry, NRR, described the LOCA analyses that were provided, including the W analysis of record, and the MOX LTA LOCA analysis, which was performed using the Framatome Appendix K model. The staff closely considered the decay heat model that Framatome used, which bounded and was more conservative than the decay heat curve contained in Appendix K. This produced conservative results. He compared the results of the W BE model to the Framatome Appendix K results, and explained that the MOX PCTs are lower than the PCTs for the W RFA fuel, because the MOX peaking factors are lower. The staff is still considering the effect of the NGF LTAs, to make sure that they do not effect the MOX LTA behavior.

Mr. Steve LaVie, NRR, explained the staff's evaluation of the radiological consequences from the LTAs. He recalled that there is a greater FP release fraction from the MOX pellets, and this was considered. The staff performed a number of independent calculations of radionuclide release, and produced lower values than did Duke.

Mr. Lyman, Union of Concerned Scientists, presented a number of concerns that are being considered by the NRC's Atomic Safety and Licensing Board as part of the license amendment hearing. He noted that the compression of the licensing proceeding is due to a request from DOE for an NRC response this summer. He thought that many of the issues that are considered to be resolved for the LTAs will have to be addressed again for the batch application, and thought that this proceeding was setting a bad example for the Russians, who were supposed to be learning about nuclear licensing from this application. He described some of the security contentions that have been submitted, and expressed the opinion that Duke has failed to account for the different behavior of MOX, compared to LEU, for LOCA scenarios.

He noted particularly the uncertainty of knowledge due to gaps in the experimental database for MOX. He reported that the French regulatory authorities have proposed to perform new tests to fill in the gaps in the experimental database. One of the issues to be considered included fuel relocation during LOCA, which is not addressed by Appendix K, and which the staff does not seem to think is significant. He thought that this effect could increase fuel PCT during a LOCA by several hundred degrees.

In addition, the effect of larger ballooning from the use of the M5 alloy has not been considered, and neither has the increased fragmentation of MOX compared to LEU, at higher burnups. He recalled that the PIRT performed by the NRC staff in 2001 could not determine whether this was a significant issue, because of the lack of experimental data. He further noted that Dr. Thadani had recently sent a letter (April 21, 2004) to Mr. Modeen (Nuclear Energy Institute) that described "significant differences [in the performance of M5] compared with Zircaloy." Overall, he concluded that the behavior of MOX fuel during a "core disruptive accident" is not well enough understood, and has not been properly considered by Duke or by the NRC staff.

Mr. Killar, Director of Nuclear Fuel Supply at NEI, expressed support for the Pu-disposition program, and for this part of the program. He believes that Pu can be used safely in power reactors, and believes that the use of the LTA process is appropriate. He noted that both the Ginna and LaCrosse plants operated for some time with MOX fuel, and the experience was positive.

Committee Action

The Committee subsequently issued a letter to the Executive Director for Operations, concluding that, under the restricted circumstances considered in both the Duke Power application and the NRC staff's safety evaluation, the four MOX LTAs in non-limiting core locations that do not contain control rods can be irradiated in either of the Catawba reactor cores with no undue risk to the public health and safety.

IV. Risk Management Technical Specifications (Open)

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

Dr. George Apostolakis, Chairman of the Reliability and Probabilistic Risk Assessment Subcommittee introduced this topic to the Committee. The Committee heard presentations by the staff of the Office of Nuclear Reactor Regulation (NRR), NEI, and the South Texas Project regarding the status of the Risk Management Technical Specifications (RMTS), Initiative 4b, Risk Informed Completion Times.

NRC Staff Presentation

The NRC presentation on Risk Management Technical Specifications (RMTS) was made by Tom Boyce and Bob Tjader of NRR. The industry presentations were made by Biff Bradley, NEI, and Wayne Harrison and Bill Stillwell of South Texas Project (STP).

The purpose of this project is to risk-inform the technical specifications. RMTS Initiative 4b is in the early stages of development and will include an approved process, requirements for PRA technical adequacy, real-time quantitative capability and configuration and cumulative risk metrics. Initiative 4b is intended to extend the completion times from a current nominal value to a predetermined maximum using configuration risk management. The staff is currently reviewing a draft guidance document from NEI and pilot proposals from STP and Fort Calhoun. Hope Creek has volunteered to also be a pilot.

RMTS is dependent upon PRA quality. Communication and training of headquarters and regional staff are essential.

Issues associated with this project are the uncertainty of and impact on completion times, the extent of incorporation of risk monitors and assessment tools into the PRAs, QA/QC of the software and its updates, the risk associated with current completion times, and the time necessary to calculate the risk. Issues associated with the current pilot review are the ability to export the pilot general acceptance criteria and the PRA quality proof of concept.

Committee Action

The staff will brief the Committee in the future regarding additional work. This was an information briefing and no Committee action was taken.

V. Trial/Pilot Implementation of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Open)

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

The Committee met with representatives of the NRC staff and NEI to discuss the current activities and plans related to the five pilot applications of RG 1.200. NEI then provided their perspective on the pilot applications of RG 1.200. In a September 22, 2003 report to the Commission, the Committee agreed with the staff's recommendation that RG 1.200 be issued for trial use with an appropriate sample of pilot plants. The staff and NEI discussed what they hoped to learn form the pilot applications. Both the staff and NEI said that applying RG 1.200 has been more intensive than anticipated.

Committee Action:

This was an informational briefing. The Committee plans to write a report on Appendix C to RG 1.200 which will endorse the ANS Standard on external events. The Committee plans to write another report on Revision 0 to RG 1.200 once the lessons learned from the trial applications have been incorporated.

VI. Good Practices for Implementing Human Reliability Analysis (HRA) (Open)

[Note: Dr. Bhagwat Jain was the Designated Federal Official for this portion of the meeting.]

The Committee heard presentations by representatives of the NRC staff regarding Draft Letter Report (JCN W6994), "Good Practices for Implementing Human Reliability Analysis (HRA)," dated April 6, 2004. The staff provided a broad overview of the HRA research program and discussed HRA good practices. The purpose of the guidance in the HRA good practices document is to ensure some level of consistency and quality in HRA analyses and their review. The staff requested the Committee's concurrence for issuing the good practices document for public comment.

Committee Action

The Committee issued a report to the EDO on this matter dated May 13, 2004, in which it recommended that the draft letter report be issued for public comment and should also be peer-reviewed by domestic and international experts.

VII. Potential Adverse Effects from Power Uprates (Open)

[Note: Dr. Bhagwat Jain was the Designated Federal Official for this portion of the meeting.]

The Committee held discussions with representatives of the NRC staff regarding potential adverse effects from power uprates. The staff discussed the issue of steam dryer cracking at certain boiling water reactor (BWR) plants. In some cases, fractured metal parts from the steam dryer have entered the reactor coolant system and steam lines. The staff briefed the Committee regarding its actions and of the industry's activities for resolving the issue.

The Committee was very critical of the staff and the industry's response to this issue and questioned whether the staff and the industry really understood the causes of steam dryer cracking at several BWRs over the past two years and how extended power uprates and existing operations affected this equipment. The Members were critical about the apparent lack of risk analyses conducted at plants with steam dryer problems and expressed concern about the staff's plans to continue granting uprates without first resolving the associated technical issues.

Committee Action

No Committee action was required on the staff's information briefing. However, the Committee will continue to be involved in the staff's plans and activities to resolve this issue.

VIII. Subcommittee Report on Fire Protection (Open)

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

The Chairman of the ACRS Subcommittee on Fire Protection provided a report to the Committee regarding the matters discussed at an April 23, 2004, subcommittee meeting. He stated that representatives of the NRC staff and the industry discussed three of the many ongoing NRC fire protection initiatives. The items discussed included resolution of post-fire circuit analysis issues, the revised Fire Significance Determination Process (SDP), and the RES-EPRI Fire Risk Requantification Study. The staff also provided status updates on rulemaking to allow operator manual actions to satisfy fire protection requirements and the voluntary adoption of the National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants."

Committee Action

The Committee plans to review the draft final rule on operator manual actions.

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/EDO Commitments

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

There were no EDO responses for discussion during this meeting.

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 May 5, 2004. The following items were discussed:

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

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

- Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through July 2004 were considered. 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

- ACRS Meeting with the NRC Commissioners

The ACRS met with the NRC Commissioners on June 2, 2004. The following topics were addressed:

1. Overview (MVB)
2. PWR Sump Performance (JDS)
3. PRA Quality (for decisionmaking) (GEA)
4. Risk-Informing 10 CFR 50.46 (WJS)
5. NRC Safety Research Program Report (DAP)
6. ESBER Pre-Application Review (TSK)
7. Interim Review of the AP1000 Design (TSK)

- Revision to ACRS Action Plan

As agreed on by the Committee during its January 29-30, 2004, retreat, the ACRS Action Plan that was issued in 2001 is being revised. A proposed revision to the Action Plan includes a discussion of planned pro-active initiatives of the ACRS. A copy of the revised Action Plan will be sent to the members following the May ACRS meeting. Members are requested to provide their comments to Mrs. Weston by May 24, 2004.

- Visit to a Nuclear Plant and Regional Office

Each year the members visit a nuclear power plant and the NRC Regional Office and meet with the licensee and the Regional staff to discuss items of mutual interest. The Committee Members will visit the D.C. Cook Nuclear Plant and the Region III Office on June 9-10, 2004, as arranged by Mrs. Weston.

- Tour of Test Facilities Used for the ACR-700 Design

The ACRS Executive Director has suggested that some ACRS members and staff tour selected test facilities in Canada that were used for the ACR-700 design. The NRR staff suggested that the ACRS Subcommittees on Future Plant Designs and Material and Metallurgy tour the Chalk River facility and hold a meeting in Canada between July 25-30, 2004, to discuss various aspects of the ACR-700 design, including materials issues.

- LINK Technologies, Inc. Report

At the request of Mr. Rosen, LINK Technologies, Inc. has prepared a report that includes recommendations for enhancing the NRC training materials for inspecting a licensee's corrective action program and explores the possibility of implementing performance indicators in the reactor oversight process for addressing the corrective action programs. During the April 2004 meeting, the members had agreed to hear a presentation on this matter from a representative of the LINK Technologies Inc. at the May 2004 ACRS meeting.

- Effectiveness of Implementing Commitments Made During the ACRS Retreat

During the January 29-30, 2004, ACRS retreat the members made several commitments. It is worthwhile for the Planning and Procedures Subcommittee to periodically assess te effectiveness of the Committee's implementation of these commitments. The following commitment was chosen for the assessment:

- Commitment

  The members should allow uninterrupted presentations for about 10 minutes.

- ACRS Review of License Renewal Applications

During the review of the license renewal applications, especially those related to SEP plants, some members raise issues that are not within the scope of 10 CFR Part 54, the License Renewal Rule. In addition, it appears that they raise questions regarding the adequacy of the current licensing basis. It is important that the Committee's review be in conformance with the License Renewal Rule.

- NRC's International Council Meeting

Mr. Snodderly, ACRS Senior Staff Engineer, attended a meeting of the NRC's International Council on April 28, 2004. It was mentioned at the meeting that China appears to be serious about ordering an AP1000 reactor. The NRC Chairman has agreed to support a four day workshop in China during July 2004 to discuss design certification of AP1000. Mr. Thadani has the lead for this workshop. The workshop may have some impact on the staff review activities associated with the future plant designs.

- Staff Requirements Memorandum on RES Activities

An April 28, 2004 SRM, resulting from the RES briefing to the Commission on April 13, 2004 stated the following:

> "The staff should inform the Commission through the budget process about how specific recommendations in the Advisory Committee on Reactor Safeguards (ACRS) report, NUREG-1635, Volume 6, 'Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program,' dated March 2004, were dispositioned by the staff."

- Subcommittee Meetings/Annual Plant Visit

The subcommittee discussed the purpose, expected outcome, and appropriateness of the subcommittee meeting dates that are scheduled through June 2004.

- Interview of Candidates for Potential Membership on the ACRS (Closed)

The ACRS Member Candidate Screening Panel reviewed several applications and selected five candidates for interview during the June meeting. The Members should discuss and decide if they would like to add any additional names to the interview list.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the 513th ACRS Meeting, June 2-4, 2004.

The 512th ACRS meeting was adjourned at 7:00 p.m. on May 5, 2004.

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

May 28, 2004

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador
Technical Secretary

SUBJECT: PROPOSED MINUTES OF THE 512th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS - MAY 5-8, 2004

Enclosed are the proposed minutes of the 512th 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

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

June 29, 2004

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

FROM: Mario V. Bonaca
Chairman

SUBJECT: CERTIFIED MINUTES OF THE 512th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), MAY 5-8, 2004

I certify that based on my review of the minutes from the 512th 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.

Logistical Solutions has a tracking system that monitors the progress of the shipments from their originating point at SONGS until they arrive to their final destination at Envirocare in Clive, Utah. The shipments are made by either rail or combination truck/rail. According to the licensee, the transportation time alone by either rail or combination truck/rail took over 16 days on average, with one shipment taking 57 days to arrive at Envirocare.

In addition to this time, administrative procedures at Envirocare and mail delivery could add up to 11 additional days. Based on historical data and estimates of the remaining waste at SONGS Unit 1, the licensee could have to perform over 100 investigations and reports to the NRC during the next five years if the 20-day shipping criteria is maintained. The licensee affirms that the low-level radioactive waste shipments will always be tracked throughout transportation until they arrive at their intended destination. The licensee believes that the need to investigate, trace, and report to the NRC on the shipment of low-level radioactive waste packages not reaching their destination within 20 days does not serve the underlying purpose of the rule and it is not necessary. As a result, the licensee states that granting this exemption will not result in an undue hazard to life or property.

The NRC has examined the licensee's proposed exemption request and concluded that it is procedural and administrative in nature. There are no significant radiological environmental impacts associated with this exemption, and it will not result in significant nonradiological environmental impacts.

III. Finding of No Significant Impact

NRC has prepared the EA (summarized above) in support of the licensee's application for an exemption request. On the basis of the environmental assessment, the NRC concludes that the proposed action will not have a significant effect on the quality of the human environment. Accordingly, the NRC has determined not to prepare an environmental impact statement for the proposed action.

IV. Further Information

The EA and the documents related to this proposed action, including the request for the exemption, are available for inspection at the NRC Public Electronic Reading Room at the following address: *http://www.nrc.gov/reading-rm/pdr.html.* The ADAMS accession number for the licensee's exemption request letter dated January 26, 2004 is ML040330945. The ADAMS accession number for the EA is ML040780782. Persons who do not have access to ADAMS, or who encounter problems in accessing the documents located in ADAMS, should contact the NRC Public Document Room (PDR) reference staff by telephone at 1- 800- 397- 4209 or 301- 415- 4737. They can also be reached via e-mail at *pdr@nrc.gov.* Documents may also be examined, and/or copied for a fee, at the NRC PDR, located at One White Flint North, 11555 Rockville Pike, Rockville, MD 20852. Any questions with respect to this action should be referred to Mr. William C. Huffman, Division of Waste Management and Environmental Protection, Office of Nuclear Material Safety and Safeguards. He can be reached at (301) 415- 1141.

For the Nuclear Regulatory Commission.

Dated in Rockville, Maryland, this 21st day of April, 2004.

Daniel M. Gillen,

Deputy Director, Decommissioning Directorate, Division of Waste Management and Environmental Protection, Office of Nuclear Material Safety and Safeguards.

[FR Doc. E4- 955 Filed 4- 27- 04; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

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

The ACRS Subcommittee on Planning and Procedures will hold a meeting on May 5, 2004, Room T- 2B1, 11545 Rockville Pike, Rockville, Maryland.

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

The agenda for the subject meeting shall be as follows:

Wednesday, May 5, 2004- 8:30 a.m.- 10:30 a.m.

The Subcommittee will discuss proposed ACRS activities and related matters. The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Members of the public desiring to provide oral statements and/or written comments should notify the Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301- 415- 7364) between 7:30 a.m. and 4:15 p.m. (e.t.) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 7:30 a.m. and 4:15 p.m. (e.t.). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the agenda.

Dated: April 20, 2004.

Medhat El-Zeftawy,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. E4- 952 Filed 4- 27- 04; 8:45 am]

BILLING CODE 7590–01–P

OFFICE OF MANAGEMENT AND BUDGET

Revised Information Quality Bulletin on Peer Review

AGENCY: Office of Management and Budget, Executive Office of the President.

ACTION: Notice and request for comments.

SUMMARY: The Office of Management and Budget (OMB), in consultation with the Office of Science and Technology Policy (OSTP), is re-proposing its new guidance designed to realize the benefits of meaningful peer review of the most important science disseminated by the Federal Government. This Notice requests comment on the revised Bulletin, now entitled "Revised Information Quality Bulletin on Peer Review." OMB originally requested comment on its "Proposed Bulletin on Peer Review and Information Quality," published in the **Federal Register** on September 15, 2003. We received 187 comments during the public comment period, listened to discussion at a public workshop at the National Academy of Sciences (NAS), and carried out an interagency review. This process led to a substantially revised Bulletin, which incorporates many of the diverse perspectives and suggestions voiced during the comment period. The public comments are posted at: *http://www.whitehouse.gov/omb/inforeg/2003iq/iq_list.html.* A summary of the public and agency comments, including responses by OMB and OSTP, is

APPENDIX II

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

Member G. Wallis did not attend

April 22, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
512th ACRS MEETING
MAY 5-8, 2004

WEDNESDAY, MAY 5, 2004

1) 11:00 - 11:05 A.M. Opening Remarks by the ACRS Chairman **(Closed)** (MVB/JTL)

2) 11:05 - 6:30 P.M. Safeguards and Security Matters **(Closed)** (MVB/RPS/RKM)
(12:30-1:30 P.M. LUNCH)
2.1) Remarks by the ACRS 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 regarding safeguards and security matters.

[NOTE: This session will be closed to protect information classified as national security information as well as unclassified safeguards information pursuant to 5 U.S.C. 552b(c)(1) and (3).]

THURSDAY, MAY 6, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

4) 8:35 - ~~10:30~~ 10:50 A.M. Use of Mixed Oxide (MOX) Lead Test Assemblies at the Catawba Nuclear Station (Open) (DAP/RC)
4.1) Remarks by the Subcommittee Chairman
4.2) Briefing by and discussions with representatives of the NRC staff and Duke Cogema Stone and Webster (DCS) regarding the license amendment submitted by DCS to obtain NRC authorization to use MOX lead test assemblies at the Catawba Nuclear Station.

Members of the public may provide their views, as appropriate.

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

5) 11:05-12:30 ~~10:45~~ - ~~12:15~~ P.M. Risk Management Technical Specifications (Open) (GEA/MWW)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff regarding the status/overview of the initiatives associated with the risk management technical specifications, and the staff's evaluation of the proposals for pilot application of the initiative on Risk-Informed Completion Times.

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

12:40-1:25 Carl Paperiello, Director – Office of Nuclear Regulatory Research

~~12:15~~ - ~~1:15~~ P.M. 1:25-2:00 *****LUNCH*****

6) ~~1:15~~ - ~~3:15~~ P.M. 2:00-2:53 Trial/Pilot Implementation of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Open) (GEA/MRS)

6.1) Remarks by the Subcommittee Chairman

6.2) Briefing by and discussions with representatives of the NRC staff regarding insights gained from the trial/pilot implementation of Regulatory Guide 1.200.

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

3:00- ~~3:15~~ - 3:30 P.M. *****BREAK*****

7) 3:30 - 5:00 ~~4:45~~ P.M. Good Practices for Implementing Human Reliability Analysis (Open) (GEA/BPJ)

7.1) Remarks by the Subcommittee Chairman

7.2) Briefing by and discussions with representatives of the NRC staff and their contractors regarding the draft report on Good Practices for Implementing Human Reliability Analysis, as well as the ongoing efforts associated with the application of the methodology, "A Technique for Human Event Analysis (ATHEANA)."

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

5:00-5:15 ~~4:45~~ - ~~5:00~~ P.M. *****BREAK*****

8) 5:15- ~~5:00~~ - 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

5:15-5:40 8.1) Use of MOX Lead Test Assemblies at the Catawba Nuclear Station (DAP/RC) FINAL

~~8.2) Risk Management Technical Specification (GEA/MWW)~~

5:40-5:48 8.3) Good Practices for Implementing Human Reliability Analysis (GEA/BPJ)

6:20-6:25 8.4) Divergence in Regulatory Requirements Between U.S. and Several Other Countries (DAP/HPN/SD)

5:48-6:20 8.5) Resolution of Certain Items Identified by the ACRS in NUREG-1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity (FPF/BPJ)

FRIDAY, MAY 7, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

9) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)

10) 8:35 - 10:35 ~~10:30~~ A.M. Potential Adverse Effects from Power Uprates (Open) (JDS/BPJ)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding adverse effects experienced as a result of core power uprates and status of ongoing and proposed activities of the industry and the NRC staff to address this issue.

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

10:35- 10:50 **~~10:30~~ - ~~10:45~~ A.M.** ***BREAK***

10:50-11:12

11) ~~10:45~~ - ~~11:00~~ A.M. Subcommittee Report on Fire Protection Issues (Open) (SLR/MDS)

Report by and discussions with the Chairman of the Fire Protection Subcommittee regarding matters discussed during the April 23, 2004 Subcommittee meeting.

11:12- 1:00 PM

12) ~~11:00~~ - ~~12:00 Noon~~ Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

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

13) 12:00 - 12:15 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.

~~12:15~~ - ~~1:15~~ P.M. *****LUNCH*****
1:00-2:00

14) ~~1:15~~ - ~~2:15~~ P.M.
2:20-3:45
Preparation for meeting with the Commissioners (Open) (MVB, et.al/JTL, et.al)
Discussion of topics scheduled for meeting with the NRC Commissioners in June 2004:

a) Overview (MVB)
b) PWR Sump Performance (JDS)
c) PRA Quality (for Decisionmaking) (GEA)
d) Risk-Informing 10 CFR 50.46 (WJS)
e) NRC Safety Research Program Report (DAP)
f) ESBWR Pre-Application Review (TSK)
g) Interim Review of the AP1000 Design (TSK)

3:45-4:05
~~2:15~~ - ~~2:30~~ P.M. *****BREAK*****

15) 2:30 - 6:30 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

15.1) Use of MOX Lead Test Assemblies at the Catawba Nuclear Station (DAP/RC) FINAL
~~15.2) Risk Management Technical Specification (GEA/MWW)~~
4:05-4:55 15.3) Good Practices for Implementing Human Reliability Analysis (GEA/BPJ) FINAL
15.4) Divergence in Regulatory Requirements Between U.S. and Several Other Countries (DAP/HPN/SD)
5:10-6:55 15.5) Resolution of Certain Items Identified by the ACRS in NUREG-1740 Related to Differing Professional Opinion on Steam Generator Tube Integrity (FPF/BPJ)

SATURDAY, MAY 8, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

16)	8:30 - 12:00 Noon	Preparation of ACRS Reports (Open) Continue discussion of proposed ACRS reports listed under Item 15.
17)	12:00 - 12:30 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) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX III: MEETING ATTENDEES

512TH ACRS MEETING
May 5-8, 2004

NRC STAFF (May 5)

A. Ramey Smith, RES
M. Cunningham, RES
E. Thorsbury, RES
A. Kuritzky, RES
B. Tegeler, RES
R. Sullivan, NRR
S. Ali, RES
C. Tinkler, RES
D. Helton, RES
J. Schaperow, RES

NRC STAFF (May 6)

T.R. Tjader, NRR
P. J. Habighorst, NRR
C. Carpenter, NRR
A. Levin, RES
J. Craig, RES
B. Kemper, OIG
A. Kugler, RES
R. Landry, NRR
S. Levie, NRR
U. Shoop, NRR
T. Attard, NRR
J. Wermiel, NRR
S. Coffin, NRR
B. Martin, NRR
S. Klementowicz, NRR
S. Sakai, NRR
R. Meyer, RES
W. Smith, NMSS
D. Harrison, NRR
A. El-Bassioni, NRR
J. Hong, NRR
N.T. NRR
T. Boyce, NRR
S. Magruder, NRR
C. Paperiello, RES
G. Parry, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC
(May 6, 2004)

S. Nesbit, Duke Power
M. Scott, Duke Power
D. Horner, McGraw-Hill
B. Bradley, NEI
S. Kauffman, NR-DOE
M. Cash, Duke Energy
D. Alberstein, DOE
G. Meyer, Framatome ANP
A. Cottingham, Winston & Strawn
K. McCoy, Framatome ANP
F. Killan, NEI
W. Hamson, STPNOC
B. Stillwell, STPNOC

NRC STAFF (May 7)

L. Rossbach, NRR
D. Terao, NRR
T. Scarbrough, NRR
J. Hernandez, NRR
S. Malik, RES
P. Gunter, NIRS
D. Hiser, RES
T. McMurfray, NRR
R. Aluck, NRR
D. Weaver, OEDO
B. Elliot, NRR
W. Krotiuk, RES
J. Hong, NRR
J. Fleck, RES
E. McKenna, NRR
D. Diec, NRR

ATTENDEES FROM OTHER AGENCIES AND GENERAL PUBLIC

G. Ohlemachere, DTE Energy
C. Roberts, GE
K. Hutke, PSEG
D. Lochbaum, UCS
P. Negris, GE
B. Hoffman, Public Citizen
J. Meyer, ISL
D. Distel, Exelon
J. Weil, McGraw-Hill
C. Nichols, Entergy
Brian Hobbs, Entergy

APPENDIX IV: FUTURE AGENDA

INSERT A COPY OF THE NEXT MEETING, TYPE APPENDIX IV IN THE RIGHT HAND CORNER

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

May 12, 2004

SCHEDULE AND OUTLINE FOR DISCUSSION
513th ACRS MEETING
JUNE 2-4, 2004

WEDNESDAY, JUNE 2, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

1) 8:30 - 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/JTL/SD)
   - 1.2) Opening Statement
   - 1.2) Items of current interest

2) 8:35 - 10:30 A.M. Draft Final 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors" (Open) (GEA/MRS)
   - 2.1) Remarks by the Subcommittee Chairman
   - 2.2) Briefing by and discussions with representatives of the NRC staff and Nuclear Energy Institute (NEI) regarding the draft final 10 CFR 50.69, and draft final Regulatory Guide DG-1121, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power plants according to their Safety Significances," which endorses NEI 00-04, "10 CFR 50.69 SSC Categorization Guideline."

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

3) 10:45 - 11:45 A.M. Revised License Renewal Review Process (Open) (MVB/MDS)
   - 3.1) Remarks by the Subcommittee Chairman
   - 3.2) Briefing by and discussions with representatives of the NRC staff regarding the revised process for the staff's review of the license renewal applications.

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

11:45 - 12:45 P.M. ***LUNCH***

4) 12:45 - 1:15 P.M. Preparation for Meeting with the NRC Commissioners (Open) (MVB, et.al/JTL, et.al)
   Discussion of the following topics scheduled for the ACRS meeting with the NRC Commissioners:
   - a) Overview (MVB)
   - b) PWR Sump Performance (JDS)
   - c) PRA Quality for Decisionmaking (GEA)
   - d) Risk-Informing 10 CFR 50.46 (WJS)
   - e) NRC Safety Research Program Report (DAP)
   - f) ESBWR Pre-Application Review (TSK)
   - g) Interim Review of the AP1000 Design (TSK)

1:15 - 1:30 P.M. *****BREAK*****

5) 1:30 - 3:30 P.M. Meeting with the NRC Commissioners, Commissioners' Conference Room, One White Flint North, Rockville, MD (Open) (MVB, et.al/ JTL, et.al)
Meeting with the NRC Commissioners to discuss the topics listed under Item 4.

3:30 - 4:00 P.M. *****BREAK*****

6) 4:00 - 5:30 P.M. Digital Instrumentation and Control System Research Activities (Open) (JDS/GEA/MDS)
6.1) Remarks by the Subcommittee Chairman
6.2) Briefing by and discussions with representatives of the NRC staff and their contractors regarding NRC research activities in the area of digital instrumentation and control (I&C) systems and related matters.

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

5:30 - 5:45 P.M. *****BREAK*****

7) 5:45 - 6:45 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:
7.1) Draft Final 10 CFR 50.69 and Regulatory Guide DG-1121 (GEA/MRS)
7.2) Digital I&C Research Activities (JDS/GEA/MDS)

THURSDAY, JUNE 3, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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

9) 8:35 - 10:30 A.M. NRC Staff's Response to the ACRS Report on the AP1000 Design (Open) (TSK/MME)
9.1) Remarks by the Subcommittee Chairman
9.2) Briefing by and discussions with representatives of the NRC staff regarding their response to ACRS comments and recommendations included in the March 17, 2004 ACRS report on the AP1000 design.

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

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

10) 10:45 - 12:00 Noon Proposed Revisions to Standard Review Plan (SRP) Sections and Process and Schedule for Revising the SRP (Open) (FPF/SD)

10.1) Remarks by the Subcommittee Chairman

10.2) Briefing by and discussions with representatives of the NRC staff regarding proposed revisions to SRP Sections: 5.2.3; "Reactor Coolant Pressure Boundary Materials;" 5.3.1, "Reactor Vessel Materials;" and 5.3.3, "Reactor Vessel Integrity;" as well as the process and schedule for revising various SRP Sections, including milestones for ACRS review of the proposed revisions.

12:00 - 1:30 P.M. ***LUNCH***

11) 1:30 - 2:30 P.M Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open) (MVB/JTL/SD)

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

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

12) 2:30 - 2:45 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.

2:45 - 3:00 P.M. ***BREAK***

13) 3:00 - 6:30 P.M. Preparation of ACRS Reports (Open)

Discussion of the proposed ACRS reports on:

13.1) Draft Final 10 CFR 50.69 and Regulatory Guide DG-1121 (GEA/MRS)

13.2) Digital I&C Research Activities (JDS/GEA/MDS)

13.3) Proposed Revisions to SRP Sections (Tentative) (FPF/SD)

FRIDAY, JUNE 4, 2004, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

14) 8:30 - 11:00 A.M. Metrics for Evaluating the Quality of the NRC Research Programs (Open) (GEA/HPN)

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

Discussion of the quantitative metrics for use by the ACRS in evaluating the quality of the NRC research programs.

15) 11:00 - 12:00 Noon Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports listed under Item 13.

12:00 - 1:30 P.M. * LUNCH***

16)	1:30 - 4:00 P.M.	Preparation of ACRS Reports (Open) Continue discussion of the proposed ACRS reports listed under Item 13.
17)	4:00 - 4:30 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) hard copies and (1) electronic copy of the presentation materials should be provided to the ACRS.**

APPENDIX V
LIST OF DOCUMENTS PROVIDED TO THE COMMITTEE
512TH ACRS MEETING
MAY 5-8, 2004

[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 May 5-8, 2004
2	Use of Mixed Oxide (MOX) Lead Test Assemblies at the Catawba Nuclear Station
	2. MOX Fuel Lead Assembly Program presentation by S. Nesbit, MOX Fuel Project Manager, Duke Power [Viewgraphs]
	3. NRC Staff Review of Mixed Oxide Lead Test Assemblies at Catawba Nuclear Station presentation by NRR [Viewgraphs]
	4. SRXB Review of the Mixed Oxide Fuel Lead Test Assemblies presentation by U. Shoop, NRR [Viewgraphs]
	5. Catawba MOX LTA LOCA presentation by R. Landry, NRR [Viewgraphs]
	6. Use of Mixed-Oxide Lead Test Assemblies at Catawba presentation by E. Lyman, Union of Concerned Scientists [Viewgraphs]
5	Risk Management Technical Specifications
	7. Risk Management Technical Specifications presentation by NRR [Viewgraphs]
	8. Risk Management Technical Specifications Initiative 4B presentation by NEI [Viewgraphs]
	9. STP Risk-Informed Technical Specifications Application presentation by STP Nuclear Operating Company [Viewgraphs]
6	Trial/Pilot Implementation of Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"
	10. RG 1.200 (and SRP 19.1) "An Approach for Determining the Technical Adequacy of PRA Results for Risk-Informed Activities presentation by RES and NRR [Viewgraphs]
7	Good Practices for Implementing Human Reliability Analysis
	11. Good Practices for Implementing Human Reliability Analysis presentation by RES, Sandia National Laboratories, and SAIC

10. Potential Adverse Effects from Power Uprates

    12. Potential Adverse Flow Effects From Power Uprates
    13. Draft Research Plan to Assess Potential Adverse Flow Effects During BWR Power Uprates

11. Subcommittee Report on Fire Protection Issues

    14. Fire Protection Subcommittee Report

12. Future ACRS Activities/Report of the Planning and Procedures Subcommittee

    15. Future ACRS Activities/Final Draft Minutes of Planning and Procedures Subcommittee Meeting - **May 5**, 2004 [Handout #**12.1**]

Appendix V
512th ACRS Meeting

MEETING NOTEBOOK CONTENTS

TAB	DOCUMENTS
4	Mixed Oxide (MOX) Fuel Lead Test Assemblies (LTAs)
	1. Table of Contents
	2. Proposed Schedule
	3. Status Report
	4. List of Correspondence between NRC staff, Duke Energy, Framatome and DCS related to MOX LTAs at Catawba
	5. List of Admitted Contentions
	6. Letter from R. Martin G. Barron, April 5, 2004, "Safety Evaluation for Proposed Amendments to the Facility Operating License and Technical Specifications to Allow Insertion of Mixed Oxide Fuel Lead Assemblies (TAC Nos. MB7864, MB7864, MC0824, and MC0825)
	7. Draft Meeting Summary, Reactor Fuels Subcommittee, April 4, 2004
	8. Letter from H. Barron to USNRC, April 16, 2004, "Proposed Amendments to the Facility Operating License and Technical Specification to Allow Insertion of Mixed Oxide (MOX) Fuel Lead Assemblies (MOX in Catawba 1 Cycle 16)"
	9. Presentation Slides for the NRC staff, April 23, 2004, "Next Generation Fuel and MOX"
5	Risk Management Technical Specifications (RMTS)
	10. Table of Contents
	11. Proposed Schedule
	12. Status Report
6	Trial/Pilot Implementation of Regulatory Guide 1.200 "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"
	13. Table of Contents
	14. Proposed Schedule
	15. Status Report
	16. Report dated September 22, 2003, from Mario V. Bonaca, Chairman, ACRS, to Nils J. Diaz, Chairman, NRC, Subject: Draft Final Regulatory Guide x.xxx, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Formerly DG-1122)
	17. US Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" February 2004
7	Good Practices for Implementing Human Reliability Analysis
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11 Subcommittee on Plant Fire Protection
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- 25. Overview
- 26. PWR Sump Performance
- 27. PRA Quality (for Decisionmaking)
- 28. Risk-Informing 10 CFR 50.46
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- 30. ESBWR Pre-Application Review
- 31. Interim Review of the AP1000 Design

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
512TH FULL COMMITTEE MEETING

MAY 5-8, 2004

May 5, 2004
Today's Date

NRC STAFF PLEASE SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Ann Ramey Smith	RES
Mark Cunningham	RES
Eric Thorsbury	RES
Alan Kuritzky	RES
BRET TEGELER	RES
R Sullivan	EPPO (last presentation only)
Mario Bonaca	ACRS MEMBER
G. Lutch	
W. Shack	
J. Sieber	
S. Rosen	
D. Powers	
Ym Ransom	
Tom Kress	
R. Warner	ACNW MEMBER
SYED ALI	RES
C TINKLER	RES
DON HELTON	RES
J. Schaperow	RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
512TH FULL COMMITTEE MEETING

MAY 5-8, 2004

May 5, 2004
Today's Date

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R. MAJOR	ACRS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
512TH FULL COMMITTEE MEETING

MAY 5-8, 2004

May 5, 2004
Today's Date

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

512th FULL COMMITTEE MEETING

MAY 6-8, 2004

May 6, 2004
Today's Date

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NAME	AFFILIATION
Steve Nesbit	Duke Power
Michael Scott	Duke Power
Daniel Horner	McGraw-Hill
~~[illegible]~~	
Biff Bradley	NEI
S. KAUFFMAN	NR-DOE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

512th FULL COMMITTEE MEETING

MAY 6-8, 2004

May 6, 2004
Today's Date

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NAME	AFFILIATION
RALPH LANDRY	NRR/DSSA/SRXB
Anne Cottingham	WINSTON & STRAWN
Mike Cash	Duke Energy
DAVID ALBERSTEIN	DOE/NNSA NA-26
George Meyer	DCS/Framatome ANP
KEVIN McCOY	FRAMATOME ANP
Steve LaVie	NRR/DSSA/SPSB-C
UNDINE SHOOP	NRR/DSSA/SRXB
Tony Attard	NRR/DSSA/SRXB
Jared Wermiel	NRR/DSSA/SRXB
Stephanie Coffin	NRR/DLPM/PDII-1
Bob Martin	NRR/DLPM/PD II-1
Felix Killar	NEI
Stephen Klementowicz	NRR/DIPM
Stacie Sakai	NRR/DIPM
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WILKINS SMITH	NRC, NMSS
Donnie Harrison	NRC/NRR/DSSA/SPSB
Mark Reinhart	NRC/NRR/DSSA/SPSB
A. El-Bassioni	NRC/NRR/DSSA/SPSB
JIANG HONG	NRR/DE/EEIB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

512th FULL COMMITTEE MEETING

MAY 6-8, 2004

May 6, 2004
Today's Date

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NAME	AFFILIATION
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N. T. Saltos	NRR / DSSA / SPSB
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John Craig	RES / DOD
Gareth Parry	NRR / DSSA

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

512th FULL COMMITTEE MEETING

MAY 6-8, 2004

May ~~7~~ 6, 2004
Today's Date

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NAME	AFFILIATION
BILL KEMPER	NRC/OIG
ANDY KUGLER	NRC/RES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

512th FULL COMMITTEE MEETING

MAY 6-8, 2004

May 7, 2004
Today's Date

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Larry Rossbach	NRR/DLPM
David Terao	NRR/DE/EMEB
Thomas Scarbrough	NRR/DE/EMEB
Jorge Hernandez	NRR/DE/EMEB
SHAH MALIK	RES/DET/MEB
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DON HELTON	RES/DSARE/SMSAB
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Raj Anand	NRR/DRIP
Doug Weaver	OEDO
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William Krotiuk	RES/DSARIE/REAHFB
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JOHN FLACK	RES/DSARE
Eileen McKenna	NRR/DRIP

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS AND SECURITY

512th FULL COMMITTEE MEETING

MAY 6-8, 2004

May 7, 2004
Today's Date

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NAME	AFFILIATION
Glen Ohlemacher	DTE Energy / BWROG
Christina Roberts	General Electric
KENNETH HUTKC	PSEG
Dave Lochbaum	Union of Concerned Scientists
Paige Negus	GE
Brendan Hoffman	Public Citizen
Jim Meyer	ISL
DAVID J. DISTEL	EXELON
~~William~~	
David Dmc	NRC
Jenny Weil	McGraw-Hill
CRAIG NICHOLS	ENTERGY
Brian Hobbs	Entergy

1

ITEMS OF INTEREST

512th ACRS MEETING

MAY 5-8, 2004

ITEMS OF INTEREST
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
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STAFF REQUIREMENTS MEMORANDUM

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NRC ANNOUNCEMENTS

COMMISSION ORDERS

SIGNIFICANT ENFORCEMENT ACTIONS

INSIDE NRC ARTICLES

IN RESPONSE, PLEASE
REFER TO: M040413

April 28, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette Vietti-Cook, Secretary **/RA/**

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RESEARCH PROGRAMS, PERFORMANCE, AND PLANS, 9:30 A.M., TUESDAY, APRIL 13, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff from the Office of Nuclear Regulatory Research (RES) on the programs, performance, and plans for the office.

The staff should communicate research results, particularly those involving conservative bounding analyses, to the public using plain English and in a manner to facilitate better understanding of the context and limitations of the information presented. When research reports are misused and quoted out of context, the staff should respond promptly.

The staff should inform the Commission through the budget process about how specific recommendations in the Advisory Committee on Reactor Safeguards (ACRS) report, NUREG-1635, Volume 6, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," dated March 2004, were dispositioned by the staff.

The Commission requested that the staff keep them currently informed on progress in the research on reactor material degradation issues.

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

April 12, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

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

SUBJECT: STAFF REQUIREMENTS - SECY-04-0030 - DEVELOPMENT OF A MORE ROBUST MATERIALS RESEARCH PROGRAM

The Commission has approved option 2 of SECY-04-0030, to initiate a more pro-active radiation protection research program, subject to the following. The staff in both the Office of Nuclear Regulatory Research (RES) and in the Office of Nuclear Material Safety and Safeguards (NMSS) should continue to look for ways to build a more robust materials program in RES by evaluating if there are NMSS activities that more appropriately belong in RES.

The staff should maintain strong oversight of this more pro-active program to ensure it focuses on achieving the strategic goals and objectives of the agency and the programmatic needs of the various offices it is designed to support. Specific research projects should be clearly aligned with NRC goals and the strategies for meeting those goals, and resources for specific projects should be addressed through the normal planning, budgeting and performance management (PBPM) process. Regarding the staff's proposal for a more robust forward thinking research program, the Commission can certainly understand the need to be conscious of new and better ways to efficiently and effectively conduct NRC business. At the same time, the NRC should devote the majority of its limited resources to addressing critical needs. The Commission expects very strong management in the PBPM process over this aspect of the proposed research program.

In addition, to reduce costs, this program should be initiated with greater reliance on in-house staff rather than contractors. For the international effort, staff and management should focus on the strategic goals of the Commission and limit international travel to the defined needs of the Commission.

Key areas that NMSS should consider for research user needs include the development of better, i.e., more realistic, models to address health effects (either through a realistic model or by establishing an approach that determines a reasonable range of likely consequences), atmospheric dispersion, and source terms.

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

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION
Office of Public Affairs Telephone: 301/415-8200
Washington, DC 20555-001 E-mail: opa@nrc.gov
Web Site: http://www.nrc.gov/OPA

No. S-04-005

Special Session of the 37th JAIF Annual Conference

Status and Future Outlook for Regulation of Nuclear Power Plants in the US

(a Regulatory Program for the 21th Century)

Nils J. Diaz, Chairman, U.S. NRC
April 21, 2004
Tokyo, Japan

Introduction

Good morning. It is indeed an honor to address the Japan Atomic Industrial Forum and a real personal pleasure to share with this distinguished group my views on a nuclear power plant regulatory program for the first quarter of this century; for today and possibly the next 25 years. The ideas and activities that I will be discussing have been developed in the context of the U.S. nuclear reactor program. The regulators and the nuclear industry in Japan and elsewhere must decide if these are useful for their country. Today, I will expand on some thoughts I presented at the 2004 U.S. - Japan, Nuclear Energy Workshop on the subject of "The Role of Nuclear Regulation in a Changing World."

The regulation of nuclear power plants in the U.S. has an established and functional foundation, yet it is in a transitional phase. Building on the traditions, approaches, and decisions of the past, we are developing, testing, and using state-of-the-art safety methods and technologies, including a risk-informed and performance-based regulatory approach to safety that is realistic and conservative, to implement a regulatory program for now and the near future. The existing regulatory fabric, woven piece-by-piece, and stitched together during the 1960's, 70's, 80's and 90's, has served us well; but that patch work is not efficient for existing plants and definitely not sufficient or effective enough for a new generation of nuclear power plants. We need, and we are constructing, a regulatory program that better meets our present needs, one that will be maintained in-phase with the technological developments of the 21th century. It is worthwhile to note that these regulatory improvements are, in many ways, enabled by a nuclear power industry that has been improving safety and reliability performance for many years.

The NRC, amid a changing world scenario, is continuing strong oversight of the 104 operating reactors in the U.S., and our review of applications for license renewal, power up-rate, and other licensing changes is effective and efficient. Furthermore, standard design certification work is ongoing and we have begun our oversight of new areas, including Early Site Permits and Combined Operating Licenses. New reactor design and pre-application work is also being conducted. The new regulatory fabric is being woven, in a systematic, disciplined and open manner. This new regulatory fabric requires the seamless weaving or inter-weaving of numerous safety issues, as well as their integration with associated technical considerations. Some of them are new and some are old, and most have been seen and addressed as separated and isolated issues in the past.

The U.S. NRC regulatory framework is more risk-informed and becoming more performance-based. It increasingly relies on Probabilistic Safety Assessments to make sound regulatory decisions. Probabilistic Safety Assessment has been recently woven together with traditional, defense-in-depth engineering approaches and with performance monitoring techniques to establish risk-informed and performance-based regulation. Reactor safety, physical security and emergency preparedness are being woven together into a single broader concept of safety. Realism and conservatism are being woven into realistic conservatism. The oversight of operation, maintenance, design and other aspects of nuclear power plant safety are being woven into a safety management program (some call it safety culture, but I still prefer safety management). And now we can see the need to connect them, and the possibility of unifying them. I believe it is both possible and necessary to combine these regulatory modules into a single architecture where the interactive determinants and outcomes of safety/security/emergency preparedness areas are understood, and managed through a risk-informed and performance-based approach supported by realistically conservative analyses. The driver and overall outcome is the reasonable assurance of adequate protection of public health and safety, the environment and the common defense and security. Allow me to take a few minutes to address the importance of improving the regulatory process in general and some of the areas of improvement in more detail.

The Proper Role of Regulation

I believe the outlook for nuclear energy is very good, if we consider the improved state of the technology, the assured supplies of fuel, the expectations of the world for an improved quality of life and for socio-political stability, and when appropriate and effective regulatory programs are available to provide reasonable assurance of safety and protection of the environment. We still need to communicate all of the above better, but that is another topic in itself.

The viability, and the probable growth, of nuclear power is inextricably linked to its regulation. I want to be crystal clear in addressing this issue. There is no way, presently and in the foreseeable future, to maintain and to advance the use of nuclear power in a free society without a strong, predictable and credible regulator. Therefore, it is essential that regulatory infrastructures be all that they can be: safety-focused, with state-of-the-art know-how in every important safety aspect. As regulators we make independent decisions, listening to and respecting different views, but without undue interference. We at the NRC should be willing to risk criticism by communicating both the good and the not-so-good safety performance, as well as assessing and explaining potential risks with realistically conservative analysis, always based on providing assurance of protection of the public.

For example, we recognized our shortcomings related to the Three Mile Island accident 25 years ago, and we recognized what should have been done better with the Davis-Besse vessel head degradation. We should be good at identifying our deficiencies; we should also be good at learning from them.

Regulation is a tool of society to achieve predictable and beneficial use of an activity. I have said many times: "Regulation must result in a benefit or it will result in a loss." I dare to say this is particularly true in the case of nuclear power, a technology that is always in the public eye and subjected to public perception, in a still unforgiving environment regarding its performance.

Good regulation provides for the proper exercise of democratic and free-market processes to enhance the common good. It is established to provide a framework that allows for the conduct of individual, industrial, commercial, financial, and other activities. Although regulations restrict, regulation should not deter beneficial activities, but frame them and guide them. Thus, the minimal amount of regulation that achieves the primary objective is best for our society.

Poor regulation, on the other hand, provides too few or too many controls, focusing more on restricting, limiting, and controlling, losing sight of the common good. This is in direct contradiction to the fundamentals of a democratic society and the free market. Poor regulation can create the illusion of being "protective" while stripping freedom, all the way to the individual.

It is frequently too easy to do a little more "regulation," to appear a bit more "protective," and to add another ounce of "conservatism." More regulation can appear enticing. I am convinced that the right goal should be to have less but better regulation. I believe this to be true because we have powerful self-correcting forces that will act promptly in favor of the people. These self-correcting forces are inherent to democracy itself, and include a free market system and the free flow of information.

And that brings us to our regulatory standard: reasonable assurance of adequate protection of public health and safety. The NRC is not in the business of zero risk. We are responsible for assuring that the risk is understood, that it is managed, and that it is acceptably low. Zero is not an option, it is a disruption. Today, with risk-informed regulatory tools, we know how to mix and match deterministic and probabilistic regulation, how to add requirements and how to decrease the unnecessary ones -- and we have the will to do it. We are learning how to define adequate protection in more precise terms, and to define it in terms that make sense to the American people. In other words, we are quantifying safety and communicating it better.

Directly connected to all of the above is the pressing need to bring state-of-the-art know-how to nuclear radiation technology and energy production, and to develop even newer and better techniques, applications and processes. With this, there is also a need for better, more functional and more realistic safety considerations; and, of course, with them the enabling regulation.

Risk-Informed and -Performance-Based Regulation

This is a year of anniversaries, 50 years of Atoms for Peace; 25 years from Three Mile Island; and even 30 years from the Wash-1400 "Reactor Safety Study," which introduced Probabilistic Risk Assessment, or PRA, as a tool to improve reactor safety analysis. Wash-1400 gained prominence with the Three Mile Island accident. Following the accident, the NRC undertook a careful and retrospective analysis of its regulations and regulatory practices in the "NRC Special Inquiry." 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 most of their recommendations, and agree with their statement that the transition to an accident-sequence-oriented approach would be "long" and "painful." It should not have been that long or that painful to achieve a risk-informed regulatory structure, but it has been. The wheels of nuclear regulatory progress turn slowly, but they are accelerating.

In 1995, nearly nine years ago, the Commission issued a formal Commission Policy Statement supporting the increased use of PRA in a manner that was well integrated with engineering approaches, including defense-in-depth, and with operational safety experiences. This integration defines risk-informed regulation. We have made significant progress in the use of PRA since 1995, but we are far from done. Further progress has been achieved by combining the concept of risk-informed regulation, where appropriate, with a performance-based approach to produce the framework of risk-informed and performance-based regulation. A performance-based regulatory approach achieves defined objectives and focuses on results. It differs significantly from a prescriptive approach in which licensees are provided detailed direction on how those results are to be obtained. It has been a long road; but that's our history and we cannot change it. We do have the opportunity to change the future, and I submit to you that we have the obligation to do so.

Two major steps on the road to a risk-informed and performance-based regulatory framework are close at hand, and they are important, practically and philosophically. I am talking about 10 CFR 50.69 and 50.46. The technical information and analytical methods are available and the will to

change is strong. Risk-informed decision-making is an everyday tool for the nuclear industry and the NRC. Risk and risk configuration management is calculated every day and used in operational safety decisions. Why not in the basic design requirements too? We have a sufficient understanding of the probabilities and consequences to be able to progress to the next rational level of regulation to improve reactor safety.

For the emergency core cooling system and LOCA proposed rule, 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 a severe accident management program, rather than as the design basis accident. Effectively, the current Large-Break LOCA would not be a design basis accident when utilizing a risk-informed approach. With this alternative approach, 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. The commitment to go forward with 50.46 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 some of the important decisions to be made. We have found this approach very useful in the past. The re-definition of the Design Basis LOCA is just one step, but a very important step, in the effort to revise the regulatory requirements to be more risk-informed and more broadly coherent.

Integration of Safety/Security/Emergency Preparedness

I mentioned reactor safety, physical security and emergency preparedness earlier. I see these areas as a tightly connected triad -- three intertwined areas, in which the programs, and their regulatory requirements work in an integrated, synergistic way to protect public health and safety. In fact, it is the holistic, functional combination of reactor safety, physical security, and emergency preparedness that provides the basis for assuring public safety.

The relationship among these three areas can be understood by looking at their contributions to overall protection provided through defense-in-depth. The concept of defense-in-depth is a centerpiece of our approach to ensuring adequate protection of public health and safety. Defense-in-depth calls for, among other things, high quality design, fabrication, construction, inspection, and testing; multiple barriers to fission product release; redundancy and diversity in safety equipment; and procedures and strategies to address the expected as well as the unexpected. It must incorporate the dynamics of risk-informed and performance-based decision making. Or better: use risk-informed and performance-based regulation to add realism to defense-in-depth conservatism.

I want to share with you my thoughts on the interrelationships among reactor safety, physical security, and emergency preparedness and their importance to our present focus on mitigation of potential terrorist threats. For example, security concerns, including terrorist threats, raise many of the same issues involved in avoiding and mitigating reactor accidents. Potential initiating events, safety functions, safety (and often non-safety) equipment and procedures, and design basis and severe accident management guidelines all converge to a simple postulate: shut down the reactor, cool the core, and maintain barrier integrity. These are things we know how to do well and should be able to do regardless of the initiating event.

Likewise, it is clear that such system requirements as redundant emergency core cooling systems, redundant and diverse heat removal systems, fire protection features (including separation and suppression systems), and station blackout capabilities (either additional AC power sources or coping capability without AC power) provide built-in means of dealing with attempted attacks on nuclear reactors. And lastly, the emergency procedures and severe accident management strategies developed for reactor accidents also provide means for mitigating the potential consequences of terrorist attacks should they occur. The U.S. nuclear industry has utilized emergency procedures and severe accident management strategies to implement enhancements required by the NRC's security orders of February 25, 2002, because these procedures and strategies are so well suited to be effective against a broad range of events involving possible terrorist activities.

With regard to emergencies, both on-site and off-site mitigating measures will be taken. When the defense-in-depth procedures and strategies are used on-site, they are generally considered part of the reactor safety approach; when they go beyond the plant boundaries, they are generally considered part of "Emergency Preparedness." In treating emergency preparedness as another level of defense-in-depth, we are recognizing it as an integral part of our approach to protecting the public. Reactor fuel, reactor coolant systems, containment, emergency preparedness -- these are four barriers, each one complementing the others, and each one designed, tested, and inspected to provide a reasonable assurance of protecting the public and the environment from radiological releases.

Realistic Conservatism

I have used the term realistic conservatism a few times; let me explain what I mean. I am convinced nuclear regulation now needs to be anchored in realistic conservatism (or conservative realism), and especially so if we are to avoid the twin pitfalls of under-regulation and over-regulation. I see realism and conservatism as enabling factors for safety and reliability.

For purposes of simplicity, I use "conservatism" in the sense of preserving adequate safety margins, and I use "realistic" in the sense of being anchored in the real world of physics, technology and experience. Let me now turn to what I mean by "realistic conservatism": it combines the essence of the above-mentioned definitions, and uses prudence and hard-headed common sense, firmly grounded in real-world conditions, coupled to a commitment to make informed decisions and move on. The consistent implementation of these sets of conditions and outcomes is not easy; nevertheless, it is what is demanded from a nuclear regulatory agency in the 21st century: the application of safety margins using safety-engineering value judgments, aided by risk analysis methods. However, I believe that it is essential for an effective safety program to apply safety margins in a thoughtful and consistent manner. When engineering margins are applied to input parameters, they can distort our understanding of what is truly important. Safety margins are better discerned when they are applied at the decision-making stage rather than at the analysis stage of an issue. The overall effect of the safety margin is better understood and more meaningful when done in this manner.

Safety Management

Now let me turn my attention to Safety Management. Safety Management refers to the integration of three interrelated elements:

First, a functional and executable commitment to operational, maintenance and engineering safety, imbedded in every activity of the organization,

Second, the technical expertise that is applied where and when it should be; able to receive, process, form and communicate technical issues, cognizant of safety functions and safety systems, with licensing and regulation as boundary conditions but taken beyond them by the pursuit of safety and reliability.

Third, the people, programs, and processes to implement a safety program effectively.

Simply stated, safety management involves commitment to safety, the technical expertise to understand what is important, and good management to put the commitment and expertise into action. These elements taken together achieve the requisite adequate protection we demand and the reliability the nuclear industry needs.

I recognize that safety management is not easy; and that they are difficult and complex situations, issues and decisions that both regulators and licensees need to face. But I also recognize that these difficulties are manageable when we have a clear understanding of what is important and what is not; and when we have policies, programs and practices which recognize and appropriately address what is important and what is not; and when we have talent, training, and tools to help us implement these concepts. The NRC supports a regulatory approach in which safety management is implemented through commitment, competence, and the appropriate application of resources - Commitment to doing the right things - knowing "what the right things" are, and the capability to "reduce them to practice" through the application of appropriate resources.

Summary

A key, real and present crisis of our times was clearly portrayed by George Gilder when he stated:

> "It was [is] the survival of unprecedented multitudes of human beings at ever increasing standards of living, together with a new intolerance toward the persistence of conditions of poverty that had previously been accepted as inevitable."

In many ways, this succinct yet poignant statement expresses a fundamental social, political and economical issue confronting mankind, because it is a root cause of many of today's great problems,

and it has to be addressed with urgency and with solutions. And strongly tied with economic development, quality of life, health and safety is the global issue of environmental protection.

I happen to believe that energy, well distributed and affordable, is one of the key solutions to the existing crisis. And, I also believe that nuclear energy, safely deployed, can be part of the solution. Yet, for nuclear power to occupy its rightful place in the energy portfolio of the world, much work is still needed. This work is a shared responsibility.

Every nuclear operator needs to be committed to safety first and foremost; only through effective safety management can reliability and productivity be achieved. Every nuclear regulator is given a mandate to enable the beneficial uses of nuclear energy and radiation, and entrusted with the responsibility of assuring protection of the public and the environment. We know that the mandate and the responsibility are compatible and doable.

With this in mind, I am convinced that 21st century nuclear regulation needs to be driven by a thoroughly integrated set of safety concepts, a seamless fabric, a construct which includes risk-informed and performance-based regulation; which treats reactor safety, physical security and emergency preparedness in a holistic manner; which employs realistic conservatism in analysis and employs safety management in operational decisions. I see this regulatory construct as a fundamental, enabling factor for the safety and reliability of the existing and future nuclear power plants. And it not only has to be done well, it has to be communicated to decision-makers and the public very well!

The U.S. Nuclear Regulatory Commission is committed to fulfill its mandate and discharge its responsibility in a manner that fits the changing needs of our people and for their common good.

I want to thank you for the opportunity to share my views with you and wish you a successful conference, safety and reliability.

NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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Spent Fuel, A "Critical" Element of the Nuclear Fuel Cycle

The Honorable Jeffrey S. Merrifield
Commissioner
U.S. Nuclear Regulatory Commission
at the
Dry Storage Fuel Forum 2004 Conference
Naples, Florida

May 3, 2004

Introduction

Good morning. Thank you for the opportunity to be one of the keynote speakers for this session of your conference. This morning I would like to discuss what I believe will be one of the greatest challenges in the history of the Nuclear Regulatory Commission: the review of an application for a high-level waste repository. While the Commission has been preparing for this challenge for many years, there is always a frenzy of last minute preparations at the dawn of any great landmark occasion. With the Department of Energy (DOE) representing that it will submit an application for a High-Level Waste repository at Yucca Mountain by December of 2004, both the NRC and DOE are actively preparing for that submission.

Today, I would like to discuss our role as the regulator of DOE. To put the significance of our role into perspective, it's useful to consider that DOE has a strong presence in Washington with more than 25 times the workforce of the NRC, and a budget of nearly $25 billion, compared with the NRC's budget of $600 million. Like David faced with the proverbial Goliath, our agency is faced with a significant challenge dealing with a much larger agency.

In addition to discussing this challenge, I would also like to take this occasion to review the multiplicity of fuel cycle activities in which the NRC is currently involved.

The Atomic Energy Commission

The DOE application will mark a significant change in the relationship between the NRC and DOE. Before the NRC was created, nuclear regulation was the responsibility of the Atomic Energy Commission (AEC). Beyond its principal role as the steward of our nation's nuclear stockpile, the AEC was charged by Congress with the mission of encouraging the use of nuclear power as well as regulating its safety. By 1974, however, the AEC had come under such strong attack for its internal conflicts of interest that Congress decided to abolish the agency. Both supporters and critics of nuclear power agreed that the promotional and regulatory aspects of the AEC needed to be assigned to different agencies. As a result, the NRC was created as an independent agency in 1975 and assumed the regulatory responsibilities for civilian uses of nuclear material from the former AEC. DOE, for its part, was the agency that inherited the AEC's promotional function.

To date, the NRC and DOE have coexisted and interacted as separate federal agencies. There are slight overlaps in jurisdiction between the two agencies, but it is rare that NRC has been in the position to regulate DOE. Congress has periodically required pilot programs to evaluate if the NRC should serve as the external regulator of DOE, although none of these pilots resulted in a definitive conclusion upon which both agencies could agree. For my part, I think the NRC could do a very good job of providing external regulation for a broad range of DOE activities. Nonetheless, in my view this has never gone beyond the pilot phase, principally because there remains within DOE a strong reluctance to be subject to external regulation. This 30-year relationship is about to change. Congress declared that the NRC will regulate any high-level waste repository.

For the NRC, reviewing a high-level waste repository license application will be a much larger licensing project than we are used to, but for the most part, the NRC will be acting in its traditional role as the regulator. DOE on the other hand will have to take on the unfamiliar role of an NRC license applicant. In order for the NRC and DOE to meet Congress' expectations, DOE must shift from the role of an independent operator to the role of NRC license applicant. The rules of our interaction with DOE have changed and the sooner DOE's managers and staff come to terms with this, the smoother this application process can proceed.

This change will clearly prove challenging for the DOE to accept, but they have no choice in the matter. Until now, DOE has been an independent actor, unilaterally determining what was necessary for their programs and implementing those determinations without interference from any other agency. Now, the NRC will be questioning their decisions and analyses and requiring that they submit very detailed information in support of their application just as the NRC does with all other license applicants. Everything DOE submits must be sufficiently descriptive to convince both the NRC and the public that their proposals are protective of the public health and safety.

Although we have not yet received the license application, we have received other materials from DOE for NRC review. Some of these have been of insufficient detail or have contained technical problems, and the NRC has been working with DOE to improve the standard of submitted documents. This experience is somewhat analogous to that of the early pioneers of spent fuel cask production. The applicants for the original spent fuel cask designs were unfamiliar with how to deal with a government regulator or the NRC licensing process. Early on, there were a number of problems that were solved only after the applicants better understood what the NRC required in an application and the NRC better articulated these requirements. To be perfectly frank, some of the early problems were not resolved

until power reactor licensees, who are very familiar with the NRC licensing process, became more directly involved in the cask certification process.

We, the NRC, are again struggling to become accustomed to regulating a new licensee, as well as becoming accustomed to the idea that DOE must be treated like any other licensee. The bottom line in this situation is that we cannot accept a half-hearted effort from any of our license applicants, including DOE. Our ability to meet the 3-4 year application review deadline, which has been mandated by Congress, is dependent on DOE submitting a high quality application. The NRC cannot be held responsible if DOE fails to meet this challenge. We are working diligently to meet the challenges facing the NRC, and I am confident we will be ready by December to perform an efficient, effective and timely review of the license application.

NRC Preparations for the License Applications

Turning inward toward our Agency, I am pleased to say that the NRC has been working full bore to prepare for DOE's application. Staff in almost every office of the agency are working diligently to ensure we have the appropriate infrastructure in place to support NRC's review. Once the application is docketed, the NRC must conduct extensive technical reviews, as well as public hearings which will be overseen by the Atomic Safety and Licensing Board. After completion of the hearings, the Board will forward its initial decision to the Commissioners for their review. The NRC is engaging the challenges presented by this process head-on.

The technical staff who will be responsible for reviewing the application are currently familiarizing themselves with the key technical issues that will be part of the application review, as well as attending technology exchanges with DOE to enable them to understand DOE's submission and to formulate questions on the application materials. They are also participating in public outreach activities and tribal workshops in the state of Nevada. Concentrated efforts are also being made to hire experts in technical areas where the NRC does not already have staff available.

The legal staff who will be responsible for representing the NRC in the public hearings are also gearing up for receipt of DOE's application. The Office of the General Counsel recently created a High Level Waste division that currently contains four attorneys dedicated to the Yucca Mountain project. The number of attorneys in this division will grow over the next two years to an ultimate total of twelve attorneys. The legal staff is also working to become more familiar with the technical and legal issues that are likely to be the subject of litigation, and counseling the staff in the application of NRC's High Level Waste regulations.

The Atomic Safety and Licensing Board (ASLB) faces the challenge of presiding over multiple, in-depth hearings related to the Yucca Mountain application. There will most likely be at least three panels simultaneously handling the numerous contentions expected in the hearing. To meet these resource needs, the ASLB will be hiring approximately four new legal judges and four new technical judges. The ASLB is also actively working with the agency's information technology staff to establish the Digital Data Management System (DDMS). The DDMS is a state of the art information management system that will allow any document or piece of evidence submitted in the case to be pulled up electronically at desktop computers in the hearing room. It is a web-based system with an audio-visual component that will allow real time court reporting and webstreaming so those who

cannot be present in the hearing room can have real time access to the proceedings, while also allowing parties access to information from any computer on which they have access to the internet.

Agency staff involved with information technology are working very hard to ensure that the Licensing Support Network (LSN) is up and running in time to receive documents submitted by DOE, the NRC, and other parties participating in the public hearings. The LSN is an electronic information management system that will hold documents related to the Yucca Mountain proceeding so that parties have access to those documents at any given time. The LSN is designed to provide full text search and retrieval access, as well as providing for electronic submission of filings by the parties and orders and decisions of the Atomic Safety and Licensing Board. The LSN is the largest database ever created by the agency and it poses many new technical hurdles that must be tackled by the staff in the near future.

Finally, the Commission is readying itself for receipt of the application and related legal proceedings. One major step we have taken is to establish the Commission Adjudicatory Technical Support Program. This division is home to the technical experts that will advise the Commission during its review of the Atomic Safety and Licensing Board's initial decision on the application. These staff members will be segregated from the rest of the agency to prevent any predecisional interactions between them and those staff performing the initial review of DOE's application. This is necessary to guarantee that the Commission's final decision on the application is impartial and untainted by improper communications between the Commission and the staff conducting the first-line review of the application.

All of these activities are aimed at achieving a fair, efficient and timely review process. This is the most significant application we have received in the history of the NRC, and we will be ready to meet the many challenges that such an application is likely to generate.

Other Fuel Cycle Activities

Today I would also like to highlight for you NRC's activities related to the nuclear fuel cycle. Spent nuclear fuel storage and transportation activities are extremely important to support the overall national picture of nuclear power. We currently have several applications in-house that could have significant impacts on fuel fabrication and storage in the U.S.

Louisiana Energy Services has submitted an application to build a new centrifuge enrichment facility in New Mexico, and we anticipate receiving a second equivalent application from the U.S. Enrichment Corporation in late summer. This is a significant step forward in fuel enrichment in the U.S. considering that there is currently only one plant operating in Paducah, Kentucky, and it utilizes gaseous diffusion technology rather than centrifuge technology. These are important applications and they will receive a focused and disciplined review by our agency.

Currently, we are reviewing an application filed by Duke, Cogema, and Stone and Webster to operate a mixed oxide fuel fabrication facility in South Carolina. If approved, this facility would disposition 25 metric tons of weapons grade plutonium into mixed oxide fuel, which could then be used in commercial reactors. There are both technical and regulatory issues associated with using MOX fuel that the NRC and the industry must resolve before this endeavor can move forward.

We are also reviewing an application from NFS Erwin to operate a blended low-enriched uranium facility in Tennessee that would be capable of dispositioning highly-enriched uranium from our weapons program that could also be used as fuel in commercial reactors. The NRC has issued a notice of opportunity to request a hearing and the staff has prepared the necessary hearing file. If the schedule follows those of similar hearings, the entire process will take approximately one year to complete.

Private Fuel Storage has submitted a first of a kind application for an Independent Spent Fuel Storage Installation (ISFSI), not co-located with a reactor, to be built on the lands of the Goshute Indian tribe in the State of Utah. This facility would be capable of storing 4,000 spent fuel canisters until a permanent repository can be completed. Hearings on all environmental and safety contentions were held in 2000 and 2002. The sole remaining issue involves aircraft crash hazards which will be the subject of hearings to be held later this year. The Commission should make a final decision on this application later in calendar year 2004.

Dry Cask Storage and Transportation Activities

At the moment, the NRC regulates 30 operating independent spent fuel storage installations. This number has more than doubled from what it was about five years ago. Based on current projections, there could be approximately 50 independent spent fuel storage installations by 2010. One indication that this projection is accurate is the continued interest the NRC has experienced in new cask designs from the industry. I would like to note that the dry cask storage industry is a maturing industry which is producing robust and safe products.

To date we have certified 14 cask designs, submitted by 5 vendors, that are approved for storage of spent fuel. Some of these designs are dual purpose and are approved for transportation as well as storage. Evolving cask designs are pushing the technical envelop and require that a more detailed technical analysis be performed by NRC staff when reviewing new design applications. This requires considerable NRC resources, as well as resources on the part of the applicant. In addition, the public is exercising its right to a hearing for some sites, which can also be resource intensive. A few notable examples of recent site specific license applications which have received considerable public interest and have typically involved significant, technically complex issues include Private Fuel Storage, Diablo Canyon, Humboldt Bay, and the Spent Fuel Facility at the Idaho National Engineering and Environmental Laboratory.

Emerging technical issues that evolve from new cask designs must be addressed to provide our staff with the necessary technical basis to support regulatory decisions on whether to accept or reject applicant requests. This regulatory guidance focuses on ensuring the safety of dry cask storage and transportation of spent nuclear fuel. Some examples of issues in this area that the staff continues to address are:

–High burnup fuel thermal issues
–Allowance for burnup credit
–Moderator exclusion for transport

I expect all three of these technical issues will be discussed at this conference, if not in direct presentations, then at least in the halls during the workshop. I will note that for high burnup fuel

thermal issues and allowance for burnup credit, the NRC has provided partial burnup credit but needs more data to justify further credit. The issue of moderator exclusion for transport is more complex in that it will require a change in the philosophy of the NRC. Up until now, our philosophy has been that criticality will not occur even if water should get into the transportation cask. If the Commission were to approve excluding the moderator for transport, it would allow each cask to transport more spent fuel, but it would also allow for the possibility of a criticality if sufficient water were to get into the cask. Moderator exclusion would require a sound technical basis to remove the requirement with associated assurances that a cask would not flood with water after a severe accident, and it would also need to be addressed through rulemaking.

In response to the event of September 11, 2001, the NRC has been evaluating the response of spent fuel storage casks and transportation packages to a terrorist event. I am limited on any details that I can discuss of these classified studies in a public, open forum, but I can assure you that these studies are receiving high priority attention by both management and staff. These studies are to be completed this year, and based on their outcome, may or may not result in staff proposed mitigative measures. At an appropriate time, we will communicate with the industry our assessment of the results of these analyses, and will interact with industry on implementing any potential mitigative measures that the Commission believes are necessary.

As I mentioned earlier, the Commission is actively preparing for a license application from DOE on Yucca Mountain. This action will trigger significant interest in transportation of spent fuel. The Commission has committed to Congress that we will perform some type of package performance study which will be a full scale test of a transportation cask or casks. NRC staff conducted several open public meetings soliciting input on what type of tests should be performed. The meetings had wide ranging stakeholder response including national, state, and local governments as well as several interest groups. The staff submitted a proposal which addressed all the public concerns and the action is now up before the Commission for a decision. The Commission is currently considering various options for the package performance study, taking into account significant stakeholder comments, study objectives, use of study results, and costs.

In closing, I would like to recognize that this assembled group is involved in a dynamic aspect of the nuclear industry. I would commend you for the significant improvement you have made in addressing regulatory issues in an appropriate manner. You have technical issues that must still be addressed, but I am sure you will strive to be successful in this area as well. Thank you again for your time and attention.

NRC Announcement

April 21, 2004 - Staff Changes: Directors of Communications and Office of Public Affairs Named

The Chairman has named William N. Outlaw as the Director of Communications and Eliot B. Brenner as the Director of the Office of Public Affairs. Mr. Outlaw is currently on board and Mr. Brenner will be here shortly. More details on these appointments are contained in the press release below.

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NRC NEWS
No. 04-045
April 21, 2004

NRC NAMES DIRECTORS OF COMMUNICATIONS AND OFFICE OF PUBLIC AFFAIRS

The Nuclear Regulatory Commission has named William N. Outlaw as its Director of Communications, a newly created position, and Eliot B. Brenner as the Director of its Office of Public Affairs. Both are veteran communications professionals.

Outlaw previously worked as Associate Administrator of Public Affairs for the Federal Highway Administration, part of the U.S. Department of Transportation, from April 2002 until last August. Before that, he served for nine years as the Director of Communications for The Road Information Program, a non-profit transportation research group.

In addition, Outlaw has worked as press secretary to the late U.S. Senator Strom Thurmond (R-S.C.) and as a press officer at the U.S. Agency for International Development. His journalism experience includes working as a reporter for the Washington Times, as a reporter and editor for the Associated Press in North and South Carolina.

Outlaw holds a bachelor's degree in journalism and a master's degree in mass communications from the University of South Carolina. He also served in the U.S. Air Force, including a tour of duty in Vietnam.

Last August, NRC Chairman Nils J. Diaz announced the establishment of the position of Director of Communications. Mr. Outlaw will report directly to the Chairman and is responsible for oversight of the offices of Public Affairs and Congressional Affairs. He will also provide policy and guidance for communication activities across the agency.

Brenner has worked since early 2001 as a communications consultant, specializing in aviation issues. Prior to that, he served for over four years as the Federal Aviation Administration's Assistant Administrator for Public Affairs, a position that entailed directing external and internal communications for that agency during high profile aviation disasters and air traffic control modernization. In addition, he has worked as a speechwriter for secretaries of Defense and Treasury. He spent nearly two decades working for the United Press International (UPI) news service. During his time with UPI, he served as Senior National Security Correspondent and directed its Pentagon coverage of the Gulf War. He also co-authored a book, "Desert Storm: The Weapons of War."

He holds a bachelor's degree in journalism from Georgia State University in Atlanta and attended Oxford College of Emory University.

Brenner will be responsible for the agency's public affairs program, which involves interacting with the media and members of the public on NRC-related issues, issuing press releases and fact sheets, and providing advice to agency officials, among other responsibilities.

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NRC NEWS

U.S. NUCLEAR REGULATORY COMMISSION

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No. 04-055 May 5, 2004

NRC PROVIDES UPDATE ON REVIEW PROCESS FOR VERMONT YANKEE UPRATE REQUEST

The Nuclear Regulatory Commission today announced it will utilize a new engineering assessment inspection as part of its review of Entergy Nuclear's request to increase the power output of the Vermont Yankee nuclear power plant by 20 percent.

The NRC's intentions are discussed in the agency's reply to the Vermont Public Service Board's (PSB) request for assurances about Vermont Yankee's reliability following an uprate. Although the NRC's regulatory authority does not cover reliability specifically, the agency oversees many safety-related systems and functions that contribute to a plant's reliable operation.

"The agency remains committed to ensuring continued safe operation of Vermont Yankee. I have given the Governor my assurances on this," NRC Chairman Nils Diaz said.

In addition to its substantial uprate review process, the NRC has decided to also conduct a new engineering design inspection, which has been under development for several months to enhance the Reactor Oversight Process. The inspection will provide additional information for the NRC and be responsive to the PSB's concerns. "The NRC staff considered a number of factors, including the Board's request for an independent engineering assessment, and concluded it is appropriate to conduct this engineering inspection at Vermont Yankee," Chairman Diaz said.

The NRC will use the new inspection to proactively identify any latent issues in a nuclear power plant's design, focusing on those components and systems devoted to safety. The design inspection will include an evaluation of changes to the plant's licensing basis to ensure safety margins remain adequate. At Vermont Yankee, the inspection process will involve three weeks of on-site inspections and more than 700 hours of direct inspection time.

The NRC's inspection team of approximately six will include experienced NRC inspectors, some of whom have not had recent oversight involvement with Vermont Yankee, and at least two contractors with experience in reactor design. The agency will share the inspection schedule with Vermont officials to facilitate state representative participation, as allowed by NRC regulation and policy.

The NRC will not approve the Vermont Yankee uprate, or any proposed changes to a reactor's license, unless the agency can conclude the changes can be implemented safely. The full text of the NRC's letter to the PSB is provided.

Mr. Michael H. Dworkin, Chairman
Vermont Public Service Board
112 State Street, Drawer 20

Montpelier, Vermont 05620-2701

Dear Mr. Dworkin:

I a[illegible]esponding on behalf of the U.S. Nuclear Regulatory Commission (NRC) to your letters dated March 15 and 31, 2004, r[illegible]ng the request by Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (Entergy), to amend th[illegible]rmont Yankee Nuclear Power Station license to increase the power level of the facility. In those letters, the Vermont Public Service Board requested that the NRC conduct its review of the proposed power uprate in a way that would provide Vermont a level of assurance about plant reliability equivalent to an independent engineering assessment. The NRC has decided to conduct a detailed engineering inspection that we believe will be appropriate for addressing our oversight responsibilities and is also responsive to the Board's concerns. This inspection will be performed as part of a new engineering inspection program that the NRC has been developing to enhance the Reactor Oversight Process.

NRC regulations and its oversight process focus on ensuring nuclear safety, whether the facility is operating at power or shut down. The NRC's statutory authority does not extend to regulating the reliability of electrical generation. The NRC recognizes, however, that there is some overlap between attributes that result in safe operation and those that contribute to overall plant reliability.

The Commission understands that the Board is concerned about the reliability of Vermont Yankee following an increase in power level, especially in light of operational issues that have occurred at some other plants that have recently implemented extended power uprates. The NRC recognizes the importance of these issues and is taking steps to ensure that they are satisfactorily addressed to maintain safety. For example, in response to instances of steam dryer cracking at some boiling water reactors, outside technical experts are assisting NRC staff in performing an audit of General Electric's analyses related to steam dryer performance and specific issues related to Vermont Yankee. We continue to engage the industry to ensure resolution of these issues and will consider additional regulatory action, if needed.

The NRC's established review process for power uprate applications is independent, thorough, and comprehensive. A description of the review process is enclosed. Engineering assessments have always been an integral part of the NRC's safety activities. Under our current Reactor Oversight Process, NRC resident inspectors and regional specialists routinely evaluate the work performed by the licensee's engineering organization to determine whether engineering analyses ad[illegible]ately support safe operation. Over the past several months, the NRC has been developing a new engineering i[illegible]on program which we intend to pilot at selected plants. The NRC staff considered a number of factors, including the B[illegible]s request for an independent engineering assessment, and concluded it is appropriate to conduct this engineering inspection at Vermont Yankee. This new engineering assessment inspection incorporates the best practices of the existing and past engineering inspections. The NRC will use this inspection to verify that design bases have been correctly implemented for a sampling of components across multiple systems and to identify latent design issues. The inspection process uses operating experience, risk assessment, and engineering analysis to select risk-significant components and operator actions, and will ensure that adequate safety margins exist. Although the specific sampling of components is still being developed, it will include components from multiple systems that are potentially affected by a power uprate such as the emergency core cooling systems, the containment system, power conversion systems, and auxiliary systems. The inspection will be performed by a team of approximately six inspectors, including some NRC inspectors who do not have recent oversight experience with Vermont Yankee and at least two contractors with design experience. Three weeks of on-site inspection and over 700 hours of direct inspection time will be conducted. This level of effort exceeds that of the biennial safety system design inspection. The Commission believes it is appropriate for addressing the NRC's oversight responsibilities and is also responsive to the Board's concerns. The NRC staff will inform the State of Vermont of the schedule for this inspection to facilitate participation by State representatives, consistent with NRC policy.

The NRC Advisory Committee on Reactor Safeguards (ACRS) will also review the Vermont Yankee power uprate request. The ACRS is a statutory committee that reports directly to the Commission and is structured to provide a forum where experts representing many technical perspectives can provide advice that is factored into the NRC's decision-making process. The NRC staff will provide the results of its review efforts, including relevant inspection findings, to the ACRS for review. After the ACRS completes its review, it will make an independent recommendation regarding whether the proposed power uprate amendment should be approved.

The NRC will not approve the Vermont Yankee uprate, or any proposed change to a plant license, unless the NRC staff can conclude that the proposed change will be executed in a manner that assures the public's health and safety. In response to your request, the NRC staff has taken a close look at proposed inspections and technical reviews to ensure that they will id[illegible] and address potential safety concerns for operating at uprated power conditions. The staff has concluded that the de[illegible] technical review, prescribed in the Extended Power Uprate Review Standard, coupled with the normal associated program of power uprate and engineering inspections, will provide the information necessary for the NRC staff to make a

decision on the safety of operation of Vermont Yankee under uprated power conditions. The Commission believes that the results of NRC reviews and inspections, particularly the new engineering inspection, will assist in addressing the Board's concerns regarding the future reliability of Vermont Yankee. The NRC staff is prepared to meet with the Board to explain further our review process and scope, including the engineering assessment inspection.

Sincerely,

/RA/

Nils J. Diaz

Enclosure:
Established NRC Power Uprate Review Process

Established NRC Power Uprate Review Process

The NRC's established review process for power uprate applications is independent, thorough, and comprehensive. A team of engineers with specialties in a minimum of 17 different technical areas will review the Vermont Yankee power uprate application. The NRC plans to expend about 4000 hours to perform a comprehensive assessment of the engineering, design, and safety analyses related to the uprate. The NRC's "Review Standard for Extended Power Uprates" guides the staff in its review of the application. The Review Standard also provides guidance for determining when and what type of audits should be performed at the plant or vendor sites, as well as for performing our own confirmatory analyses and independent calculations to supplement the review.

The NRC's review of the power uprate application also includes on-site inspections. NRC inspections will review selected activities and modifications made to allow operation at higher power levels to verify that changes to plant systems will support safe plant operation and are in accordance with Vermont Yankee's licensing and design bases. The NRC will use Inspection Procedure 71004, "Power Uprates," as well as a number of our baseline inspection procedures to inspect issues specifically related to power uprate. These inspections will assess changes that could impact the integrity of barriers (e.g., higher flow rates which could increase vibration at specific support points), safety evaluations, plant modifications, post maintenance and surveillance testing, heat exchanger performance, and integrated plant operation. Additionally, our other baseline inspection activities, while not specifically directed at power uprate activities, will provide additional information about Vermont Yankee's ability to operate safely at a higher power level.

The NRC will adjust, as necessary, our technical review, audit plans, confirmatory analyses, or inspection activities if any issues are identified which may have a bearing on our decision on the Vermont Yankee power uprate application. For example, a recent examination of the steam dryer at Vermont Yankee identified cracks on both interior and exterior structures of the steam dryer. The steam dryer is an important component in the process for converting steam to electrical energy, but is not used to mitigate any accidents. The NRC is interested in steam dryer cracking because of the potential for parts to break loose and impact the performance of safety-related equipment. Entergy has indicated that the cracks are in low-stress, low-steam flow areas of the dryer and not in the areas where cracks were observed at other plants that implemented extended power uprates. NRC inspectors monitored Entergy's steam dryer inspection activities, and we will thoroughly review Entergy's follow-up actions as part of our evaluation of Vermont Yankee's request to operate at a higher power level.

Assessment of engineering has always been an integral part of the NRC's safety mission. In the 1990s, the NRC performed extensive reviews at plants across the country to determine if licensees were operating plants in accordance with their design bases. As part of this review, two team inspections were conducted at Vermont Yankee in 1997. One of these inspections was led by staff from NRC headquarters and included six contractors. In 1998, the NRC conducted an engineering inspection, as well as a team inspection to address operability issues resulting from Vermont Yankee's configuration improvement program. Under our current Reactor Oversight Process, NRC resident inspectors and regional specialists routinely evaluate the work performed by the licensee's engineering organization to determine whether the engineering analyses adequately supports safe operation. Our inspectors conduct both routine engineering inspections, as well as an in-depth team inspection every two years. Since the Reactor Oversight Process was implemented in 2000, the NRC has conducted two such safety system design team inspections.

Enclosure

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED 04/21/04

SERVED 04/21/04

COMMISSIONERS

Nils J. Diaz, Chairman
Edward McGaffigan, Jr.
Jeffrey S. Merrifield

In the Matter of	)	
	)	
DUKE ENERGY CORPORATION	)	Docket Nos. 50-413-OLA, 50-414-OLA
	)	
(Catawba Nuclear Station,	)	
Units 1 and 2)	)	

CLI-04-11

MEMORANDUM AND ORDER

This is a license amendment proceeding to authorize the use of four lead test assemblies of mixed oxide (MOX) fuel in one of Duke Energy Corporation's Catawba commercial nuclear reactors. Duke has appealed the Licensing Board's decision to grant the Blue Ridge Environmental Defense League's (BREDL) hearing request. We dismiss Duke's appeal, without prejudice, as premature. We also accept the Board's April 12, 2004 certification of questions regarding a security contention and set out a briefing schedule.

I. BACKGROUND

This litigation arises from Duke Energy Corporation's license amendment request to revise the McGuire and Catawba Technical Specifications to allow insertion of four MOX lead test assemblies at either the McGuire or the Catawba Nuclear Station.[(1)] Following publication of a notice of opportunity for hearing in the *Federal Register*,[(2)] BREDL and the Nuclear Information and Resource Service (NIRS) filed petitions to intervene and requests for hearing. Neither Duke nor the NRC Staff contested the standing of either organization.

On October 21, 2003, both NIRS and BREDL filed supplemental petitions containing, respectively, five and nine contentions unrelated to security. The NRC Staff opposed admission of all except parts of two of BREDL's contentions and all of NIRS's contentions. Duke opposed all of the contentions of both petitioners. The Board heard oral argument on the contentions on December 3-4, 2003.

On December 2, 2003, BREDL submitted four late-filed contentions. Both Duke and the NRC Staff opposed the late-filed contentions on substantive grounds, as well as on grounds of failure to meet the criteria of 10 C.F.R. § 2.714(a)(1) regarding late-filed contentions.[3] The Board heard oral argument on the late-filed contentions on January 15, 2004.

BREDL submitted its security-related contentions on March 3, 2004, after the Commission resolved the parties' disputes a[illegible] BREDL's "need to know" certain safeguards information.[4] The content of the security contentions is not at issue in D[illegible] appeal.

The Board issued its order on standing and non-security contentions on March 5, 2004.[5] The Board found that NIRS had submitted no admissible contentions and thus denied NIRS's request for a hearing. The Board stated that portions of several of BREDL's contentions were admissible. The Board then "consolidate[d], reframe[d], and admit[ted]" the following contentions:

Contention I: The LAR [license amendment request] is inadequate because Duke has failed to account for differences in MOX and LEU [low enriched uranium] fuel behavior (both known differences and recent information on possible differences) and for the impact of such differences on LOCAs [loss of coolant accidents] and on the DBA [design basis accident] analysis for Catawba.

Contention II: The LAR is inadequate because Duke has (a) failed to account for the impact of differences in MOX and LEU fuel behavior (both known differences and recent information on possible differences) on the potential for releases from Catawba in the event of a core disruptive accident, and (b) failed to quantify to the maximum extent practicable environmental impact factors relating to the use of the MOX LTAs [lead test assemblies] at Catawba, as required by NEPA.

Contention III: The Environmental Report is deficient because it fails to consider Oconee as an alternative for the MOX LTAs.[6]

Duke appealed the Board's decision.[7] The NRC Staff supported and BREDL opposed the substance of the appeal. BREDL also argued that Duke's appeal is premature and requested that it be held in abeyance pending issuance of the Board's decision on the security contentions.

A[illegible]uke filed the instant appeal, the Board issued an order on the five security contentions BREDL filed on March 3, 2004.[8] The Board certified Security Contention 1, along with associated questions, to the Commission, admitted one reframed contention, and denied the remaining three.[9]

II. DISCUSSION

Today we hold that Duke's appeal is premature and, therefore, dismiss it without prejudice to a later timely appeal. We also accept the Board's certification regarding one of BREDL's security contentions. We turn first to a discussion of Duke's interlocutory appeal.

A. Duke's Appeal

Duke stated that it appealed pursuant to 10 C.F.R. § 2.714a(c).[10] Under that regulation, "[a]n order granting a petition for leave to intervene and/or request for a hearing is appealable by a party other than the petitioner on the question whether the petition and/or the request for a hearing should have been wholly denied."[11] Although LBP-04-04 does indeed grant a petition to intervene and request for hearing, we hold that the order is not appealable, for it is too early to tell if BREDL's petition should have been "wholly" denied. As explained below, to be appealable under § 2.714a(c), the disputed order must dispose of the entire petition so that a successful appeal by a non-petitioner will terminate the proceeding as to the appellee petitioner. But at the time Duke filed its appeal, the Board had not yet ruled on any of BREDL's security contentions.

For the Board's order to have been appealable when Duke filed its appeal, we would have to interpret § 2.714a(c) as granting a right to appeal any hearing request the Board grants erroneously, whether or not the Board rules on the entire peti[illegible].[12] Although it was only a partial ruling on BREDL's petition, LBP-04-04 did specifically grant the petition to int[illegible]e, and it ruled on both standing and admissibility of contentions. But, before Duke's appeal, BREDL had submitted thr[illegible]oups of contentions, the Board in LBP-04-04 had granted a hearing based on the first two groups, and the third

group remained pending. By appealing LBP-04-04, Duke implicitly argues that the appealable question is whether the Board should have granted a hearing on the basis of the subset of materials the Board actually considered in making its incomplete ruling on BREDL's petition to intervene. Under this view, the Board's continued consideration of other pending contentions is immaterial.

Duke's apparent conception of Section 2.714a(c) is not incompatible with the language of the regulation. An authoritative Appeal Board decision, issued 17 years ago in the *Shoreham* proceeding, held that appeals lie only when a party challenges a Licensing Board's dispositive ruling on the entire petition to intervene:

[A] party may appeal from the acceptance or rejection of contention(s) at the threshold if, but only if, such acceptance or rejection controlled the Licensing Board's disposition of the petition for intervention advancing the contention(s). Thus, for example, a would-be intervenor may appeal immediately the rejection of *all* of its contentions and the resultant denial of its petition. . . . Conversely, in circumstances where an intervention petition is granted on the strength of the acceptance of one or more of the contentions set forth therein, another party to the proceeding may appeal at once if its claim is that all of the contentions should have been rejected and the petition therefore denied.[(13)]

We agree that, for a hearing petitioner to take an appeal pursuant to Section 2.714a(b), the petitioner must claim that, after considering all pending contentions, the Board has erroneously denied a hearing. And for a license applicant, like Duke, to take an appeal under the counterpart regulation, Section 2.714a(c), the applicant must contend that, after considering all pending contentions, the Board has erroneously granted a hearing to the petitioner.

Although *Shoreham* presented circumstances different from here,[(14)] we endorse the Appeal Board's interpretation of Section 2.714a. Moreover, two earlier Appeal Board decisions involving attempted appeals of *incomplete* rulings by Licensing Boards are factually similar to the instant case and reinforce our ruling today.[(15)] In those cases, the Appeal Board refused to entertain appeals by license applicants challenging partial Board fulings -- *i.e.,* rulings not considering all pending contentions.

Based on the Appeal Board's rulings -- which continue to reflect the Commission's stance on appeals under Section 2.714a -- the Commission dismisses Duke's appeal without prejudice. When Duke took its appeal the Licensing Board had not yet ruled on BREDL's security contentions. Duke can renew or modify its appeal after the Board rules on BREDL's entire petition; *i.e.*, Duke can appeal on an interlocutory basis if a successful appeal would dispose of the case.[(16)] We turn next to the second matter before the Commission.

B. The Board's Certified Questions

The Board has sought further guidance from the Commission on the admissibility of BREDL's Security Contention 1. Under 10 C.F.R. § 2.718(i),[(17)] the Board has certified the questions "specifically raised in Security Contention 1, and those that arise out of and relate to it, the responses to it, and also to issues addressed in CLI-04-06, as discussed [in the Board's order of April 12, 2004.]"[(18)]

Consistent with our customary practice,[(19)] we accept the Board's certification and seek briefs on the admissibility of Security Contention 1 and on what the Board characterized as "several pertinent related questions." The briefs shall not exceed 30 pages and should be filed simultaneously by May 5, 2004. Reply briefs, containing only rebuttal, shall not exceed 10 pages and should be filed simultaneously by May 12, 2004. The Board shall move forward expeditiously on all other issues in this adjudication.

III. CONCLUSION

For the foregoing reasons, the Commission (1) *dismisses* Duke's appeal *without prejudice*; (2) *accepts* the Board's certification; and (3) *invites* the parties to submit briefs.

IT IS SO ORDERED.

For the Commission

/RA/

Annette L. Vietti-Cook
Secretary of the Commission

D[illegible] at Rockville, Maryland,
the 21st day of April 2004

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1. MOX is a mixture of uranium and plutonium oxides. As part of a cooperative program with the Russian Federation, the U.S. Department of Energy plans to dispose of weapons grade plutonium by converting it into MOX fuel for commercial nuclear reactors. Under contract with DOE, Duke initially intended to test four MOX fuel assemblies in one of its Catawba or McGuire reactors. Duke later narrowed its request to placing the four test assemblies into the 193-assembly core of one of the Catawba reactors. After irradiation, the MOX assemblies will be tested to verify their properties. Prior to "batch use" of the fuel, a subsequent license amendment will be necessary.

2. *See* "Duke Energy Corporation et al., Catawba Nuclear Station, Units 1 and 2; McGuire Nuclear Station, Units 1 and 2; Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing," 68 Fed. Reg. 44,107 (July 25, 2003).

3. The NRC has recently amended its adjudicatory procedural rules, 10 C.F.R. Part 2. *See* "Changes to Adjudicatory Process: Final Rule," 69 Fed. Reg. 2182 (Jan. 14, 2004). The new rules of procedure apply to proceedings noticed on or after Feb. 13, 2004. Thus, the NRC's adjudicatory regulations which were in effect before Feb. 13, 2004, apply to the Duke proceeding. Except as otherwise noted, references to 10 C.F.R. Part 2 in this order cite the earlier version of the rules.

4. *See Duke Energy Corp.* (Catawba Nuclear Station, Units 1 and 2), CLI-04-06, 59 NRC __ (Feb. 18, 2004) for detailed information regarding these disputes.

5. LBP-04-04, 59 NRC __ (Mar. 5, 2004).

6. *Id.* at __, slip op. at 63.

7. Pursuant to 10 C.F.R. § 2.714a(b), NIRS had a right to appeal the Board's decision, but did not do so.

8. On April 8, 2004, BREDL also filed amended contentions on Duke's security plan submittal. The Board has not yet ruled on these amended contentions.

9. *See* unpublished "Memorandum and Order (Ruling on Security-Related Contentions)" (April 12, 2004). This order contains safeguards information and therefore will not be made public.

10. In view of our holding today, we need not address Duke's substantive arguments: (1) that BREDL's late-filed contentions were inexcusably late; (2) that none of BREDL's 13 non-security contentions were admissible; (3) that the Board's "reframing," using (allegedly inadmissible) bits and pieces of BREDL's contentions, exceeded its authority; and (4) that the contentions as reframed were also inadmissible.

11. 10 C.F.R. § 2.714a(c).

12. We often refer to the Statement of Considerations as an aid in interpreting our regulations. For section 2.714a, however, the Statement of Considerations is not illuminating. *See* 37 Fed. Reg. 28,710-28,711 (Dec. 29, 1972). We also note that there is no material change in the language of the corresponding regulation in our new rules. The new rule is 10 C.F.R. § 2.311(c): "An order granting a petition to intervene and/or request for hearing is appealable by a party other than the requestor/petitioner on the question as to whether the request/petition should have been wholly denied." 69 Fed. Reg. at [illegible].

13. *Long Island Lighting Co.* (Shoreham Nuclear Power Station, Unit 1), ALAB-861, 25 NRC 129, 136 (1987) (emphasis in original, citations omitted).

14. *Shoreham* addressed a non-party's attempt to appeal part of a Licensing Board's decision to admit contentions. *See Shoreham*, ALAB-861, 25 NRC at 132.

15. *See Cincinnati Gas and Electric Co.* (Wm. H. Zimmer Nuclear Power Station), ALAB-595, 11 NRC 860, 863 (1980); *Detroit Edison Co.* (Greenwood Energy Center, Units 2 and 3), ALAB-472, 7 NRC 570 (1978).

16. Because the Board's April 12, 2004 order did not rule on admissibility of one of BREDL's security contentions, this case remains unripe for appeal under 10 C.F.R. § 2.714a(c).

17. This regulation empowers a presiding officer to certify questions to the Commission, either in the discretion of the presiding officer or on direction of the Commission.

18. Unpublished order at 33 (Apr. 12, 2004).

19. *See, e.g., Private Fuel Storage, L.L.C.* (Independent Spent Fuel Storage Installation), CLI-01-12, 53 NRC 459, 461 (2001).

EA-03-230 - Waterford 3 (Entergy Operations, Inc.)

April 12, 2004

EA-03-230

Joseph E. Venable
Vice President Operations
Waterford 3
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A WHITE FINDING AND NOTICE OF VIOLATION (NRC Inspection Report No. 50-382/03-007) WATERFORD 3

Dear Mr. Venable:

The purpose of this letter is to provide you the final results of our significance determination of the preliminary "Greater Than Green" finding identified in the subject inspection report. Our preliminary finding was discussed with your staff during a briefing conducted on January 5, 2004. The inspection finding was assessed using the Significance Determination Process and was preliminarily characterized as "Greater Than Green" (i.e., an issue of greater than very low safety significance). The finding involved the failure to establish appropriate instructions and accomplish those instructions for installation of a fuel line for the Train A emergency diesel generator in May 2003. The associated performance deficiency resulted in uneven and excessive scoring of the fuel line tubing that ultimately led to a complete 360-degree failure of the fuel supply line on September 29, 2003, during a monthly surveillance test, rendering the Train A emergency diesel generator inoperable.

At your request, a Regulatory Conference was held on March 8, 2004, to further discuss your views on this issue. During the meeting, your staff acknowledged the performance deficiency and described your assessment of the risk significance of the finding. In a supplemental response, dated March 15, 2004, you provided additional information regarding your risk evaluation of the event. A copy of your supplemental response is enclosed. A summary of the Regulatory Conference was issued March 16, 2004. During the Regulatory Conference you agreed that the failure to establish appropriate instructions and accomplish those instructions for installation of a fuel oil line for Train A emergency diesel generator in May 2003 was a performance deficiency and a violation of your Technical Specifications. However, you took exception to certain aspects of the NRC's evaluation of risk associated with this event. After considering all of the information available, and for reasons explained below, the NRC has concluded that the finding is appropriately characterized as "White."

During the Regulatory Conference, your staff provided an overview of the event and the root cause, described your assessment of the significance of the finding, and provided your regulatory perspectives. We agree with your overall view of the event and the root cause determination. However, we do not agree with the approach that your staff undertook in assessing the safety significance for crediting repair of the Train A emergency diesel generator. With regard to applying credit for repair in this case, the NRC evaluated credit for repair in determining the significance of this performance deficiency. However, the modeling of specific maintenance repair activities in the context of probabilistic risk assessment (PRA) sequences is inherently complex and typically requires detailed analysis with appropriate supporting data. NRC Manual Chapter 0609, "Significance Determination Process," dated March 21, 2003, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," and the ASME PRA Standard provide guidance for modeling equipment repair. The NRC acknowledges that a fraction of PRA

models (including the NRC's SPAR 3i models) credit repair of emergency diesel generators by treating all emergency diesel generator failure modes in the aggregate, irrespective of the failure mechanisms, and establishing a mean or median-time-to-repair (MTTR). In general, these models have MTTR in the range of four to eight hours, which is a significantly longer period of time than that considered in your risk assessment.

The analysis that your staff performed for the emergency diesel generator repair at Waterford 3 deviated from the guidance provided in Regulatory Guide 1.200 and is different from the accepted approach for the use of repair, which addresses the spectrum of failure causes and the distribution of repair times for all causes. The NRC accepts that you have demonstrated the feasibility of accomplishing the repair for this particular failure mechanism under a certain set of conditions. However, in order to credit the repair of the emergency diesel generator fuel line rupture in the risk assessment, it is incumbent upon you to demonstrate the feasibility of accomplishing the repair under a reasonably bounding set of conditions. The NRC found your analysis was based on assumptions that did not appropriately consider the dependencies among those actions as well as human error probabilities. The analysis that was performed for the repair did not present sufficient justification for deviating from the guidance provided by Regulatory Guide 1.200.

During the Regulatory Conference, your staff noted that NRC had previously allowed consideration of manual actions in performing significance determination assessments and therefore the precedent had been set for allowing credit for repair in this specific instance. In reviewing the supplemental information that you provided following the conference (enclosed), each of the examples that you identified concerned situations where the NRC had allowed the use of manual actions for recovery, not repair. There are different approaches that are used for analyzing recovery and repair actions. Recovery actions lend themselves to human reliability assessment techniques, and are in principle acceptable given certain conditions where procedures exist that address the necessary actions, training has been conducted for the existing procedures under conditions similar to the scenario assumed, and any equipment needed to complete these actions is available and ready to use. The repair situation that you faced was quite different than the situations in which NRC has credited recovery actions. Plant operators and maintenance personnel were not specifically trained to make the repair to the EDG fuel supply line under a reasonably bounding set of conditions, there were no specific repair procedures in place, and there was no pre-staged equipment or tools. Also, Regulatory Guide 1.200 does not provide credit for repair actions in which no actuarial data exists, which is the case in this instance.

As a result of non-conservative assumptions in your analysis, including the reasons noted in the preceding discussions, the NRC concluded that you did not make a compelling argument for crediting the repair of the emergency diesel generator in the time assumed in your analysis. As discussed during the Regulatory Conference, the failure of the Train A emergency diesel generator fuel supply line was a stochastic occurrence that occurred after a 2.8-hour run time during a surveillance test. Depending on the operating history of the emergency diesel generator after the performance deficiency occurred, the failure could have occurred in significantly less than 2.8 hours or could have occurred in significantly greater than 2.8 hours. The NRC staff noted that a failure in less than 2.8 hours would have caused a greater increase in the risk estimation than the corresponding decrease in risk estimation associated with a failure following a period of greater than 2.8 hours. We also noted that your analysis did not adequately consider the spectrum of conditions that could occur in a station blackout scenario, some of which may be less conducive to successful timely repair.

The NRC staff agrees that there were conservatisms in our safety assessment for the emergency diesel generator run failure rate and the 4-hour battery depletion time. However, we do not agree that we neglected the 2.8-hour Train A emergency diesel generator run time before fuel oil line failure. Notwithstanding that an earlier failure was possible, the initiating event frequency was adjusted to account for the 2.8-hour run time. Overall, the NRC found that these conservatisms were sufficient to change the NRC 's overall safety significance determination from "Yellow" to "White" for the case in which no repair of the Train A emergency diesel generator is credited.

Therefore, after considering the information developed during the inspection and the information you provided at the conference, as well as the information provided in your supplemental response, the NRC has concluded that the inspection finding is appropriately characterized as White, (i.e., an issue with low to moderate increased importance to safety, which may require additional 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 violation of the requirements of 10 CFR Part 50, Appendix B, Criterion V, is cited in the attached Notice of Violation (NOV). 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.

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 column, 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 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/

Bruce S. Mallett
Regional Administrator

Docket: 50-382
License: NPF-38

Enclosures:
1. Notice of Violation
2. Entergy Supplemental Response

cc w/Enclosures:

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ENCLOSURE 1

NOTICE OF VIOLATION

Entergy Operations, Inc.
Waterford 3

Docket No. 50-382
License No. NPF-38
EA-03-230

During an NRC inspection completed January 5, 2004, 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 Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, and drawings.
>
> Contrary to this requirement, during the overhaul of Train A emergency diesel generator in May 2003, the licensee failed to establish adequate instructions to ensure proper installation of the fuel supply line of Train A emergency diesel generator. This failure resulted in uneven and excessive scoring of the tubing that ultimately led to a complete 360 degree failure of the fuel supply line on September 29, 2003, during a monthly surveillance test.
>
> This violation is associated with a white significance determination process finding.

Pursuant to the provisions of 10 CFR 2.201, Entergy Operations, Incorporated 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 facility that is the subject of this Notice of Violation (Notice), within 30 days of the date of the letter transmitting this Notice. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (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 previously 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 http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room). 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 2.390(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 12th day of April 2004

NEI: Commission `Not Well Served' By Latest NRC Staff Paper on 50.46

Inside NRC
Volume 26 / Number 8 / April 19, 2004

A senior staffer at the Nuclear Energy Institute (NEI) expressed last week the industry's strong dissatisfaction with a recent NRC staff paper to the commission on efforts to risk inform the emergency core cooling system (ECCS) acceptance criteria (10 CFR 50.46) and change the current definition of a large-break loss-of-coolant accident (LOCA).

Anthony Pietrangelo, NEI's senior director of risk regulation, told the Advisory Committee on Reactor Safeguards (ACRS) April 15, that while he has "tremendous respect" for NRC staffers working on this issue, he would be "less than candid" if he didn't tell ACRS that the industry was "extremely disappointed" by the staff paper (Secy 04-37) (INRC, 22 March, 4).

In that paper, the staff said it needed commission guidance on how broad or narrow a rule change redefining the maximum LOCA break size should be. The staff said that the original staff requirements memorandum, dated March 31, 2003, could be interpreted as supporting either a broad or narrow scope rule.

But Pietrangelo said the paper reflected a "cone of silence" that the NRC staff placed on this issue seven months ago when the agency stopped a dialog with the industry on possible 50.46 changes. One glaring omission from the paper, he said, was that there was "no mention of any potential safety benefits" from 50.46 changes. The commission, he said, was "not well-served by this Secy."

The paper, he said, reflects a lot of concerns about what licensees might do if 50.46 were revised. But the industry is under no illusions about the amount of technical work that would be necessary to support specific applications using an alternate pipe break size rather than a break from the largest pipe in the reactor coolant system.

He argued that the 50.46 effort is "sorely in need" of a pilot application using an alternate break size. He raised with the ACRS the idea of taking a risk-informed tack to resolving generic safety issue 191 on the performance of

PWR containment sumps and using that work as a pilot for the LOCA redefinition effort.

The NRC staff, however, is apparently skeptical about how much safety benefit there is to be realized from 50.46 changes. At a briefing April 1 before an ACRS subcommittee on this same subject, NRC's Mark Rubin said that it was "rare to see a risk decrease in risk-informed submittals, but occasionally one sees it."

ACRS member William Shack said at that same meeting that one should not discount the "social benefit" of more electricity production coming from plants that are able to operate more flexibly.

At both the April 1 and April 15 briefings, the NRC staff outlined eight key issues that it was seeking commission guidance on, including:

— What might be the practical effect of removing specific events and structures, systems, and components from the design basis?

— Should the rule be very specific about what can be changed or should it merely provide a process by which changes could be made?

— What level of mitigation capability should be retained for LOCAs that formerly were in the design basis?

— How should adequate defense-in-depth be assured under this rule?

At both briefings, the NRC staff also presented preliminary results from an expert elicitation process that looked at generic BWR and PWR piping and non-piping passive system LOCA frequency distributions as a function of break size and operating time. Among the insights from that process were that complete failures of the smallest plant piping are more likely than the partial failure of larger piping, that aging may have the greatest effect on intermediate-sized piping (6 inches to 14 inches), and that estimating non-piping failure frequencies is more challenging than estimating piping failure frequencies.

The results of the elicitation will be published in a Nureg report. The ACRS is scheduled to receive a briefing in early summer on a draft of that report.
—Michael Knapik, Washington

Staffer Presses on with Dissent on Chemical Safety of MOX Plant

Inside NRC
Volume 26 / Number 9 / May 3, 2004

Although the Office of Nuclear Material Safety &Safeguards (NMSS) has completed nearly all of the tasks to which it committed in response to two differing professional views (DPVs) on chemical safety issues at the planned DOE mixed-oxide (MOX) fuel fabrication plant, the DPV process does not appear to be near completion. The NRC staffer who submitted the two DPVs is continuing to pursue the points he raised, said sources familiar with the proceedings.

However, the filer—Alexander Murray, the lead NRC chemical safety reviewer for the MOX plant construction authorization request (CAR)—is going to another management level to seek redress for his concerns. For one of his DPVs (NMSS-DPV- 2003-01) Murray is elevating his complaint to a differing professional opinion (DPO), a staffer said. In the DPV Murray contends the NRC staff prematurely closed a chemical safety issue—designated CS-5—in the draft safety evaluation report on the CAR.

In the second DPV (NMSS-DPV-2002-03), which deals with the scientific code that DOE contractor Duke Cogema Stone & Webster (DCS) plans to use to model the dispersion of potential hazardous releases at the MOX facility site, Murray has written to top NMSS management to express his dissatisfaction with the current resolution and plans to raise the issue anew when Jack Strosnider replaces Martin Virgilio as NMSS director this week, another staffer said. In both cases, Virgilio had issued a Director's Decision, based on the reports of review panels that he had appointed.

The decisions, issued last year (INRC, 28 July '03, 16; 29 Dec. '03, 13), included follow-up actions for the NMSS division of fuel cycle safety and safeguards (FCSS). In a pair of letters— dated Dec. 23 and Jan. 12, but released to Adams last month— FCSS Director Robert Pierson provided an update on actions to carry out Virgilio's instructions. With regard to DPV-2003-01, Virgilio had said that CS-5 should not be revisited but that NRC should ensure that DCS' application provided adequate information to support the safety rationale for its chemical-safety analysis. In the Dec. 23 letter, Pierson said, "We have reviewed the [DCS] application and concluded that sufficient information does exist to support the regulatory safety decisions we have made involving chemicalsregulated by NRC."

One staffer said he believed Murray did not necessarily have a technical disagreement with the decision to close out the chemical safety issue but was concerned the staff hadn't adequately documented its reasoning. The staffer added that docketing the rationale was particularly important because of the two-stage licensing process for the MOX facility, with the operating license not scheduled to be considered until several years after completion of the CAR. Given the large number of NRC staffers who are near retirement age, many involved in the CAR review may not be there later to assess the DCS request for the operating license, the staffer said. That expected transition elevates the importance of having a clear decision-making trail, he said. But the other staffer suggested that Murray's complaint was more fundamental. The staffer noted that Virgilio had not accepted the DPV-2003-01 review panel's recommendation to reopen CS-5 or open a new

item. According to the first staffer, a DPO review panel has held a "kickoff" meeting, at which the members received key documents.

As a next step, the panel will meet with Murray and possibly others, the staffer said. The panel was appointed by Executive Director for Operations William Travers and will report to him. The chair of the panel is Theodore Quay of the Office of Nuclear Reactor Regulation (NRR); the other members, the staffer said, are NMSS' Walter Schwink, who also was part of the DPV-2003-01 panel, and John Voglewede of the Office of Regulatory Research (RES).

Code questioned

The Jan. 12 letter from Pierson to Virgilio addressed the assignments Virgilio had made with regard to DPV-2002-03, which deals with DCS use of the Arcon 96 scientific code. Murray had questioned the decision to allow the use of what he said was an insufficiently conservative code, but the review panel for that DPV did not support that point. It said that the code is "general in nature" and therefore is "generically applicable to any site including fuel cycle facilities." However, the panel said, "the reasonableness of site specific application" must be determined in each case. Virgilio accepted that conclusion and asked FCSS to ensure that applicability of the code to the MOX site had been demonstrated. In the Jan. 12 letter, Pierson said it had.

One of the staffers said NRR had originally developed Arcon 96 for reactors but that the code had "evolved" in ways that made it more broadly applicable. NMSS was not aware of those developments, he said. Pierson also said his division had carried out the instructions to ensure that staffers who use scientific codes such as Arcon 96 receive sufficient guidance in how to apply them. Murray also said use of the code within NRC should be consistent and coordinated. In response, Virgilio said FCSS should raise that point in the next NMSS-RES "user-need" meeting. In the Jan. 12 letter, Pierson pointed to a Nov. 21 memo from RES Director Ashok Thadani to Virgilio establishing a timetable for developing collaboration between RES and NMSS on automated scientific codes. The estimated completion date for the effort is September 2005, making that task the only one still pending for DPV-2002-03.

Adversarial process

The first staffer also said the MOX DPVs raised a broader issue about the NRC dissent process. He said the current DPV/DPO structure makes the process "intensely adversarial," in large part because senior managers may not know that an issue is controversial until the DPV is filed. These managers can "get blind-sided," the staffer said. Last year, in a Sept. 3 memo on DPV-2003-01, Pierson mentioned a "newly developed FCSS non-concurrence process." An attachment to the memo is a view-graph summary of the new process. One slide suggests that in cases of disagreement, staffers first apply "the usual problem-solving process of discussing the issue with line management." The slide continues, "If a solution to the concern is not agreed between staff and section, branch and division management, then a non-concurrence may be appropriate." But two FCSS staffers involved in the Murray DPVs said last week they were not familiar with that non-concurrence process and did not know its status.
—Daniel Horner, Washington

Industry, NRC Staff Resume Talks to Repair MSPI

Inside NRC
Volume 26 / Number 9 / May 3, 2004

With a push from the NRC commissioners, NRC staffers and industry representatives agreed last month to resume work on the mitigating systems performance index (MSPI), which the staff previously had determined would create more problems than benefits in the agency's reactor oversight process (ROP). A key NRC staffer indicated the agency staff was prepared to go only so far in trying to reach agreement with industry.

The MSPI has been under construction and testing for nearly three years as a potential replacement for the safety system unavailability (SSU) performance indicator (PI). Twenty units participated in a six-month pilot project to capture performance data, which was then analyzed over several months by the staff. Office of Nuclear Regulatory Research (RES) staffers came out in support of adopting the MSPI. It concluded the MSPI offered a fix to the limitations of the SSU PI by providing plant-specific risk insights. The RES staffers noted that the MSPI addresses both the reliability and availability of five plant safety systems, plus their support systems.

The Office of Nuclear Reactor Regulation (NRR) staff agreed with RES staff on the advantages. But it said the policy and technical problems and implementation issues outweighed the benefits. The NRR staff outlined what it found to be MSPI deficiencies in its paper, Secy 04-53. The paper, released April 21, can be accessed on NRC's Adams recordkeeping system under accession number ML040620267. In its paper, the staff was critical of the MSPI's built-in "risk limiter"—the so-called front stop—which prevents the agency from taking action for initial single system failures. The MSPI also does not take into account the risk contribution from external events, internal flooding, shutdown and large early release frequency, the staff said. It would create enforcement "inconsistencies" because of the elimination of the significance determination process for single failures. There would be high costs of implementation and inspection, the probabilistic risk assessments would not be available for public scrutiny, and the complexity of the concept may be difficult for the public to grasp.

The commission directed the staff in an April 8 memorandum to look for "creative and practical approaches" for establishing a more risk-informed, performance-based indicator to replace the SSU PI. The commission suggested that the staff might want to consider eliminating the front stop. It did not specifically instruct the staff to salvage the MSPI, although the commission clearly indicated at a March 24 briefing that it wanted the staff to try to rework the index it already had developed.

At an April 27 meeting, Stuart Richards, chief of the inspection program branch, said the staff would follow the commission's direction and look for creative approaches to accomplish the goal of the MSPI. He also said the staff would adhere to commission's wishes to complete the task in a timely manner. He said the staff planned to collect from stakeholders any concerns about the MSPI and compile them in one document, which will be made publicly available. Anthony Pietrangelo, director of risk regulation at the Nuclear Energy Institute, asked at the meeting whether the staff would try to make the existing MSPI work, or if the plan was to start over. "I'm not saying it's MSPI or nothing at all. But at least let's give MSPI a shot," Pietrangelo said, adding, "It sounds like your effort is going beyond MSPI."

Richards said the staff and industry might be able to arrive at a "middle ground" on MSPI, but if that was not possible, the staff would try to develop another risk-informed indicator.

Donald Dube of RES said his office has dedicated about three full-time equivalents and spent more than $1-million for piloting and evaluating the MSPI. "To abandon that, in my opinion, would be silly," he said. But Richards said he couldn't predict the outcome of the coming discussions. "They didn't say go back and make MSPI work," Richards said of the commission's direction. "They gave us more latitude." Rather, the thrust of the commission memorandum, he stressed, was to get the staff talking with stakeholders immediately and not wait until it had all of its "creative approaches" worked out.
—Jenny Weil, Washington

NRC research staff sets focus on risk-informing regulations

Inside NRC
Volume 26 / Number 8 / April 19, 2004

The phased use of risk analysis to provide "realistically conservative" approaches to reactor regulation is among the top priorities this year for the NRC Office of Nuclear Regulatory Research (RES), RES Director Ashok Thadani said last week.

At an April 13 briefing before the commission on the office's 2003 accomplishments and future activities, Thadani cited emergency core cooling systems, loss-of-coolant accident frequencies, and fuel cladding requirements as areas where regulation might be made more performance-based. RES staff will also provide support for the advanced reactor review and the so-called package performance study on spent fuel casks, which is being conducted by Sandia National Laboratories for NRC.

Analyses of plant security and vulnerability have been "a top priority" since the Sept. 11, 2001 terrorist attacks, Thadani said, and "it seems it will remain that way." But NRC Chairman Nils Diaz later noted that "the bulk of that work is essentially coming to an end" and is "being completed and reviewed." Diaz said he "didn't want to leave the impression that this is an open field," and Thadani agreed.

Despite substantial budget increases in recent years, RES is still understaffed and the attrition rate is "somewhat higher than anticipated," Thadani said. About 12% of research staff has been with the office less than one year. A "corporate memory program" will be implemented this summer to facilitate sharing of expertise with newer employees, he said.

Risk communications stressed

Commissioner Edward McGaffigan emphasized the need to improve NRC risk communication. Bounding engineering analyses sometimes incorporate "many orders of magnitude of conservatism," McGaffigan said, allowing media and advocacy groups to neglect caveats and "excerpt one little nugget" to make a misleading point.

McGaffigan cited a 2001 draft Nureg which analyzed the risk of spent fuel pool fires at plants being decommissioned as one example of "prime material for someone who wants to misuse it" to foster public concern. In March 2003, McGaffigan roundly criticized eight scientists who cited the draft Nureg in their paper assessing the potential consequences of a terrorist attack on a spent-fuel pool (INRC, 21 April '03, 1). Thadani agreed that RES had not focused in the past on communications issues or addressed them systematically. However, adherence to newly established guidelines for NRC risk communication (Nureg-BR/0308), as well as the addition of summaries in plain English to NRC reports, should help, Thadani said.

Diaz identified risk communication as an issue "near and dear to my heart" and stressed that NRC has an obligation to communicate its work to the public in an understandable manner.

NRC vs. licensee PRA models

McGaffigan asked the staff if NRC should continue to conduct plant-specific analyses based on the agency's standardized plant analysis risk (SPAR) models or rely instead on models developed by each licensee.

Mark Cunningham, deputy director of the division of risk analysis and applications at RES, noted a resource tradeoff between development and benchmarking of SPAR models and reviews by NRC regional offices of every licensee model. The SPAR-based approach requires fewer resources and provides more standardization, according to Cunningham.

Asked by Diaz if RES had analyzed differences in results generated by SPAR and licensee models, Cunningham replied that SPAR is benchmarked against licensee models for each plant where it is utilized to discover the most significant differences, and added that those findings would be used to upgrade future versions of SPAR.

Commissioner Jeffrey Merrifield asked about the review of RES research programs conducted by the Advisory Committee on Reactor Safeguards (ACRS) and released in March (Nureg 1635, Vol. 6). "By and large, we agree" with the findings, Thadani replied, adding that RES has begun discussions with ACRS to review the report's conclusions. Thadani cited degraded containments and cable aging as examples of research areas that RES planned to phase out based on ACRS recommendations.

Merrifield also expressed concern about shifting resources from one program to another, noting that RES sometimes allocates far more money to certain programs than had been approved by the commissioners in the annual budget.

McGaffigan suggested that RES consider waiting to begin new, lower-priority research programs until midway through the fiscal year, when it would have a clearer sense of available resources. Thadani noted that emerging safety issues, such as sump debris clogging, sometimes necessitate reprogramming, but he pledged to make the process more transparent to the commissioners.

The commission commended Thadani for his 30 years of work at NRC. Thadani will soon become director for international research and development projects, a newly created position. He will be succeeded by Carl Paperiello, currently the deputy executive director for materials, research, and state programs. The moves are part of a recent, widespread reassignment of senior managers initiated by Diaz (Inside NRC, 5 April, 1).—***Steven Dolley, Washington***

(2)

MOX Fuel Lead Assembly Program

Advisory Committee on Reactor Safeguards
NRC Offices - White Flint, MD

Steve Nesbit
MOX Fuel Project Manager, Duke Power
May 6, 2004

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MOX MIXED OXIDE FUEL PROJECT

Outline of Presentation

⇒ **Introduction**
- **MOX fuel - general**
- **Safety evaluation**
- **Environmental evaluation**
- **Summary**

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MOX MIXED OXIDE FUEL PROJECT

Plutonium Disposition Program

- 1994 National Academy of Sciences Report – surplus weapons material poses a "clear and present danger"
- September 2000 – U.S.-Russian agreement that each country will dispose of 34 tonnes of its surplus weapons grade plutonium
- Approach – fabrication into mixed oxide (MOX) fuel and use in commercial nuclear reactors
- The lead assembly program is an essential element of the plutonium disposition program
  - Required to qualify MOX fuel for use in United States reactors

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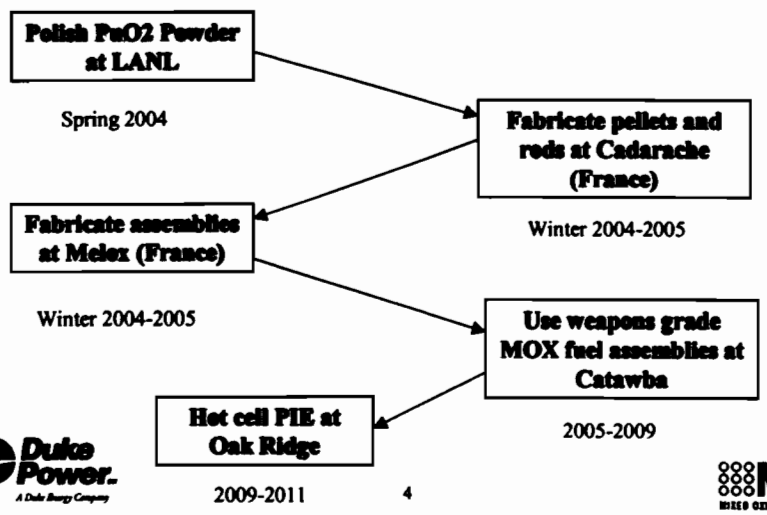
Lead Assembly Program



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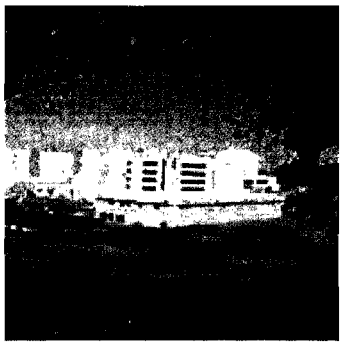
Reactor Use of Lead Assemblies

- Catawba Nuclear Station
  - York County, South Carolina
  - Unit 1 began operation in 1984
  - 3411 MW_{th} pressurized water reactors operated by Duke Power
    - Westinghouse four loop design
    - 193 fuel assemblies in each core
    - Ice condenser containments
- Catawba and McGuire (the four "mission reactors") have a common core and reactor coolant system design



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Irradiation and Examination

Cycle 1

Catawba 1 Cycle 16: Spring 2005 – fall 2006
Prototypical but not limiting power
Poolside post-irradiation examination (PIE)

Cycle 2

Fall 2006 – spring 2008
Prototypical but not limiting power
EOC2 burnup ~48GWD/t

Discharge one or more assemblies
Poolside PIE (normal and extended)
Hot cell PIE

Cycle 3

Spring 2008 – fall 2009
Low power
EOC3 burnup <60,000 GWD/t
Extended poolside PIE and optional hot cell PIE

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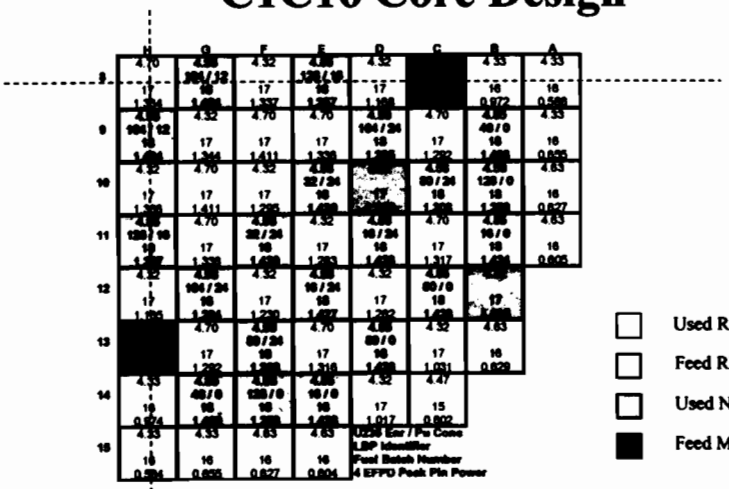
C1C16 Core Design



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MOX MIXED OXIDE FUEL PROJECT

Required Regulatory Approvals

- Duke topical reports (thermal-hydraulic, nuclear analysis)
- AREVA topical reports (COPERNIC fuel performance, fuel assembly design, MOX fuel design)
- Duke license amendment request and exemption requests
- Duke security plan changes and exemption requests
- DOE export license application
- Duke Cogema Stone & Webster transportation package certifications

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MOX MIXED OXIDE FUEL PROJECT

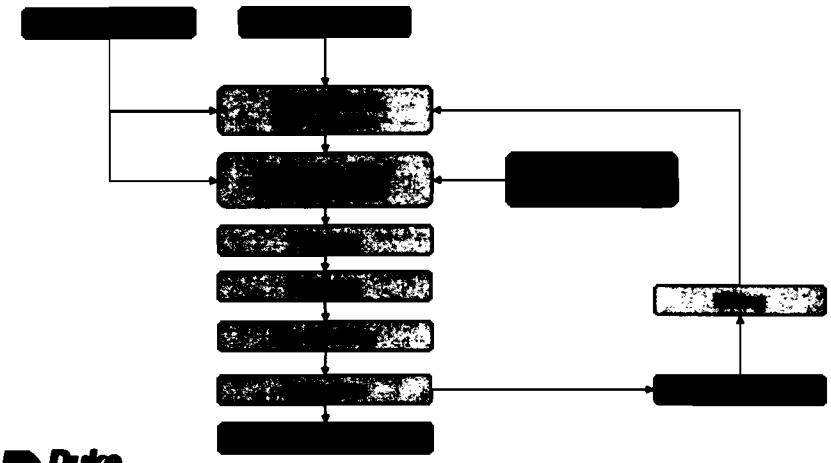
Outline of Presentation

- Introduction
- ⇒ MOX fuel - general
- Safety evaluation
- Environmental evaluation
- Summary

MOX Fuel Pellet Manufacturing

- Micronized Master Blend (MIMAS) process
- Decades of experience in Belgium and France
  - Plutonium from reprocessed reactor fuel
  - "Reactor grade" isotopics – more Pu-240 than weapons grade Pu
- Pellet structure
  - Uniform distribution of plutonium at a macroscopic scale
  - Heterogeneous microstructure at a micronic scale
    - Plutonium-rich particles (agglomerates)
    - Coating phase
    - UO_2 phase

MOX Fuel Pellet Manufacturing (MIMAS Process)



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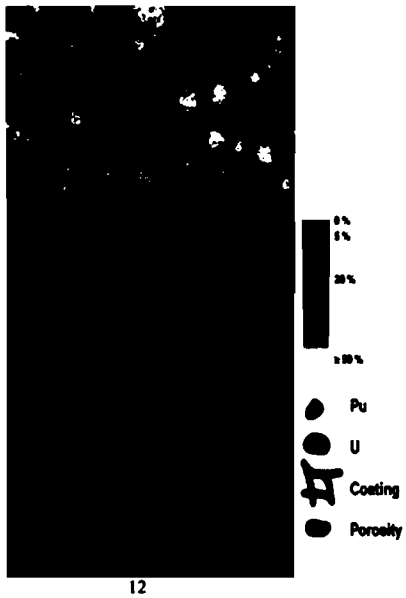
Unirradiated MIMAS MOX Fuel Microstructure

EPMA: quantitative analysis of Pu distribution

As-measured image

After image analysis



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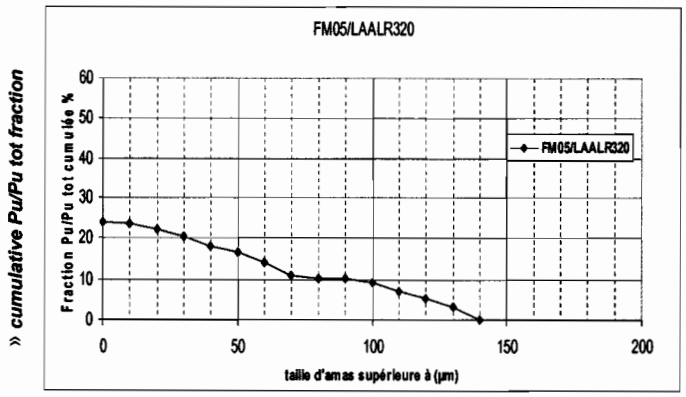
Cumulative Distribution of Plutonium

- *24 ± 6 % in PuO_2 agglomerates*
- *72 ± 6 % in coating phase*
- *4 ± 1 % in UO_2 phase*



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MOX
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Weapons Grade MOX Fuel Physical Characteristics

- **Sintered ceramic oxide fuel pellets**
- **Predominantly (>95%) uranium**
- **Material properties similar to LEU fuel**
- **Lower decay heat than LEU during time frame of interest for transient/accident analyses**
- **Small impact on global core physics parameters and core radionuclide inventories**
  - **Four out of 193 assemblies**
  - **(~2% of core)**

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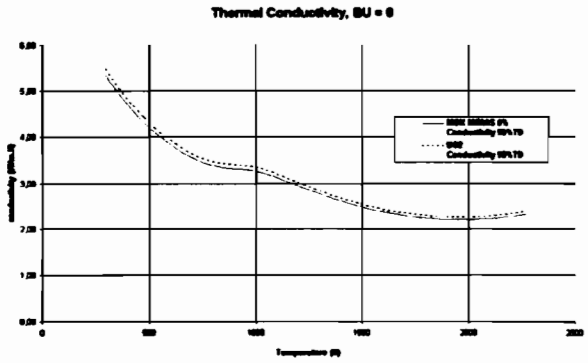
Thermal Conductivity



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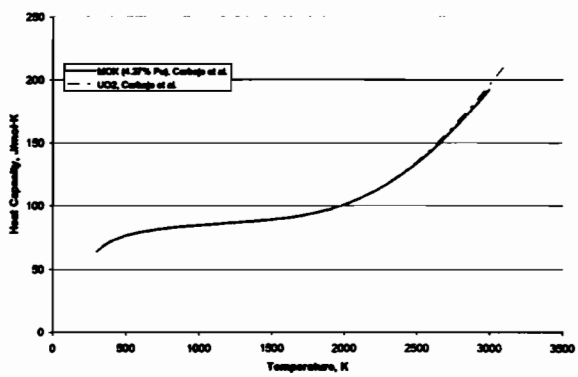
Heat Capacity



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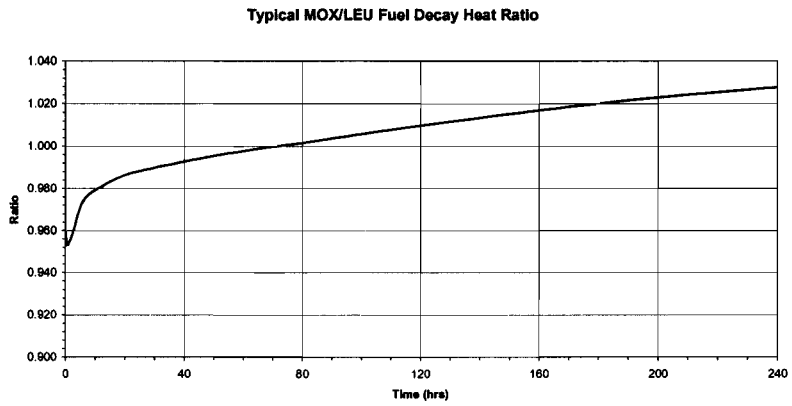
Decay Heat



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Global Core Physics Parameters

(% change due to four MOX fuel lead assemblies)

Parameter	BOC (4 EFPD)	EOC (495 EFPD)
Effective delayed neutron fraction	-2.1	-1.0
Prompt neutron lifetime	-1.8	-1.0
Equilibrium xenon worth	-1.1	-0.5
Hot full power mod temp coefficient	-3.0	-0.9
Hot full power Doppler coefficient	-0.7	0

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MOX Fuel Lead Assembly Design Description

- **Existing U.S. fuel assembly design with MOX fuel pellets: Advanced Mk-BW/MOX1**
  - **Advanced Mark-BW Fuel Assembly**
    - **The fuel assembly design is presented in BAW-10239, "Advanced Mark-BW Mechanical Design Topical Report"**
    - **Same assembly design (nozzles, grids, materials, etc. as has been successfully demonstrated in U. S. plants with uranium fuel pellets)**
  - **European MOX Technology, Experience, and Pellet Design**
    - **MOX effects are presented in BAW-10238, "MOX Fuel Design Report"**

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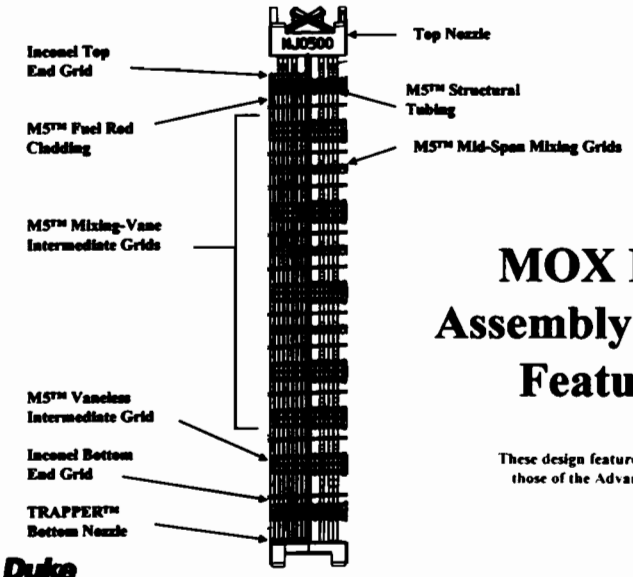
MOX MIXED OXIDE FUEL PROJECT



MOX Fuel Assembly Design Features

These design features are identical to those of the Advanced Mark-BW

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MOX MIXED OXIDE FUEL PROJECT

Fuel Rods

Fuel Rod Parameters	MOX Lead Assembly	Advanced Mark-BW
Clad Material	M5 Alloy	M5 Alloy
Fuel Rod Length, in	152.40	152.16
Cladding OD, in	0.374	0.374
Cladding Thickness, in	0.0225	0.0225
Cladding ID, in	0.329	0.329
Clad-to-Pellet Gap, in	0.0065	0.0065
Fuel Pellet OD, in	0.3225	0.3225
Design Burnup, MWd/Mthm	60,000 Lead Assy 50,000 batch	62,000 batch

MOX Fuel Experience Base

- Mature industrial-scale technology in Europe
- Substantial production capacity
  - MIMAS: French (Melox) and Belgian (Dessel) plants
  - SBR: BNFL (Sellafield) plant beginning production
- More than 3700 FAs delivered by Framatome ANP (France and Germany) as of the end of 2003
- More than 30 reactors in France, Germany, Belgium, and Switzerland are using MOX fuel

MOX Fuel Performance Test Programs

- About 100 commercial fuel rods examined in hot cells (burnup up to 63 GWd/tHM, 5 cycles)
- Power ramp testing and instrumented analytical irradiations have been or are being carried out up to high burnups (national & international programs)
  - Pellet-cladding interaction
  - Fission gas release
  - Temperature
  - In-pile densification

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MOX Fuel Performance Test Program Results

- Same behavior as LEU fuel in
  - Fuel rod growth
  - Cladding diametral deformation
  - Cladding waterside corrosion
  - Pellet solid swelling
  - ZrO2 internal layer
  - Fission product activity and release from failed fuel
- Somewhat higher fission gas release than LEU fuel at higher burnup
- Better pellet-cladding mechanical interaction than LEU
- Results summarized in IAEA TecDoc 415

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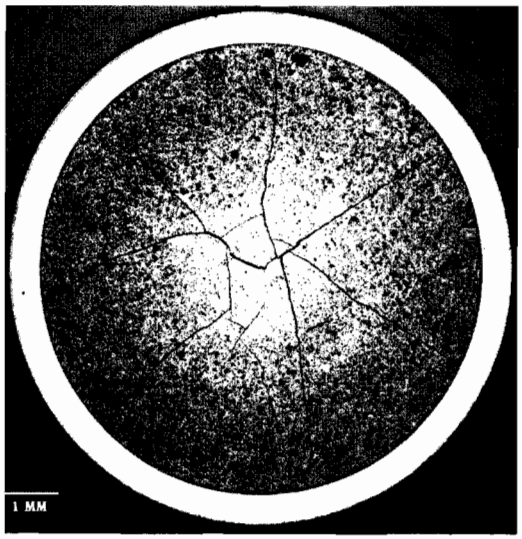
Radial Cut of a MOX Pellet (50 GWd/tM)



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MOX Fuel Fission Gas Release

- **Higher MOX fuel temperature at medium-high burnups**
  - **Neutronic properties: Higher linear heat rate at medium/high burnup**
  - **Physical properties: Slightly lower thermal conductivity**
- **Pellet microstructure:**
  - **Plutonium-rich particles from the MIMAS process**
  - **Local high burnup zones lead to the formation of dense pore populations**
- **Differences in fission gas release at medium-high burnup**

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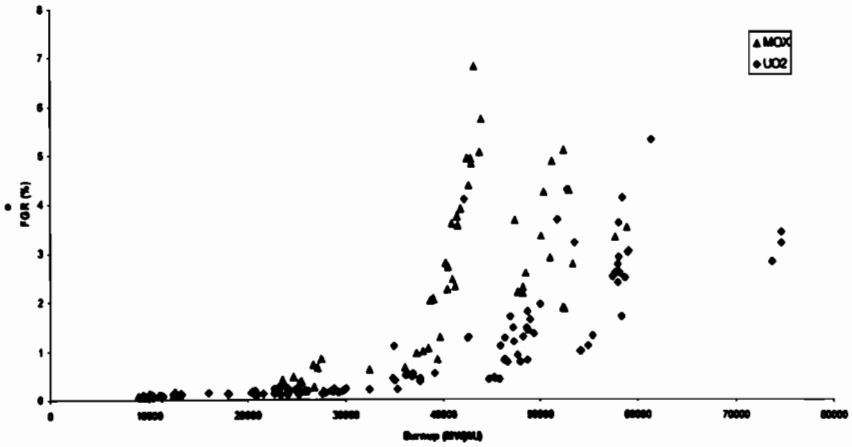
Fission Gas Release of PWR Fuel Rods



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MOX MIXED OXIDE FUEL PROJECT

Outline of Presentation

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- MOX fuel - general

⇒ Safety evaluation

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MOX MIXED OXIDE FUEL PROJECT

Basis for Safe Operation with MOX Fuel

- Similar physical characteristics between LEU and MOX fuel
- Extensive European experience base with MOX fuel
- Similar to prior U.S. MOX fuel lead assembly programs
- Proven fuel assembly design
- Analyses and evaluations of MOX fuel impacts at Catawba

LOCA Analyses

- Approach - Appendix K large break LOCA evaluation of MOX fuel lead assemblies
- Starting point – approved AREVA evaluation model based on RELAP5/MOD2-B&W
- Potential MOX effects were evaluated and incorporated in evaluation model as appropriate
- A MOX to LEU comparison calculation was performed
- Burnup and axial peaking studies were performed to establish LOCA limits for lead assemblies

Potential MOX Effects

Parameter	Effect	LOCA MODEL
1. Thermal Conductivity	Small	MOX used
2. Volumetric Heat Capacity	Essentially none	LEU used
3. Decay Heat	120% of 1971 ANS 5.1 standard plus actinides is conservative	120% of 1971 ANS 5.1 standard plus actinides used
4. Void Reactivity	More negative than LEU	LEU appropriate for core loading
5. Delayed Neutron Fraction	Less than LEU fuel (Conservative for LOCA)	LEU appropriate for core loading
6. Initial Fuel Temperature	Small	MOX (COPERNIC) used

Duke Power. A Duke Energy Company

MOX MIXED OXIDE FUEL PROJECT

Stylized MOX/LEU Comparison Analysis

Parameter (Time-In-Life)	LEU (0 GWd)	MOX (0 GWd)
Total Peaking (F_Q)	2.4	2.4
PCT (°F) Pin #1 (2.3 % Pu)	1981	2018

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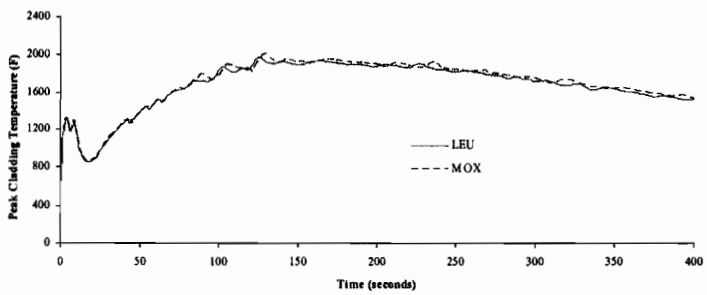
MOX MIXED OXIDE FUEL PROJECT

MOX/LEU Comparison



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MOX
MIXED OXIDE FUEL PROJECT

Other Criteria and Evaluations

- All 10 CFR 50.46 criteria are met for large break LOCA
  - Peak cladding temperature
  - Maximum cladding oxidation
  - Maximum hydrogen generation
  - Coolable geometry
  - Long-term cooling
- Small break LOCA
  - Not limiting for Catawba
  - MOX/LEU differences insignificant
- No adverse MOX impact on LEU fuel (no mixed core penalty)

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MOX
MIXED OXIDE FUEL PROJECT

LOCA Summary

- Specific evaluations were performed for MOX fuel lead assemblies using conservative Appendix K models appropriately adjusted for MOX fuel
- Analysis results were fundamentally similar to LEU fuel
- Sensitivity studies were performed on plant operating conditions
- Peaking criteria were established that ensure that MOX fuel remains within 10 CFR 50.46 acceptance criteria

Non-LOCA Evaluations

- Plant response to most non-LOCA design basis events is driven by global core physics parameters, system thermal-hydraulics, stored energy, and decay heat
  - Impact of four MOX fuel assemblies on global physics parameters is typical of cycle-to-cycle variations
  - System thermal-hydraulics are unaffected by MOX fuel
  - Four MOX fuel assemblies have no appreciable impact on stored energy
  - Decay heat is lower for MOX fuel during the time period of interest for transient analysis
- Some events require more detailed evaluation due to the potential for local effects

Non-LOCA Evaluations of Specific Events

- **Control rod withdrawal or drop**
  - Location with control rods limiting
  - MOX not loaded under control rods in first two cycles
- **Steam line break**
  - Same as control rod withdrawal/drop
- **Control rod ejection**
  - Representative analyses indicate much less than 100 cal/g in MOX fuel
- **Fuel assembly misloading**
  - Prevention measures equally effective for MOX fuel
  - Detection more effective (MOX fuel preferentially loaded in instrumented locations)

Non-LOCA Summary

- **The impact of four MOX fuel lead assemblies on most non-LOCA design basis events is clearly negligible**
  - Similar fuel design
  - Lower decay heat
  - Impact on global physics parameters in the noise of reload design
- **Events with potential local effects were evaluated in more detail**
  - Attributes of lead assembly program obviate most potential issues
  - Cycle-specific rod ejection analyses

Radiological Consequences

- SCALE analyses show that fission product inventories are similar between MOX fuel and LEU fuel
  - Worst case ^{131}I may be as much as 9% higher in a MOX fuel assembly compared to an equivalent LEU fuel assembly
  - Potential impact on thyroid and TEDE doses
- Accidents involving numerous fuel assemblies should see no significant impact
  - LOCA, rod ejection, and locked rotor assumed to fail 11%-100% of the fuel in the core
  - Lead assemblies are only 2% of the core
  - Postulated failures in non-MOX fuel assemblies dominate impacts

Radiological Consequences (cont.)

- Maximum impact seen in postulated accidents involving one or just a few assemblies
  - Fuel handling accident (FHA) (one assembly)
  - Weir gate drop (WGD) (seven fuel assemblies)
- Explicit FHA and WGD calculations performed using Alternate Source Term methodology
  - MOX fuel-specific radionuclide inventories
  - Sensitivity study - Reg Guide 1.183 gap fractions increased 50% to reflect higher fission gas release from MOX fuel
- Offsite and control room doses ~60% higher than all-LEU fuel case, but still well within regulatory limits

MOX/LEU Dose Comparison (Weir Gate Drop)

Receptor	TEDE Dose Limit (Rem)	LEU Fuel (Rem)	MOX Fuel – Nominal Release Fractions (Rem)	MOX Fuel – +50% Gas Release Fractions (Rem)
EAB	6.3	2.2	2.3	3.5
LPZ	6.3	0.31	0.33	0.50
Control Room	5.0	2.1	2.2	3.3

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MOX MIXED OXIDE FUEL PROJECT

Radiological Consequences Summary

- **Potential for dose impacts**
  - Different radionuclide inventories
  - Higher fission gas release from MOX fuel
  - Greatest impact for accidents involving a small number of assemblies
- **Explicit analyses of fuel handling and weir gate drop accidents**
  - Conservative treatment of MOX/LEU differences
  - Alternative Source Term methodology
  - Higher consequences in MOX fuel analyses, but well within regulatory limits

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MOX MIXED OXIDE FUEL PROJECT

Outline of Presentation

- Introduction
- MOX fuel - general
- Safety evaluation
- ⇒ Environmental evaluation
- Summary

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MOX MIXED OXIDE FUEL PROJECT

Environmental Evaluation

- Assessment of potential MOX fuel lead assembly impacts on the environment
- Normal operations
  - No impact on effluents
  - Slight increase in fuel handling occupational dose
- Accident situations addressed in safety analyses and radiological consequence analyses

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MOX MIXED OXIDE FUEL PROJECT

Severe Accidents with Four MOX Fuel Lead Assemblies

- Evaluation based on DOE analysis of the impact of 40% MOX fuel cores on severe accident consequences
  - 1999 Surplus Plutonium Disposition Environmental Impact Statement
  - MOX-specific radionuclide inventories
  - Results scaled from 40% MOX fuel cores to lead assembly cores (2% MOX fuel)
  - Change in consequences relative to all-LEU core range from -0.2% to +0.7%, depending on accident sequence
- 2000 Lyman analysis
  - Scaled results indicate maximum adverse impact of 1.6% (includes worst case actinide release fractions)

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MOX MIXED OXIDE FUEL PROJECT

Severe Accidents - Summary

- Severe accident behavior will be driven by LEU fuel
- Any impact from MOX fuel lead assemblies (2% of the core) would be negligible
  - Overall uncertainties in light water reactor severe accident behavior
  - Other nuclear power plant changes with similar impacts are implemented without explicitly addressing severe accident consequences
    - Power uprates
    - Changes in cycle length

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MOX MIXED OXIDE FUEL PROJECT

Outline of Presentation

- Introduction
- MOX fuel - general
- Safety evaluation
- Environmental evaluation

⇒ Summary

The Big Picture

- All nuclear power reactors are already using Pu fuel
  - About 850 kg plutonium in Catawba LEU core at end of cycle (compared to ~80 kg in four lead assemblies)
  - About 50% core power from plutonium fissions at end of cycle
- A similar MOX fuel lead assembly program was safely conducted at Ginna in the early 1980s
- European nuclear power reactors have demonstrated the safety of using MOX fuel
  - More than thirty reactors in four countries over 25 years
  - Up to 36% core fractions
- This program - 4 assemblies out of 193 (2.1% of core)

Intervener Issues

- Interveners are contesting the MOX fuel lead assembly license amendment request
  - Impact of MOX/LEU differences on LOCA and severe accidents
  - Failure to evaluate use of MOX fuel lead assemblies at Oconee
  - Security of fresh MOX fuel
- Intervener issues have been addressed in license amendment request and ASLB filings
- Hearings scheduled for June and September 2004
- Fundamental issue – how much alleged "uncertainty" is acceptable?

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MOX MIXED OXIDE FUEL PROJECT

Conclusion

- Duke license amendment request addressed potential MOX fuel lead assembly impacts on normal operations, the full range of design basis events, and severe accidents
- Regulatory limits are met
- No significant hazard to the public

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MOX MIXED OXIDE FUEL PROJECT

NRC STAFF REVIEW OF MIXED OXIDE LEAD TEST ASSEMBLIES AT CATAWBA NUCLEAR STATION

Robert E. Martin, Senior Project Manager
Undine Shoop, Reactor Systems Engineer
Ralph Landry, Senior Reactor Engineer
Anthony Attard, Senior Reactor Engineer
Steve La Vie, Health Physicist

Presentation for the
Advisory Committee on Reactor Safeguards
May 6, 2004

3

Presentation Message

- Licensee's Application of February 27, 2003, Followed by Numerous Supplements by Licensee.
- NRC Staff Safety Evaluation of April 5, 2004.
- NRC Staff Safety Evaluation found use of MOX LTAs acceptable on the basis of evaluations presented in that Safety Evaluation.
- Approval of application requires completion of other matters.
- Issue of Next Generation Fuel addressed by Licensee's letter of April 16, 2004, is under NRC staff review.

(4)

SRXB Review of the Mixed Oxide Fuel Lead Test Assemblies

Meeting with ACRS
May 6, 2004

Undine Shoop
U.S. Nuclear Regulatory Commission

Purpose

- Discuss Areas of Staff Review
  - Thermal Mechanical Design
  - Data Collection
  - Nuclear Design
  - Non-LOCA Transient Analysis
  - LOCA

Purpose of an LTA

- Gather data on fuel performance
- Based on production design
- Pre-characterized
- Examined between irradiation cycles and after discharge
- Basis for improved fuel designs and analytical models

Thermal Mechanical Design

Undine Shoop

Fuel Assembly Design

- Lead Test Assembly (LTA)
  - Licensing framework is SRP Section 4.2
- Design Evaluation is provided in BAW-10238

Framatome MOX vs LEU Fuel Assembly Design Differences

- Longer fuel rod
- European dish and chamfer designs
- 95% theoretical density
- Use of Mixed Oxide for fissile material

May 6, 2004 FL-3

Mixed Oxide Fuel

- Depleted Uranium matrix with weapons grade Plutonium fissile material
- Significance of Isotopic Mixture
  - Fewer absorber isotopes
  - Increased fissile isotopes
  - Lower enrichment requirement for comparable reactivity than reactor grade MOX

May 6, 2004 FL-4

Gallium

- Has the potential to migrate to the cladding and embrittle the cladding
  - Removed through polishing
  - ORNL tests on gallium migration
  - 300 ppb limit for plutonium feed material

Data Collection Program

Undine Shoop

Data Collection Program

- Purposes
  - Neutronic – Startup Physics Testing
  - Fuel Behavior – Post Irradiation Examinations (PIEs)

Neutronic

- 2 LTAs will be located in core locations that are directly measured by movable in-core detectors for the first and second irradiation cycles
- Operating Data from the cycle
  - Measurements taken monthly
  - Used to verify CASMO-4/SIMULATE-3MOX
- Start up Physics Test Plan

PIE

- Poolside PIE
  - Performed between cycles
- Poolside PIE
  - Performed after assembly discharge
- Hot Cell PIE

Nuclear Design

Undine Shoop

Neutronic Impact of LTAs

- 4 LTAs and 189 other assemblies
- Insignificant impact on core wide neutronic behavior

Core Design

- Checkerboard Pattern
- LTAs in symmetric core locations
- Unrodded locations
- LTAs are not limiting, but are in prototypical locations

Key Core Physics Parameters

Key Core Physics Parameters

EFPD	POWER (percent)	BORON (ppm)			MAX ASSY POWER 2RPF			2-D PEAK PIN POWER 2PIN		
		MOX	LEU	DELTA	MOX	LEU	DELTA	MOX	LEU	DELTA
0	0	1832	1815	17	1.407	1.334	0.073	1.557	1.498	0.059
4	100	1242	1235	7	1.291	1.284	0.007	1.426	1.423	0.003
12	100	1224	1218	6	1.272	1.277	-0.005	1.411	1.418	-0.007
25	100	1234	1230	4	1.272	1.275	-0.003	1.416	1.420	-0.004
50	100	1260	1258	2	1.270	1.270	0.000	1.421	1.421	0.000
100	100	1249	1250	-1	1.321	1.317	0.004	1.401	1.397	0.004
150	100	1170	1173	-3	1.345	1.340	0.005	1.414	1.409	0.005
200	100	1046	1051	-5	1.357	1.353	0.004	1.430	1.425	0.005
250	100	892	898	-6	1.373	1.365	0.008	1.437	1.431	0.006
300	100	720	728	-8	1.375	1.366	0.009	1.435	1.425	0.010
350	100	537	545	-8	1.361	1.354	0.007	1.420	1.413	0.007
400	100	350	359	-9	1.339	1.332	0.007	1.395	1.388	0.007
450	100	164	173	-9	1.313	1.307	0.006	1.368	1.362	0.006
470	100	91	100	-9	1.302	1.297	0.005	1.357	1.351	0.006
490	100	19	28	-9	1.293	1.289	0.004	1.347	1.342	0.005
495	100	1	10	-9	1.291	1.287	0.004	1.344	1.340	0.004

Key Core Physics Parameters

EFPD	POWER (percent)	BORON (ppmb)	ITC (pcm/°F)			MTC (pcm/°F)		
			MOX	LEU	DELTA	MOX	LEU	DELTA
0	100	1832	-8.48	-8.05	-0.43	-7.03	-6.60	-0.43
0	0	1832	-3.47	-3.10	-0.37	-1.76	-1.40	-0.36
4	100	1242	-13.84	-13.47	-0.37	-12.40	-12.04	-0.36
4	0	1242	-8.15	-7.85	-0.30	-6.46	-6.18	-0.28
200	100	1046	-18.34	-17.95	-0.39	-16.85	-16.47	-0.38
200	0	1046	-10.90	-10.60	-0.30	-9.18	-8.89	-0.29
495	100	1	-37.56	-37.25	-0.31	-35.92	-35.61	-0.31
495	0	1	-26.47	-26.25	-0.22	-24.66	-24.43	-0.23

EFPD	POWER (percent)	BORON (ppmb)	DOPPLER (pcm/°F)			DIFF BORON WORTH (pcm/ppm)		
			MOX	LEU	DELTA	MOX	LEU	DELTA
0	100	1832	-1.45	-1.45	0.00	-6.19	-6.30	0.11
0	0	1832	-1.71	-1.70	-0.01	-6.54	-6.68	0.14
4	100	1242	-1.44	-1.43	-0.01	-6.30	-6.40	0.10
4	0	1242	-1.69	-1.67	-0.02	-6.66	-6.78	0.12
200	100	1046	-1.49	-1.48	-0.01	-6.49	-6.56	0.07
200	0	1046	-1.72	-1.71	-0.01	-6.82	-6.89	0.07
495	100	1	-1.64	-1.64	0.00	-7.94	-8.01	0.07
495	0	1	-1.81	-1.82	0.01	-8.28	-8.35	0.07

Note: Boron concentrations in this table are for a representative core with MOX fuel lead assemblies. Table 3-7 has the corresponding boron concentrations for an all-LEU core.

Non – LOCA Transients

Undine Shoop

Non-LOCA Transients

- Deterministic Licensing application - addresses Chapter 15 transients
- Normal reload process used
- Confirm that all physics parameters fall within reference values previously calculated

Control Rod Ejection

- Core loading Pattern precludes significant impact of RIA
  - LTAs in unrodded locations
  - LTAs not close to fuel assemblies having significant ejected control rod worth
- Peak LEU enthalpy of 54 cal/g
- Peak MOX enthalpy of 30 cal/g

Fuel Assembly Misloading

- Administrative measures
- Core power distribution measurements

5

Catawba MOX LTA LOCA

Ralph R. Landry
Reactor Systems Branch, NRR
May 6, 2004

RRL-1

MOX LTA LOCA

- Analysis of Record – Resident Fuel + Sensitivity Studies
- MOX LTA LOCA Analyses

RRL-2

MOX LTA LOCA

- Analysis of record is Westinghouse WCOBRA/TRAC Realistic LBLOCA
- Resident fuel assemblies are Westinghouse Robust Fuel Assemblies (RFA)

RRL-3

MOX LTA LOCA

- Analysis of record covers Mark-BW fuel by sensitivity study use of a surrogate, or proxy, assembly with pressure drop representative of Mark-BW assembly
- Mark-BW/MOX1 assembly pressure drop is closer to Westinghouse RFA than to Mark-BW fuel

RRL-4

MOX LTA LOCA

- MOX LTA LOCA response calculated using Framatome ANP Appendix K code, RELAP5/MOD2-B&W
- Approved code includes approved properties of M5 cladding

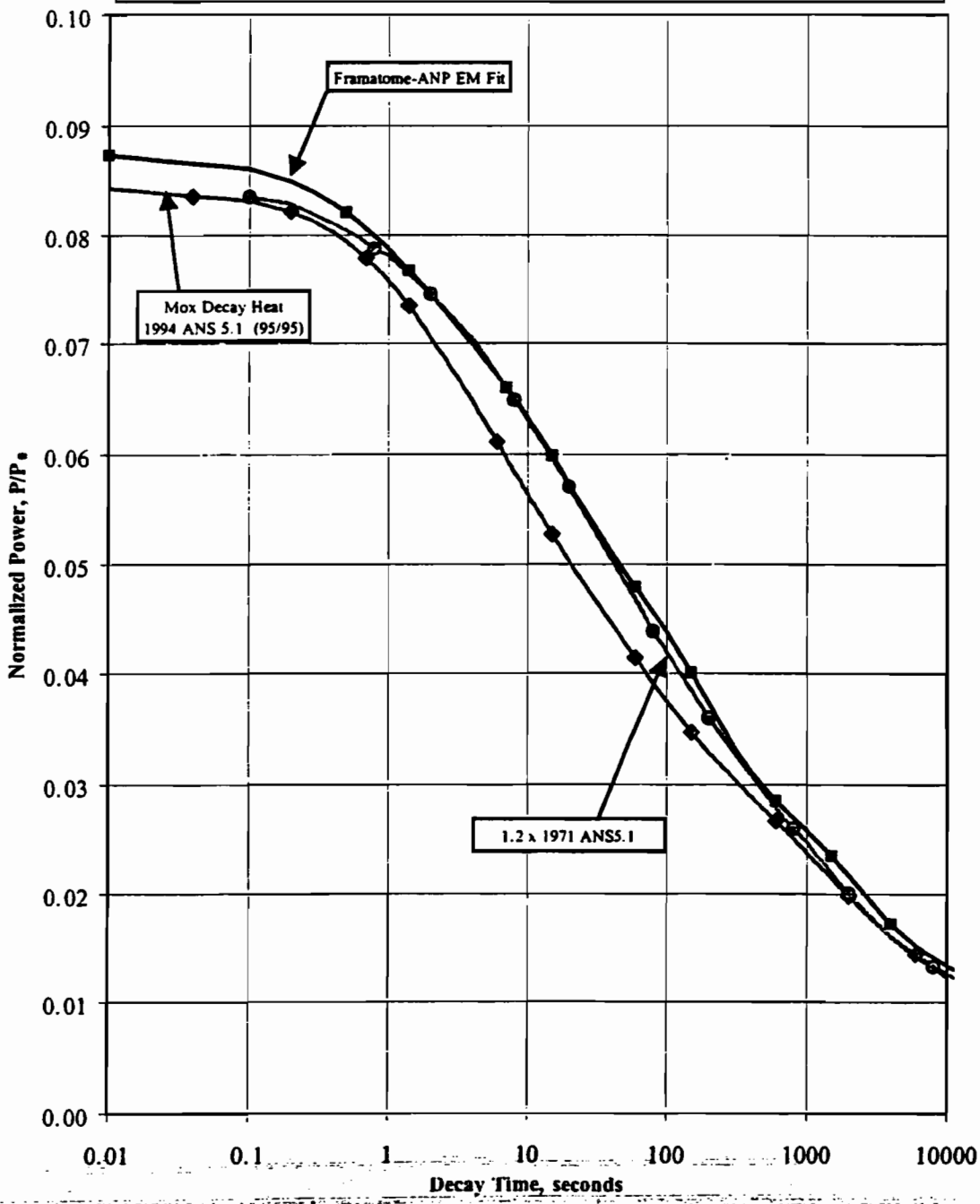
RRL-5

MOX LTA LOCA

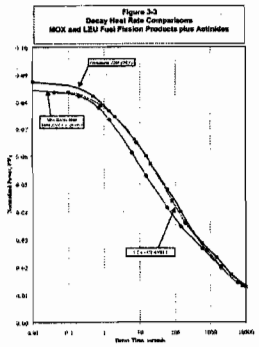
- Decay heat model used, ANSI/ANS-5.1-1994, is applicable since highly burned LEU fuel produces the majority of its energy from the fission of plutonium. Multiplier of 1.2 is used to cover uncertainties (Figure 3-3)

RRL-6

Figure 3-3
Decay Heat Rate Comparisons
MOX and LEU Fuel Fission Products plus Actinides



MOX LTA LOCA



RRL-7

MOX LTA LOCA

- ΔPCT for MOX LTA vs. RFA LEU is -38°F, or 2018°F for the MOX LTA vs. 2056°F for the RFA LEU
- MLO for MOX LTA is 4.5% vs. RFA 10%
- MOX LTA placement is in non-limiting locations

RRL-8

MOX LTA LOCA

	RFA LEU WCOBRA/TRAC	Mk-BW/MOX1 MOX RELAP5/M2-B&W	Mk-BW/MOX1 LEU RELAP5/M2-B&W
PCT	2056°F	2018°F	1981°F
MLO	10%	4.5%	4.0%

RRL-9

MOX LTA LOCA

- Staff concludes that MOX LTAs will comply with requirements of 10 CFR 50.46 when inserted in core of Westinghouse RFA LEU fuel

RRL-10

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

USE OF MIXED-OXIDE LEAD TEST ASSEMBLIES AT CATAWBA

Presentation to NRC Advisory Committee on Reactor Safeguards

Edwin Lyman
Union of Concerned Scientists
May 6, 2004

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

BREDL INTERVENTION ON MOX LTA REQUEST

- UCS is assisting the Blue Ridge Environmental Defense League (BREDL) in its challenge of Duke's MOX LTA LAR and security exemption request
- Security-related contentions
  - conducted in closed (safeguards) proceeding
- Non-security-related contentions (safety and environmental)

Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

MOX LTA HEARING

- ASLB admitted three (reframed) non-security-related contentions on 3/5/04
- Duke wants NRC to issue the LTA license amendment and security exemptions by early August 2004
  - Timetable is driven by DOE/NNSA's desire for a decision prior to shipment of plutonium to France for LTA fabrication at Cadarache (before plant shuts down)
  - ASLB attempt to accommodate this request is resulting in a highly compressed adjudicatory proceeding

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Union of Concerned Scientists

Citizens and Scientists for Environmental Solutions

MOX LTAs: THE BIG PICTURE

- Issues that are resolved only by virtue of the small number of MOX LTAs in the core will need to be reconsidered when the batch loading application is received next year
- US approval process for MOX LTA LA and security exemptions will set an example for Russian counterpart
- Thorough review should take place now

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M5 CLADDING ISSUES

- Vulnerability of zirconium-niobium alloy claddings (E110, M5) to embrittlement appears to be a function of initial surface treatment (polishing vs. etching)
  - Argonne oxidation test on etched M5 samples "showed a potential similarity to the oxide characteristics of alloy E-110" — letter from James F. Mallay, Framatome ANP, to Ralph Meyer, RES, May 5, 2003
  - " ..parallel testing at Argonne on unirradiated ZIRLO and M5 tubing has shown significant differences compared with Zircaloy." — letter from A. Thadani, NRC, to D. Modeen, EPRI, April 21, 2004
  - Raises questions regarding stability of M5 with respect to production conditions, changes under irradiation, corrosion, hydrogen uptake (see Updated Program Plan for High-Burnup LWR Fuel, August 2003)

Union of Concerned Scientists

CONTENTION II

- Reframed (Non-security) Contention II: Duke has failed to adequately account for differences in MOX and LEU fuel behavior with respect to radionuclide releases during "core disruptive accidents"
- Issues (see Expert Panel Report on High-Burnup and MOX Source Terms, ERI/NRC 02-0202, Nov. 2002):
  - Different degradation behavior of MOX
  - Enhanced release rates of some radionuclides from MOX
  - Current source term underestimates release fractions of tellurium and ruthenium isotopes (inventories greater in MOX)
- Fundamental problem: Uncertainties due to gaps in experimental database for MOX under core melt conditions
  - IRSN proposal for Phébus MOX source term test

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CONCLUSIONS

- Much research is needed to reduce the uncertainties in M5 cladding and MOX fuel performance during LOCAs and severe accidents
  - ANL LOCA tests with irradiated M5-clad fuel LEU
  - Halden fuel relocation test (LEU)
  - Proposed Phébus MOX LOCA and source term tests
- More uncertainty introduced by Duke's plan to load another type of experimental fuel simultaneously with the MOX LTAs
- BREDL/UCS maintains that experimental data is insufficient to support approval of Duke's MOX LTA LAR at this time

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CONCLUSIONS (cont.)

- Duke has not demonstrated adequately that the introduction of 4 MOX LTAs will have only an insignificant impact on risk
  - Duke should do its own risk calculation, rather than rely on a DOE estimate
  - Duke should examine impact of source term uncertainties on result
  - Duke's comparison of the increase in risk to that associated with other license amendments such as power uprates is not valid, because the "benefits" are different in each case
- Contrary to Duke's assertion, BREDL is not seeking "absolute certainty" but only "reasonable assurance" that the MOX LTA program will provide adequate protection of public health and safety

Risk Management Technical Specifications

Presentation to the
Advisory Committee on Reactor Safeguards
May 6, 2004

7

Presentation Participants

- NRC Staff
  - Tom Boyce, NRR TS Section
  - Bob Tjader, NRR TS Section
  - Mark Reinhart, NRR PRA Section
  - Nick Saltos, NRR PRA Section
- Industry Representatives
  - Biff Bradley, NEI
  - Wayne Harrison, STP
  - Bill Stillwell, STP

Opening & Closing Comments

- RMTS Initiative 4b is dependent upon PRA Quality
- Communication and Training of HQ Staff & Regions are essential; Initiative 4b is participating in the Risk-Informed Environment Initiative
- Early in the Initiative 4b Process; Learn as we go forward

FEEDBACK FROM SUBCOMMITTEES

- Comments:
  - Good idea to Risk-inform TS
  - Structure of Initiative 4b is good
- Issues:
  - Configuration Risk Monitors and Assessment Tools
    - Extent of PRA Incorporation
    - QA/QC of software & updates
  - Uncertainty and impact on CTs/AOTs
  - Licensee incentive to fix problems within CT
  - Review Risk associated with Front Stops
  - Time to calculate risk
  - Oversight of changes to PRA after Initiative 4b issued

Principles for RMTS Development

- Achieve coherence with other risk-informed regulation development (MRule, PRA Quality, 50.69)
- Credit for 50.65(a)(4) programs in RMTS Initiatives
- Licensee's risk submittals must meet standards for quality and comprehensiveness
- Involve NRC staff with cognizance for operation, training, inspection, maintenance, regions/STA, and risk assessment

STATUS OF INITIATIVES

- Reliance on existing (a)(4) Program
  - Initiative 2: Missed Surveillances (NRC Approved)
  - Initiative 3: Mode Change Flexibility (NRC Approved)
- Analysis of Specific Plant Configurations
  - Initiative 1: Modified End States (1 yr)
  - Initiative 6: LCO 3.0.3 Action Times (1 yr)
  - Initiative 7: Non-TS Support System Operability (1 yr)
- Quantitative Risk Assessment / Quality PRA
  - Initiative 4: Flexible Completion Times (2 yrs)
  - Initiative 5: Surveillance Frequency Program (2 yrs)
- Rulemaking
  - Initiative 8: Relocate non-risk significant systems from TS (3+yrs)

Initiative 4 – Risk-Informed Completion Times

- Effect: Extend completion time from a nominal value up to a predetermined "backstop" maximum using configuration risk management
- Basis: Under development, to include approved process, requirements for PRA technical adequacy, real-time quantitative capability, configuration and cumulative risk metrics
- Status: Industry submitted draft guidance document & pilot proposals; staff provided feedback. STP & Fort Calhoun are pilots. Hope Creek has formally volunteered to be pilot.

Initiative 4b Example

- See proposed 4b Tech Spec; discuss concepts
- Initiative 4b concepts
  - Front Stop; current CT
  - CRMP-based CT
  - Back Stop
  - Use of Real-Time Risk Assessment Tools and Decision Making Process

TABLE 3-1
GENERIC RISK-INFORMED CTs WITH A BACK-STOP: EXAMPLE FORMAT

Actions Condition	Required Action	Completion Time
B. One [HPSI] subsystem inoperable.	B.1 Restore SI subsystem to OPERABLE status.	72 hours
	OR	
	B.2.1 Determine that the completion time extension beyond 72 hours is acceptable in accordance with established RMTS thresholds.	72 hours
	AND	
	B.2.2 Verify completion time extension beyond 72 hours remains acceptable.	In accordance with the RMTS Program (i.e., within 24 hours of a subsequent configuration change)
	AND	
	B.2.3 Restore subsystems SI to OPERABLE status.	30 days or acceptable completion time , whichever is less.

POTENTIAL IMPLEMENTATION STRUCTURE

- Program Requirements in Technical Specifications Administrative Controls
  - PRA Quality (RG 1.200)
  - Guidance Documents (RG 1.177+, RMG)
- Licensee Program Guidance
- Oversight

RMTS INITIATIVE 4b and PRA QUALITY

- Use of "Real-time" PRA results to determine Completion Times is a significant change to Technical Specifications
  - Licensee's use of PRA
  - NRC Review & Oversight
- PRA model must be of High Quality (scope, elements, and technical attributes)
- Configuration Risk Management process and tool must be of High Quality

Pilots for PRA Quality and Initiative 4b

- PRA Quality (RG 1.200) pilot program in parallel with RMTS Initiative 4b pilot program
- RG 1.200 Pilot Plants: SONGS, CGS, STP, Limerick
- 4 of 5 Pilot Applications of RG-1.200 involve Technical Specification Amendments
- RMTS Initiative 4b Pilot Plants: STP, FCS, Hope Creek
- STP is a Pilot for both RG 1.200 and RMTS Initiative 4b
- Pilots to test:
  - Reg Guide (RG-1.200) ability to prove adequate PRA Quality
  - Necessary scope of PRA
    - Internal Fire + External Events + Shutdown & Transition Risk
  - Software for Configuration Risk Management Tool
  - Configuration Risk Management Process

CURRENT REVIEW ISSUES

- Exportability; Pilot Plant General Acceptance Criteria
  - Reliability
  - Repeatability
  - Enforceability/Oversight
- PRA Quality (proof of concept)
  - Scope
  - Level of Detail
  - Acceptability

Opening & Closing Comments

- RMTS Initiative 4b is dependent upon PRA Quality
- Communication and Training of HQ Staff & Regions are essential; Initiative 4b is participating in the Risk-Informed Environment Initiative
- Early in the Initiative 4b Process; Learn as we go forward

Risk Management Technical Specifications Initiative 4B

Biff Bradley

May 6, 2004

Foundation

- Maintenance rule (a)(4) provision implemented November 1999
  - Resulted in both deterministic (TS) and risk management (MR) regulatory requirements for plant configuration control – sometimes in conflict
  - MR risk assessment and management guidance developed with recognition that TS provided "backstop"
  - NRC recognized that MR could provide foundation for additional TS reform

NEI

Objectives

- Better align deterministic tech specs with risk management approach required by maintenance rule
- Make changes within existing tech spec framework and practice
- Maintain operator safety focus and ease of use
- Provide incentive for improved PRAs and configuration risk assessment tools

NEI

Initiative 4B approach

- Would apply to all equipment LCOs
  - Not applicable to parameters, safety limits
- Maintains existing LCO as "front stop"
  - Operator familiarity
  - Approaching front stop would trigger more extensive risk evaluation and actions
- Deterministic backstop would be established
  - 30 days irrespective of risk impact
- Actual completion times would be based on risk assessment and management using NRC approved risk management guidance

NEI

Pilot Plants

- South Texas Project (whole plant)
- Hope Creek (whole plant)
- Ft. Calhoun (system specific)
- Additional plants interested
  - All would incorporate EPRI risk management guidance method through reference in Tech Specs

Risk management guidance for 4B

- Developed by EPRI
  - Builds on existing (a)(4) guidance
  - More rigor in risk analysis, risk management actions, plant shutdown decisions
  - PRA scope and capability requirements
- One round of NRC review/feedback complete
  - 75 NRC questions posed and addressed
  - Iterative process to complete development
  - Will be finalized through pilot plant process

NEI

PRA requirements – proposed for 4B

- Minimum PRA and tool requirements
  - Internal events and LERF, NRC Reg Guide 1.200 (ASME standard)
  - External events at power (including seismic, internal fires)
  - Ability to quantify configuration risk
  - Ability to determine and track aggregate risk
  - Updating requirements

NEI

Risk assessment metrics

- Establish for:
  - Planned evolutions
  - Emergent conditions
- Guidance will address use of:
  - Temporary risk increase (ICDP)
  - Risk "speed limit" (CDF limit)
  - Cumulative risk (Δ CDF)

Risk management

- Actions based on risk metric results
- Examples
  - Existing tech spec actions
  - Planning and sequencing of activities
  - Training, prestaging of maintenance
  - Limit duration of maintenance
  - Provide for rapid recovery of equipment
  - Prohibit maintenance on opposite train
  - Shut down plant (emergent condition)

Conclusions

- Challenging risk application
- Risk management guidance is work in progress
- Pilot applications will enable further development and detail in guidance
- Goal is NRC endorsement at appropriate level of detail

NEI

STP Nuclear Operating Company

STP Risk-Informed Technical Specifications Application

Advisory Committee on Reactor Safeguards

May 6, 2004

9

Introduction

- STP Participants
  - Bill Stillwell — Risk Management Supervisor
  - Wayne Harrison — Licensing

Agenda

- Scope and content of the STP application
- STP PRA Quality (RG 1.200 Pilot)
- Implementation

Scope and Content

- Industry pilot for risk-informed Technical Specifications using configuration risk management
- Applies STP's Maintenance Rule (a)(4) approach to determine configuration based allowed outage times.
  - References the EPRI Implementation Guidelines
- Pilot application for PRA Quality RG 1.200

Scope and Content

- Current Technical Specification structure and format retained
- Allows operators to use risk management option to determine allowed outage time when the existing allowed outage time or "frontstop" time is exceeded
- Imposes a "backstop" time to return inoperable equipment to service

Scope and Content

- Selected instrumentation of TS 3.3
- Code safety valves
- Pressurizer PORVs
- Accumulators
- ECCS
- RHR
- RWST
- RCB Purge
- Containment Isolation Valves
- Containment Spray
- Containment Fan Coolers
- AFW
- MSIVs
- MFIVs
- Atmospheric Steam Relief
- Component Cooling Water
- Essential Cooling Water
- Essential Chilled Water
- SDGs and Off-site circuits
- Batteries
- ESF Buses

Draft TS 3.13.1

RISK MANAGEMENT

ALLOWED OUTAGE TIME DETERMINATIONS

LIMITING CONDITION FOR OPERATION

3.13.1 When referred to this specification, equipment that has been removed from service or declared inoperable shall be evaluated for its impact on plant risk and allowed outage times determined accordingly.

APPLICABILITY: As required by the referencing specification

ACTION:

Determine that the configuration is acceptable for Completion Time extension beyond the [Front Stop AOT],

AND

Determine that the configuration is acceptable for continued operation beyond the [Front Stop AOT] whenever configuration changes occur that may affect plant risk,

AND

Restore required inoperable [subsystem, component] to OPERABLE status within the Acceptable Allowed Outage Time Extension or 30 days, whichever is shorter.

OR

Take the ACTION required in the referencing specification for required action or completion time not met

SURVEILLANCE REQUIREMENTS

4.13.1: As required by the referencing specification

Sample Specification

PLANT SYSTEMS

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least three independent essential cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With only two essential cooling water loops OPERABLE, **within 7 days** restore at least three loops to OPERABLE status **or apply the requirements of Specification 3.13.1, OR** be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. **With two or more essential cooling water loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of Specification 3.13, OR be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.**

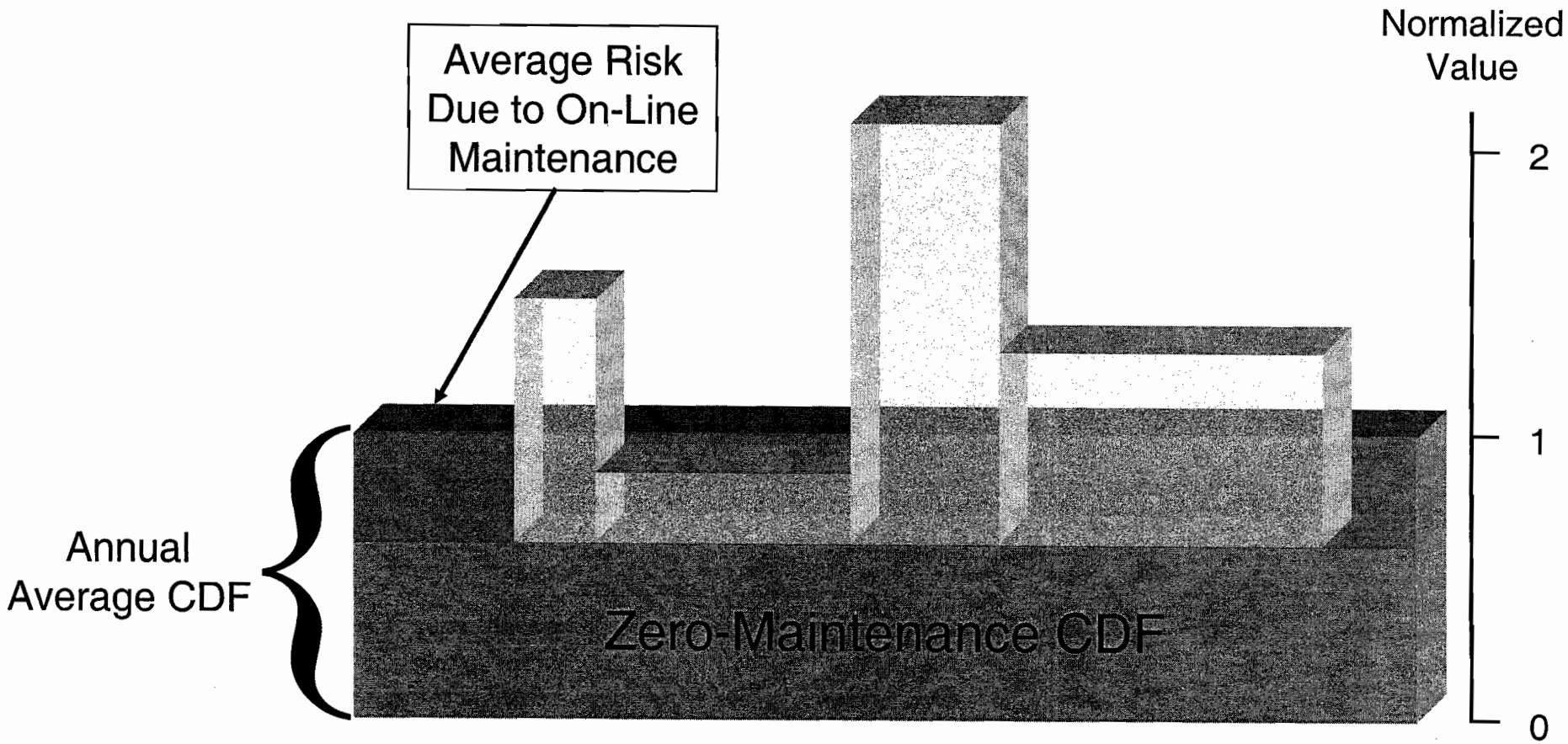
STP PRA Quality

- PRA quality issues to be addressed as part of the RG 1.200 pilot
- PRA quality scope to include industry peer review, ASME Standard (ASME RA-S-2002), and RG 1.200
- PRA quality needed for the application itself will also be evaluated.

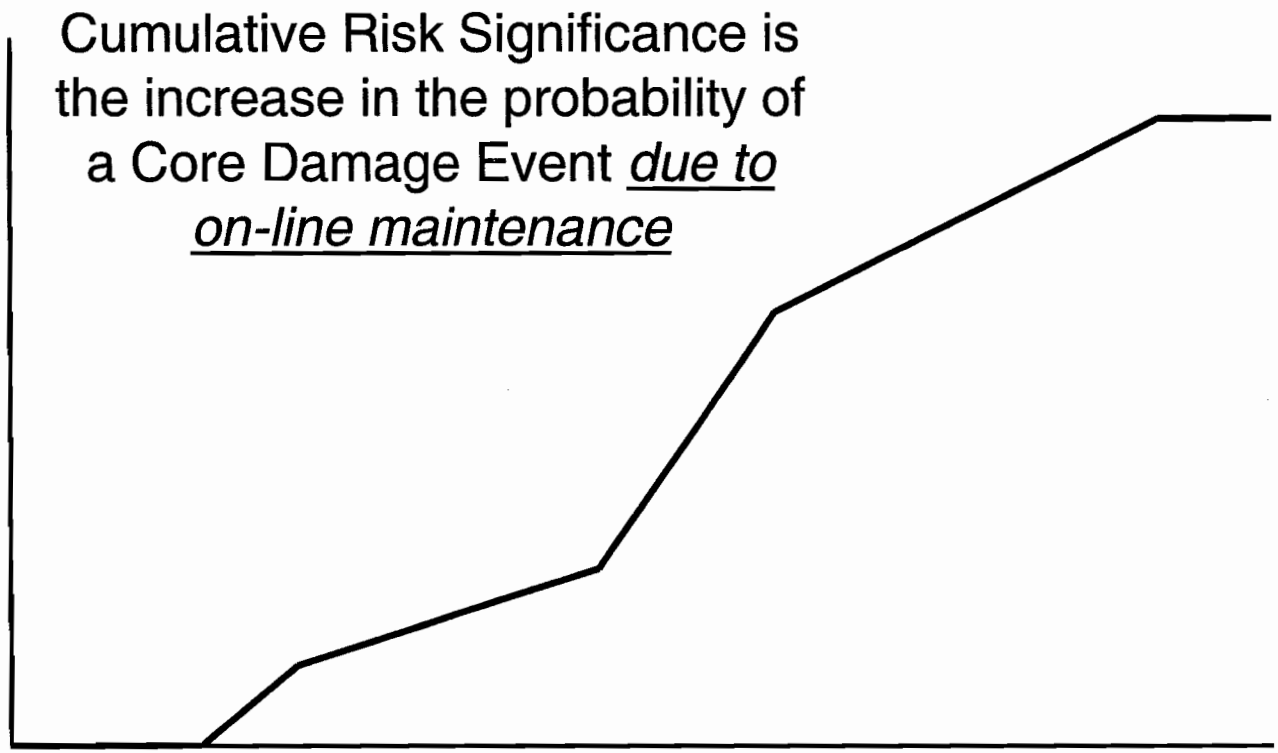
Implementation

- Applies the STPNOC Configuration Risk Management Program (CRMP)
  - Same program used for 10CFR50.65(a)(4)
  - Non-risk significant threshold (1E-06)
  - Potentially risk-significant threshold (1E-05)
- STP has extensive experience applying the CRMP
  - Routinely used to manage weekly work
  - Effectively applied to manage recent extended diesel generator outage.

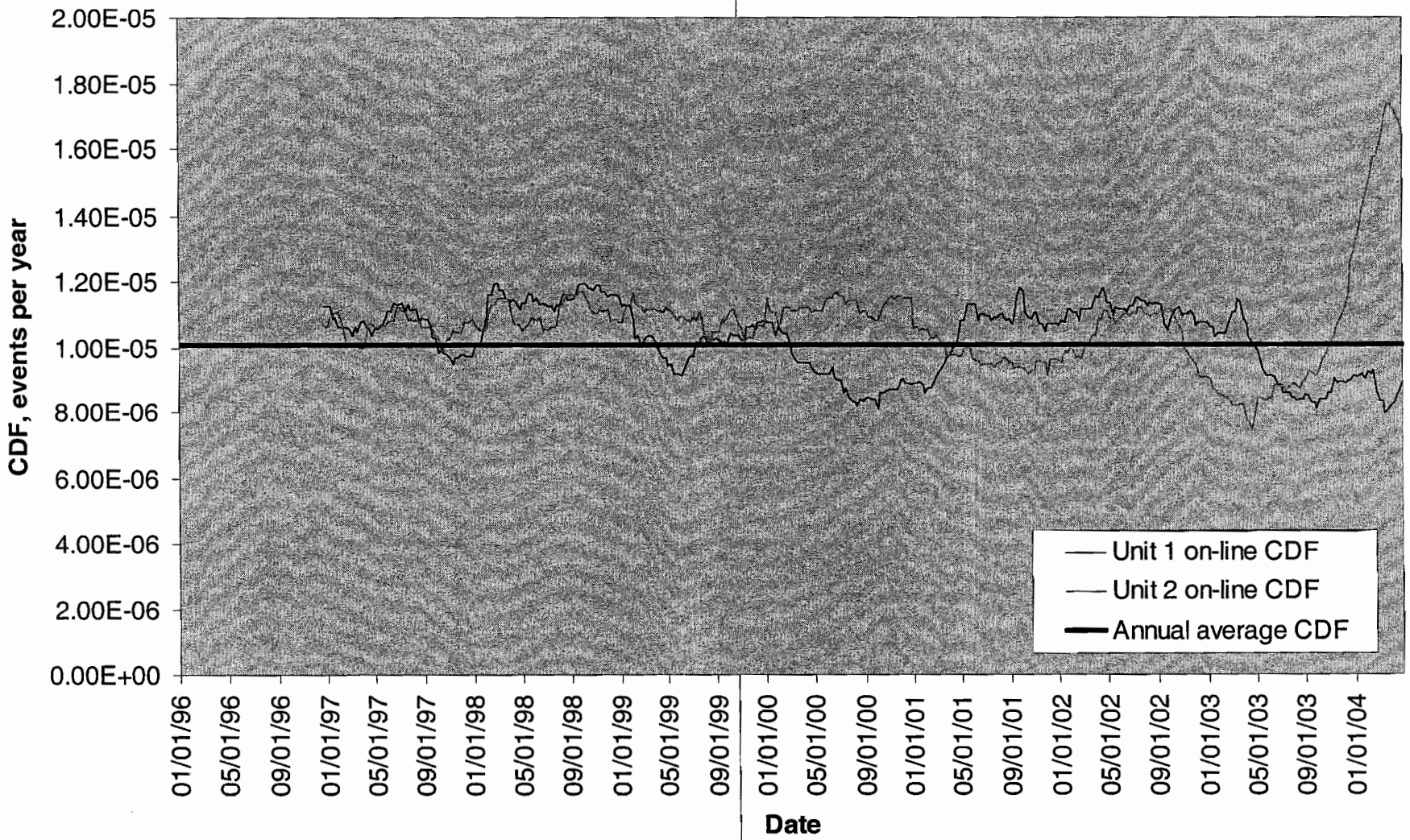
How Risk Values Stack Up



How Risk Values Add Up



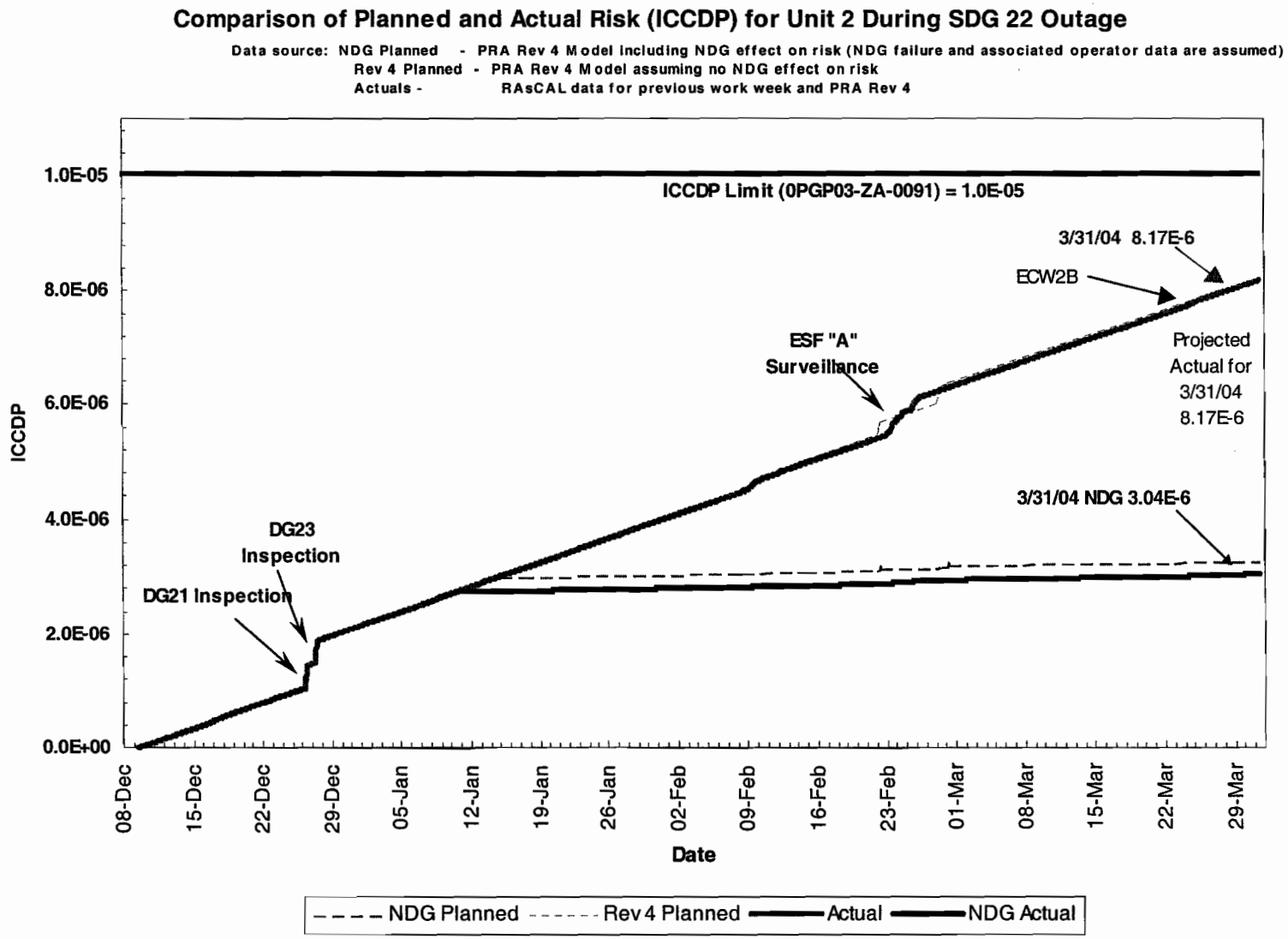
STP Actual On-Line CDF



SDG 22 113 Day Extended AOT

- AOT extension met RG 1.174 and RG 1.182 acceptance criteria
- Installed non-safety DGs (NDG) as compensatory action
  - RG criteria met without credit for NDGs
- STPNOC closely monitored the risk profile
- SDG 22 successfully returned to service

SDG 22 113 Day Extended AOT



RG 1.200 (and SRP 19.1) "An Approach for Determining the Technical Adequacy of PRA Results for Risk-Informed Activities"

Trial Implementation Phase

ACRS Informational Briefing

May 6, 2004

PURPOSE OF BRIEFING

Inform ACRS of Current Activities and Plans Related to Trial Implementation of RG 1.200 and Associated SRP 19.1

AGENDA

- ❒ Background
- ❒ Objectives of RG 1.200
- ❒ Purpose of Pilots
- ❒ Scope of Pilot Applications & Staff Review
- ❒ Schedules
- ❒ Conclusions

BACKGROUND

- ❒ASME Published ASME RA-S-2002 "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"
- ❒Most Utility PRAs Peer Reviewed Following NEI 00-02, "PRA Peer Review Process Guidance"
- ❒NEI Provided "Self-Assessment Process" to Address Differences Between ASME Standard and NEI 00-02
- ❒DG-1122 Published for Public Review and Comment
- ❒SONGS Peer Reviewed Using ASME Standard
- ❒Consensus that Staff Should Publish the RG 1.200 "For Trial Use" and Test the Guide via Pilot Applications

OBJECTIVES OF RG 1.200

- ❒ Provide Staff with Confidence that Base PRA is Adequate for the Decisionmaking Required by the Application
- ❒ Endorse Consensus Standards (e.g., ASME) as Basis for PRA Technical Adequacy
- ❒ Improve Focus and Consistency of Staff Reviews
- ❒ Reduce the Depth of the Staff Review
- ❒ Increase Public Confidence in the Adequacy of the Base PRA & the Staff Reviews

CURRENT STAFF REVIEWS

- ❒ Subjective in Scope and Level of Detail
  - Relies on Knowledge/Experience of the Staff
- ❒ Staff Relies on Previous Reviews
  - IPE/IPEEE and Associated RES Evaluations
  - Peer Review Findings
  - Licensee PRA Quality Programs
  - Prior Risk-Informed Application Reviews
- ❒ Little Guidance on What to Submit to Address PRA Technical Adequacy

PURPOSE OF PILOTS

- ❒Provide Assistance in Clarifying Aspects of RG 1.200 & SRP 19.1; for example,
  - Interpretation of Documentation Needs
  - Interpretation of Requirements
  - Interpretation on Staff Positions
- ❒Assess Licensees' Self-Assessment Approaches, Findings, & Resolution to Ensure Base PRA Properly Evaluated
- ❒Provide Guidance on Scope and Level of Detail of Licensee Application-Specific Submittals & Staff Review
- ❒Identify Specific Improvements to RG 1.200, SRP 19.1, ASME Standard, & NEI Self-Assessment Guidance
- ❒Gain Insights in Resource Levels Needed for Quality Submittals & Staff Review
- ❒Insights that Could Help Development of Other Standards

SCOPE OF PILOTS

- ❒5 Applications Identified as Pilots
  - **Columbia TS - EDG Allowed Completion Time Extension**
    - Allows Extension of Allowed Completion Time to 14 Days if Identified Risk Management Actions are Established
  - **Limerick TS - 5B Initiative**
    - Surveillance Test Intervals Placed in Licensee-Controlled Document
    - Surveillance Requirements Retained in TS
    - Surveillance Test Intervals Based on Risk-Informed Process
  - **SONGS TS - Battery Replacement/DC System Reconfiguration**
    - Allow On-line Cross-tie of DC Subsystems within a Train for up to 30 Days for Maintenance or Replacement of Batteries
  - **Surry 10 CFR 50.69 Application**
  - **STP TS - 4B Initiative**

STAFF PILOT REVIEW SCOPE

- Pilots Involve Actual Plant-Specific Risk-Informed License Applications
  - Requires Finding of PRA Technical Adequacy to Support the Staff Development of the Safety Evaluation for the Application
- Pilots will Address the Full-power, Internal Events (Excluding Internal Fires) Level 1 PRA & LERF
  - Other Aspects, such as External Events, Internal Fires, & Shutdown Operations, will be Reviewed by the Staff Consistent with Current Practices
  - When Future Standards are Developed & Endorsed (in RG 1.200) for External Events, Internal Fires, & Shutdown Operations, these Standards may also be Piloted
- Pilots will Involve More Detailed Reviews than Typical Applications to Properly "Exercise" Various Aspects of the Guidance to Gain Insights

SCHEDULING CONSIDERATIONS

- Regular Meetings
  - Held General & Individual Applicant Public Meetings
  - Will Hold Public Meetings Throughout Trial Implementation Phase
- Pilots will Involve Multiple Licensees, Multiple Types of Applications, and Multiple Staff Reviewers
  - Efficiencies Needed to Ensure Other Licensee and Staff Activities are Not Adversely Affected During Pilot Application Phase
- As Much As Possible, the Trial Application Reviews will Overlap
  - Efficiencies Gained in Staff Resources, Lessons Learned, and in Regular Scheduled Public Meetings to Status Activities

NEAR-TERM PILOT SCHEDULE

❒ Mid-May	Trial application submitted for Columbia
❒ End of May	Trial applications submitted for San Onofre & Limerick
❒ Week of June 7	Columbia site visit/audit
❒ End of June	Trial application submitted for Surry
❒ End of June	Status meeting on submittal & site visit observations
❒ Week of July 12	Limerick site visit/audit
❒ Week of Aug. 9	San Onofre site visit/audit
❒ End of August	Trial application submitted for South Texas
❒ End of August	Status meeting on submittal & site visit observations
❒ **End of August**	**RG 1.200 Appendix C Issued for Public Comment**

RG 1.200 SCHEDULE

❒ ACRS Subcommittee	November 2004 ?
❒ Update RG 1.200	December 2004
❒ Public Meetings	December 2004
❒ Issue for Public Comment	January 2005
❒ Public Meeting	February 2005
❒ CRGR/ACRS Briefing	March 2005
❒ Issue RG 1.200 Rev. 0	April 2005

CONCLUSIONS

- The Staff and Industry are Embarking on the Trial Implementation Phase of RG 1.200 and SRP 19.1
- This Trial Phase Involves Actual Licensee Risk-Informed Applications
- Lessons Learned During the Trial Phase will be Fed Back into Revising RG 1.200 and SRP 19.1
- Provides Insights for Phasing in Implementation of Future PRA Technical Adequacy Standards (e.g., External Events, Internal Fires, & Shutdown)

GOOD PRACTICES FOR IMPLEMENTING HUMAN RELIABILITY ANALYSIS

Presentation to ACRS – Full Committee

May 6, 2004

Erasmia Lois, NRC

John Forester, SNL

Alan Kolaczkowski, SAIC

Sandia National Laboratories

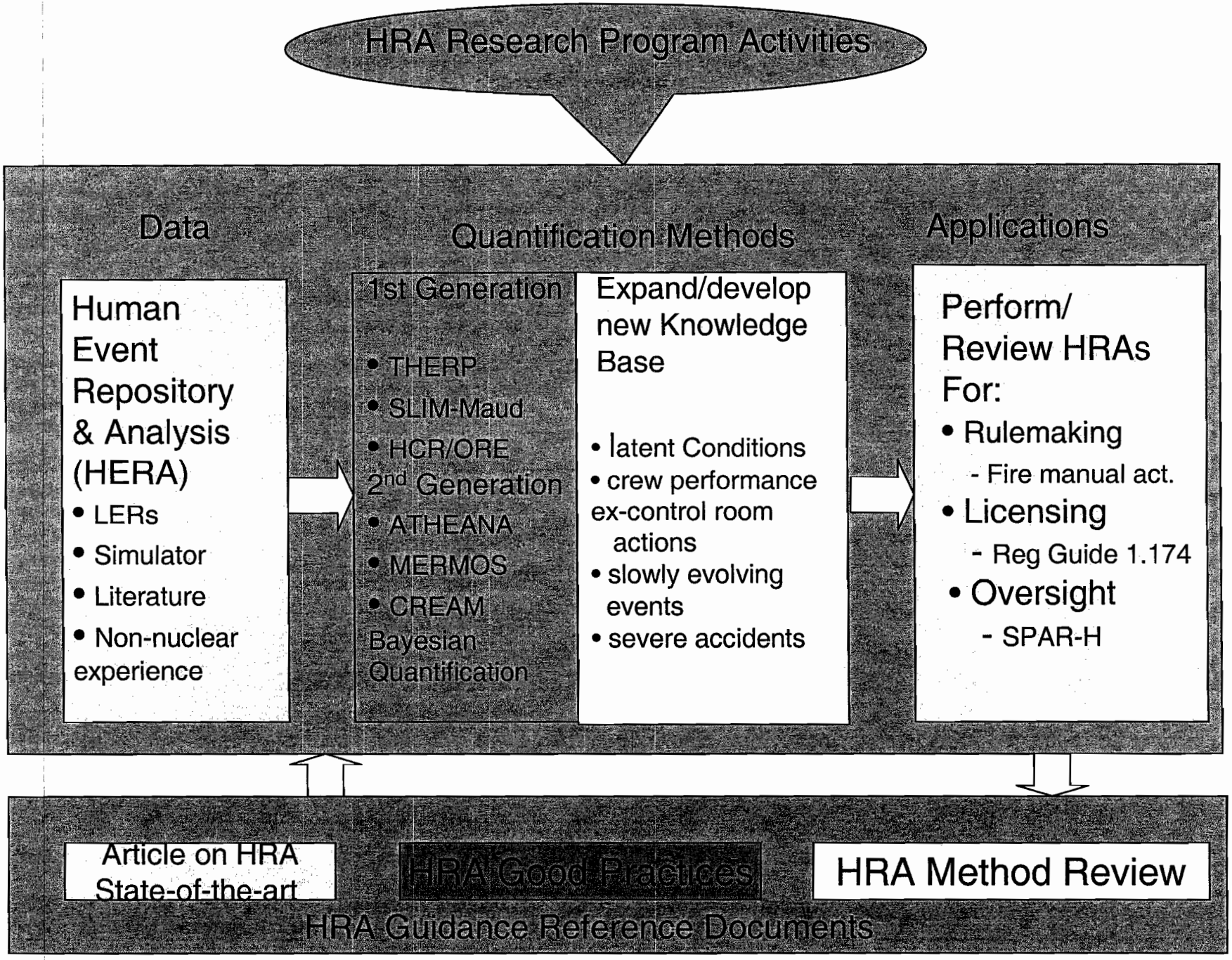
SAIC
An Employee-Owned Company

Briefing Overview & Objective

- Provide a Broad Perspective of the Human Reliability Analysis (HRA) Research Program
- Discuss in Detail the HRA Good Practices
- Request ACRS Agreement/Letter to Release the HRA Good Practices for Public Review and Comment

HRA Research Program
Issues Addressed

- HRA implementation
- Data development
- Expansion/development of new knowledge-base to address emerging NRC needs
- Specific regulatory issues


HRA Research Program Activities
Data
Quantification Methods
Applications
Human Event Repository & Analysis (HERA)
• LERs
• Simulator
• Literature
• Non-nuclear experience
1st Generation
• THERP
• SLIM-Maud
• HCR/ORE
2nd Generation
• ATHEANA
• MERMOS
• CREAM
Bayesian Quantification
Expand/develop new Knowledge Base
• latent Conditions
• crew performance ex-control room actions
• slowly evolving events
• severe accidents
Perform/ Review HRAs For:
• Rulemaking
- Fire manual act.
• Licensing
- Reg Guide 1.174
• Oversight
- SPAR-H
Article on HRA State-of-the-art
HRA Good Practices
HRA Method Review
HRA Guidance Reference Documents

HRA Guidance

3- step Approach

- Document 1: High level summary of the HRA state-of-the-art
  - Final Dec 04
- Document 2: "HRA Good Practices," provides technical guidance for performing/reviewing
  - Public Review: July 04
  - Final Dec 04
- Document 3: Evaluation of 1st and 2nd generation HRA methods w/r to good practices
  - Draft Sept 05
  - Public Review and Comment: June 06
  - Final: Dec 06

HRA Good Practices

Issue

- PRA/HRA being used
- HRA quality is important
  - HRAs need to sufficiently represent the anticipated operator performance
  - "Modeling of human performance needs to be appropriate" NRC SRP 19
- Reg. Guide 1.200 reflects ASME RA-S-2002 and NEI 00-02
  - These address "what to do" but less on "how to do it"

Solution

- Develop HRA good practices
  - Useful to HRA non-experts as well as practitioners
- A "Good Practices for HRA" document is being created
  - Working level (how to do)
  - Will produce desired HRA
  - Draft for public comment, July 2004
  - Final, December 2004

Bases and Approach for HRA Good Practices

- Bases for HRA Good Practices
  - ASME Standard/NEI PRA Review Guidance
  - Existing HRA methods and tools
  - Insights from literature
  - PRA/HRA applications
  - Experiences of authors & reviewers of the document
- Approach for development of HRA Good Practices
  - Consensus of experts at NRC
  - Internal NRC reviews
  - ACRS feedback
  - Public comment
  - International HRA input

Scope of the HRA Good Practices

- Specifically for reactor, full power, internal events; but should be useful for external events, and to some extent other modes and non-reactor applications
- Does not endorse a specific method/tool
- Linked to the ASME Standard
- Provides possible impacts of not performing good practices and additional remarks
- Focused on HRA <u>process</u> (not, for example, data)

Organization of HRA Good Practices

Organized by Logical Analysis Activities

- Overall/general
- Pre-Initiators
  - Identification
  - Screening
  - Modeling
  - Quantification
- Post-Initiators
  - Identification
  - Modeling
  - Quantification
  - Add recovery actions
- Errors of Commission (EOCs)
- Documentation

Overall/General Good Practices

1. HRA is a multi-disciplined, integrated effort within the PRA

2. Some combination of talk-throughs, walkdowns, field observations, and simulations is used as appropriate, to confirm judgments and assumptions

3. HRA addresses both core damage and large early releases

Pre-Initiator Human Event Good Practices

Identify Pre-Initiators

- GPs address
  - What to review
  - What to primarily include
    - Single or "common mode" actions affecting redundant or multiple diverse equipment

Screen Pre-Initiators

- GPs address
  - How to focus the analyses on the most important contributors
- Main points
  - Do not screen (i.e., eliminate from the analysis) failures that simultaneously affect multiple (redundant or diverse) equipment
  - Revisit the original PRA screening for applications

Model Pre-Initiators

- Covered by 1 GP that addresses:
  - How and where to include pre-initiator events in the PRA model
  - when it is acceptable to combine multiple individual acts in a single event

Quantify Pre-Initiators

- Main points from 8 GPs
  - Detailed analysis for events the were not eliminated during the screening process
  - Revisit the screening process when the PRA/HRA results are to be used for specific applications
  - Consideration of performance-shaping factors
  - Treatment of dependencies
  - Criteria for reasonable human error probabilities (i.e., make sense)

Post-Initiator Human Event Good Practices

Identify Post-Initiators

- Covered by 3 GPs that address
  - What to review
  - How to review
  - Examples of general types of actions expected to be included

Errors of Commission (EOCs)

- Encourages EOC searches
- Ensure that future plant changes do not introduce conditions prone to EOCs
- These conditions include:
  - When information to the operator could lead to a higher potential for misdiagnosis
  - When procedures and/or training could lead to a greater chance of implementation errors

Model Post-Initiators

- GPs address
  - How to model & at what level (i.e., function, system, train, component level)
  - Modeling should be based on plant & accident sequence specific characteristics
    - Sequence timing
    - Cues
    - Procedures & training
    - Location of the act
    - Insights from talk-throughs, walkdowns, & simulations

Quantify Post-Initiators

- GPs address:
  - Modeling both cognitive and execution failures
  - Quantitative screening
  - Detailed analysis of the remaining events
  - Revisit estimations for specific applications
  - Use of performance-shaping factors
  - Treatment of Dependencies
  - Mean values & uncertainties
  - Check the reasonableness of resulted estimates (i.e., make sense)

HRA Documentation

- Summary of approach, disciplines involved, and extent that talk-throughs, walkdowns, simulations were used
- Summaries of methods, processes, tools to:
- Assumptions, judgments & their bases including impacts on results/conclusions
- More detail on important HFEs (e.g., PSFs, specific dependencies…)
- Sources of data and their bases for quantification (including uncertainties)
- Results (listing of important HFEs/HEPs) and conclusions

Usefulness of HRA Good Practices

- Analysts performing HRA and particularly for plant change submittals
- Reviewers reviewing HRA and when examining plant changes for acceptability

POTENTIAL ADVERSE FLOW EFFECTS FROM POWER UPRATES

David Terao
Thomas G. Scarbrough

Mechanical and Civil Engineering Branch
Division of Engineering
NRC Office of Nuclear Reactor Regulation

May 7, 2004

INTRODUCTION/BACKGROUND

- Since 1970s, licensees have been implementing power uprates to increase NPP electric output.

- Power uprates categorized as
  - Measurement Uncertainty Recapture (about 1.5%)
  - Stretch (about 6%)
  - Extended Power Uprate (up to about 20%).

- Cracking of RPV internals is long-standing issue in BWR plants without power uprates.

- Some NPPs experiencing additional problems with safety-related and non-safety related equipment during power uprate operation.

- Quad Cities Units 1 and 2 experienced catastrophic failures of steam dryers during EPU operation.

SCOPE OF ADVERSE FLOW EFFECTS FROM POWER UPRATE OPERATION

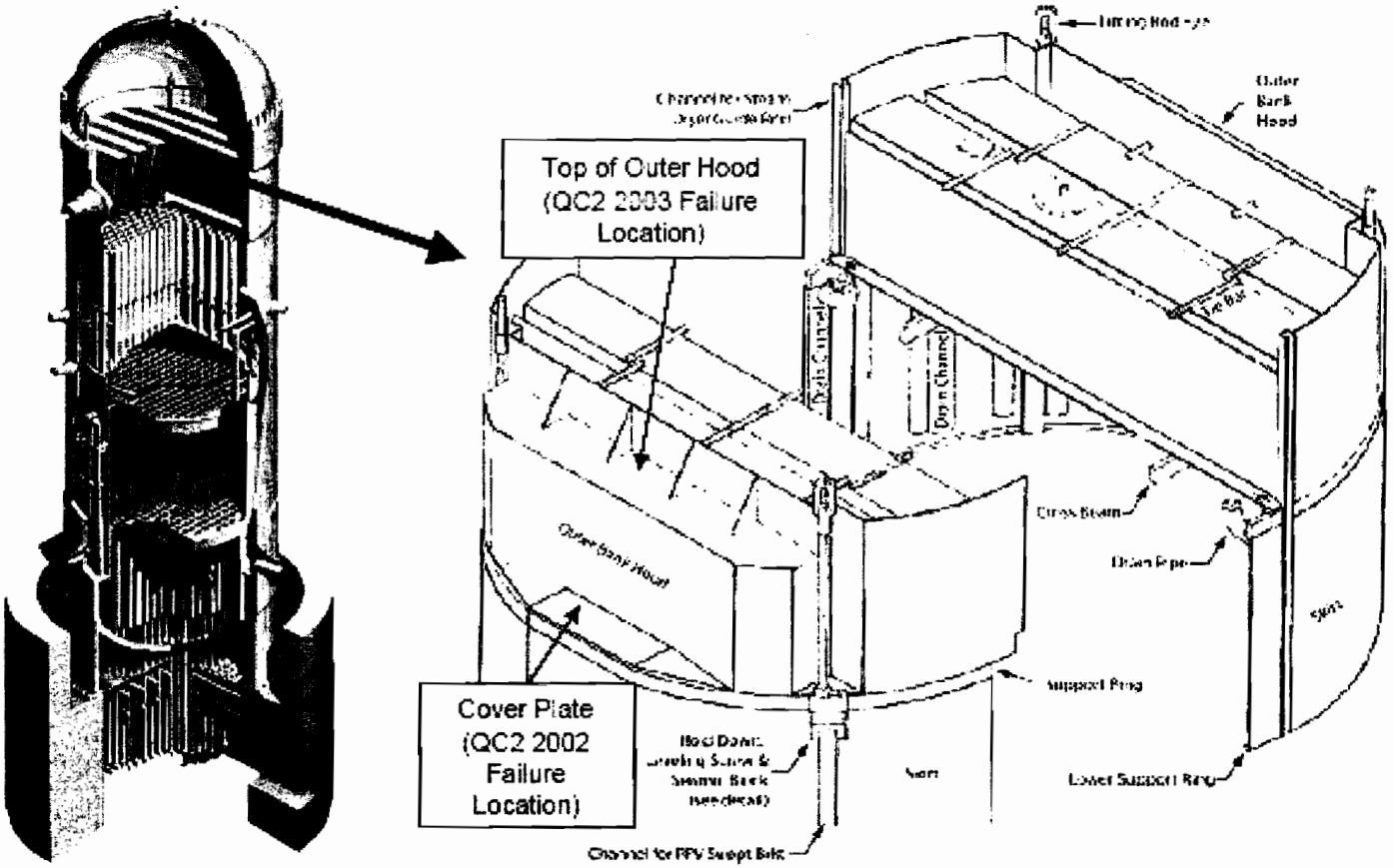
- Quad Cities Unit 2 - June 2002:

  After 90 days of EPU operation, steam dryer cover plate fails with pieces found on steam separators and in main steamline.

- Quad Cities Unit 2 - June 2003:

  After additional 300 days of EPU operation, steam dryer experiences failure of hood, internal braces, and tie bars.

QC2 Steam Dryer Failures 2002 and 2003



SCOPE OF ADVERSE FLOW EFFECTS FROM POWER UPRATE OPERATION (continued)

- Dresden Unit 2 - October 2003:

  During RFO inspection after two years of EPU operation, 4-inch cracks identified in steam dryer hood panels.

  Holes found in feedwater sparger from broken sampling probe.

SCOPE OF ADVERSE FLOW EFFECTS FROM POWER UPRATE OPERATION (continued)

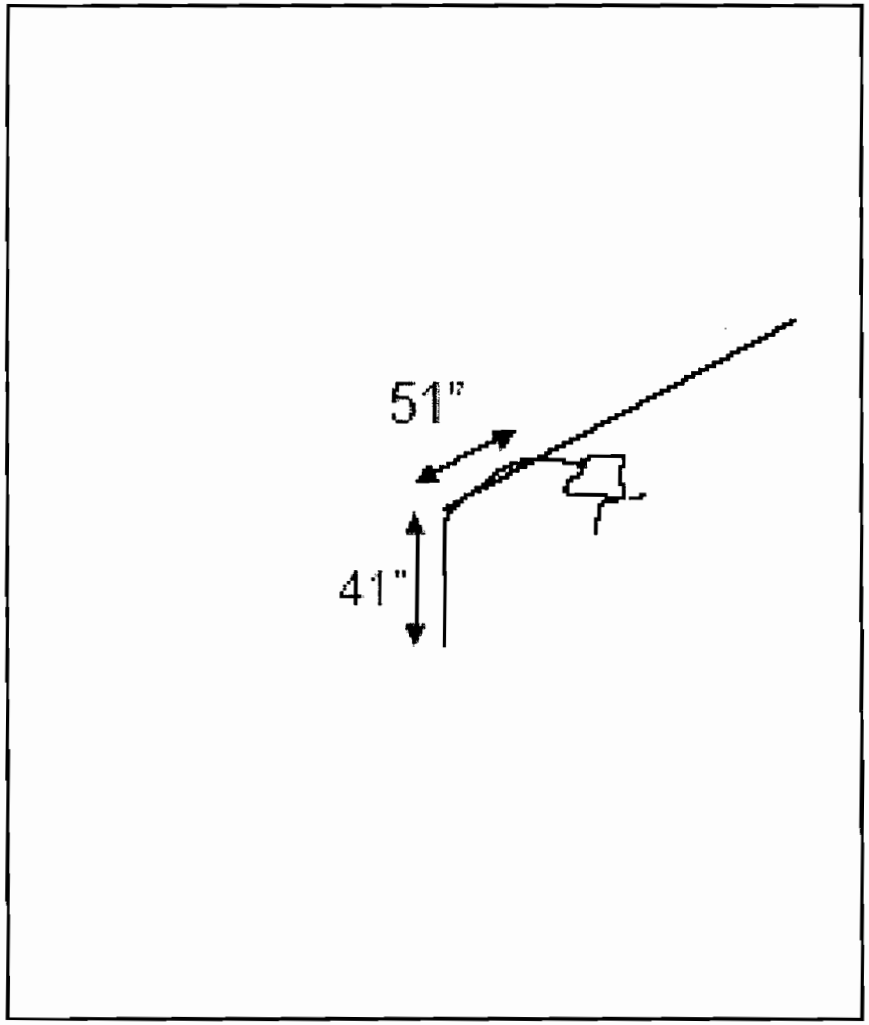
- Quad Cities Unit 1 - November 2003:

  After about one year of EPU operation, steam dryer hood experiences significant cracking with 6x9 inch piece of outer bank vertical plate missing.

  Damage also found to

  main steam electromatic relief valve (ERV),
  steamline supports, and
  HPCI steam supply motor-operated valve.

QC1 Steam Dryer Failure
November 2003



270° Side

QC1 Steam Dryer Failure
November 2003
(close-up)

Missing portion of outer bank vertical plate, approx. 6 in. x 9 in.

SCOPE OF ADVERSE FLOW EFFECTS FROM POWER UPRATE OPERATION (continued)

- Dresden Unit 3 - Dec 2003:

  During shutdown inspection after about 10 months of EPU operation,

  two 4-inch through-wall cracks identified in steam dryer hood, and

  two FW sampling probes found in sparger.

  Licensee determines FW sampling probe missing from installed location.

SCOPE OF ADVERSE FLOW EFFECTS FROM POWER UPRATE OPERATION (continued)

- Quad Cities Unit 2 - March 2004:

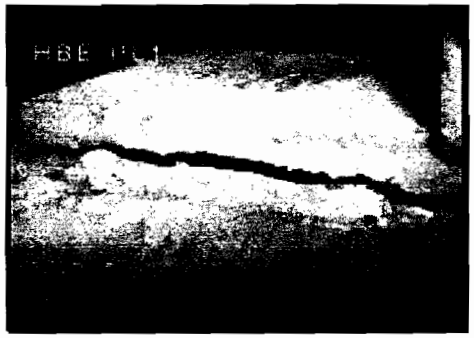
  After about 8 months of EPU operation, numerous steam dryer indications identified during refueling outage inspection including

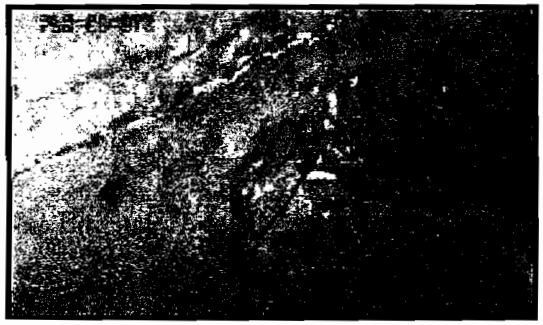
  cracking near gussets installed in 2003,

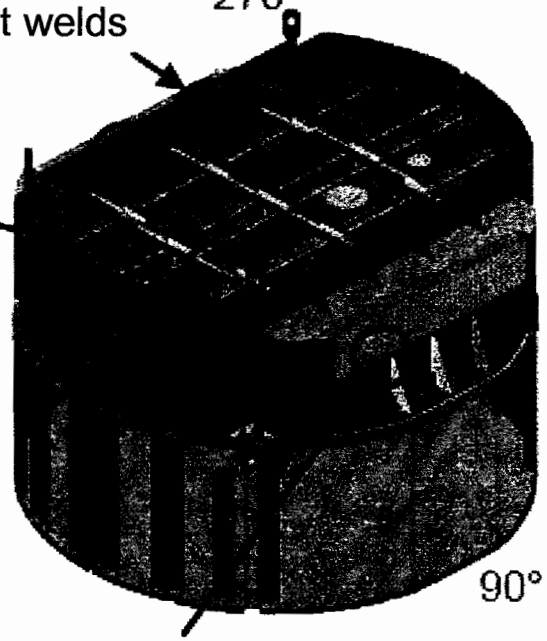
  broken tie bar welds, and

  damaged stiffener plate weld.

QC2 Steam Dryer Failure
March 2004







SCOPE OF ADVERSE FLOW EFFECTS FROM POWER UPRATE OPERATION (continued)

- Other BWR steam dryer inspections in Spring 2004:
  - Nine Mile Point Unit 2 (curved hood steam dryer) finds a thin 18-inch crack along a weld after several years of operation at 4.3% power uprate.
  - Brunswick Unit 1 (slanted hood steam dryer) finds only minor cracks after 2 years of operation at 13% power uprate.
  - Vermont Yankee (square hood steam dryer) finds minor but numerous cracks after operation at original licensed power.

POTENTIAL CAUSES OF ADVERSE FLOW EFFECTS

- July 2002 QC 2 steam dryer cover plate:

  high cycle fatigue due to high frequency resonance (180 Hz) as a result of alignment of cover plate natural frequency, standing acoustic wave frequency, and vortex shedding frequency.

- July 2003 QC 2 steam dryer hood:

  high cycle fatigue due to low frequency pressure loading (0 - 50 Hz).

- November 2003 QC 1 steam dryer:

  high cycle fatigue from fluctuating pressure loading with acoustics.

- 2003 Dresden FW probes: resonance frequency vibration.

POTENTIAL CAUSES OF ADVERSE FLOW EFFECTS (continued)

- Quad Cities and Dresden more susceptible to adverse flow effects:

  Steam dryer with square hood experiences greater stress than slanted or curve hood design.

  Main steam lines with smaller diameter have higher steam velocity.

  EPU power uprate involves more significant changes from original power level.

POTENTIAL CAUSES OF ADVERSE FLOW EFFECTS (continued)

- January 2004:

  GE identifies fluctuating pressure load in acoustic range as potential failure cause of QC steam dryers.

  Exelon study of vibration effects determines QC ERVs unable to withstand EPU vibration for full cycle.

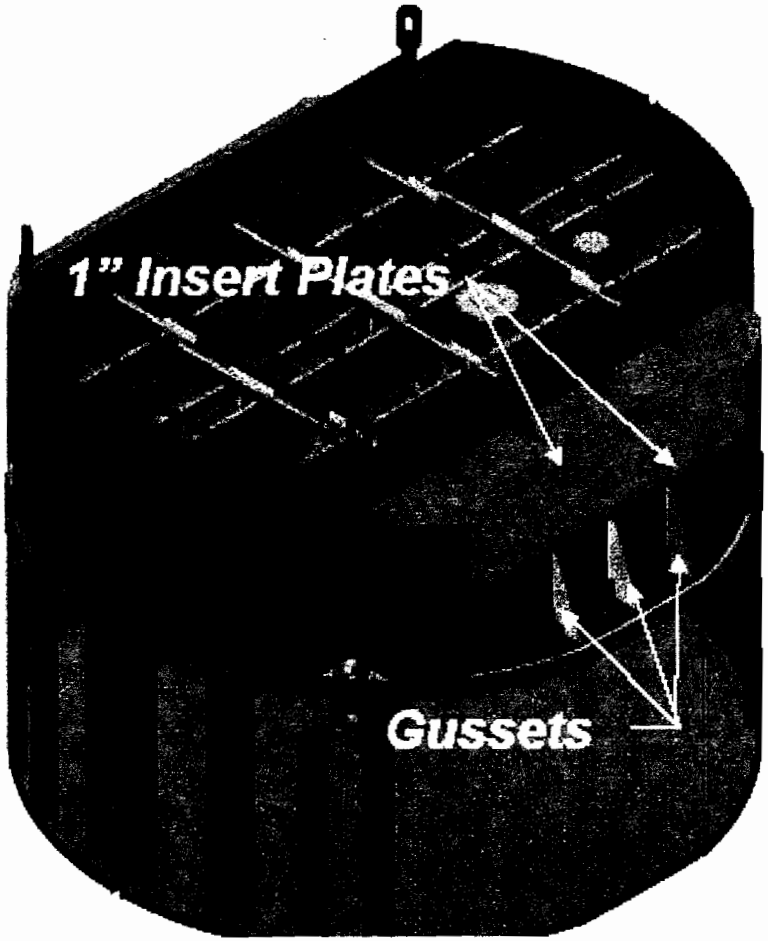
- March 2004 QC 2 steam dryer:

  inadequate design of previous gusset repair, movement of high stress point during tie bar repair, and poor installation practice for stiffener plate.

PLANT-SPECIFIC CORRECTIVE ACTIONS

- July 2002: QC 2 steam dryer cover plate increased from 0.25 to 0.5 inch.
- July 2003: QC 2 steam dryer outer hood plates increased from 0.5 to 1 inch with gussets installed and braces removed.
- Oct 2003: Dresden 2 steam dryer modified similar to QC 2 (July 2003).
- Nov 2003: QC 1 steam dryer modified similar to QC 2 (July 2003).
- Dec 2003: Dresden 3 steam dryer repair improved over QC 1 and 2.

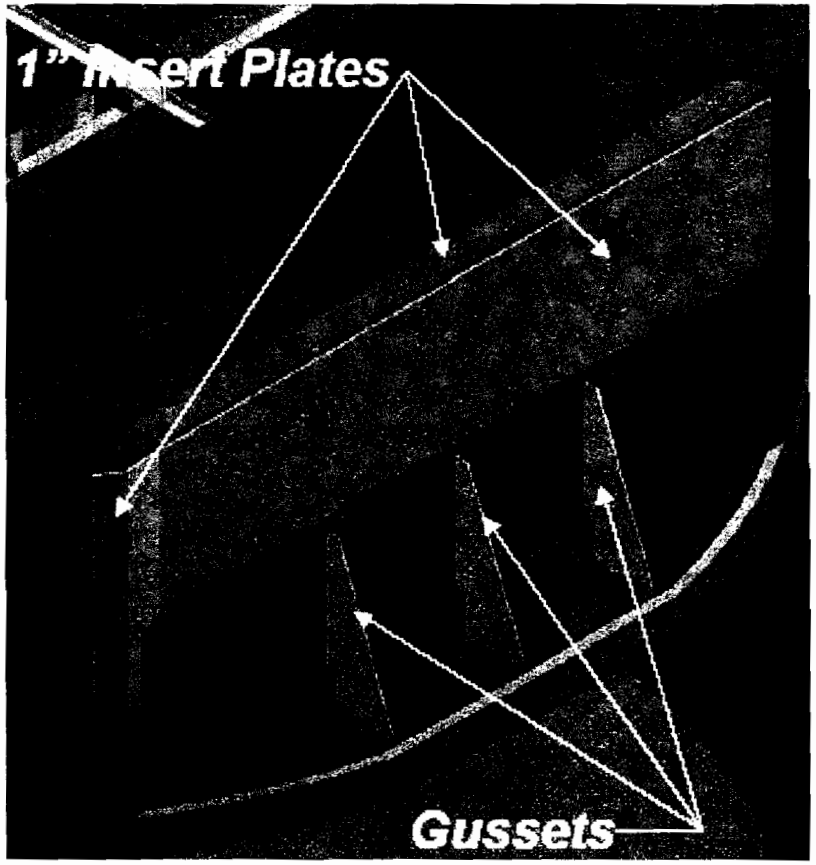
QC1 Steam Dryer Repairs
November 2003



Similar to:
QC2 - June 2003
D2 - Oct 2003

QC1 Steam Dryer Repairs
November 2003

(close-up)



PLANT-SPECIFIC CORRECTIVE ACTIONS (continued)

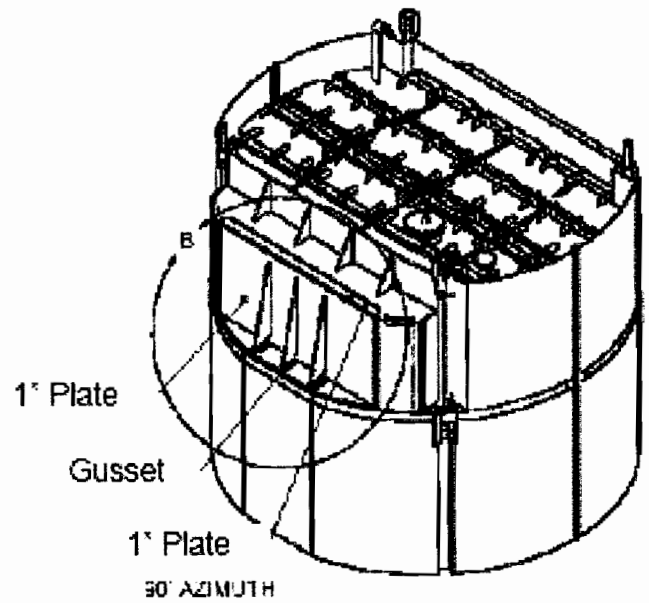
- Quad Cities Unit 2 - March 2004:

  Replacement of entire vertical plate of steam dryer hood.

  Installation of full-length gussets on vertical plate.

  ERVs strengthened to support 2-year operation.

QC2 Steam Dryer Repairs
March 2004



INDUSTRY ACTION

- Aug 2002: GE SIL 644 for square-hood steam dryers to monitor moisture carryover and RFO inspections.
- Sept 2003: Supplement 1 to SIL 644 to all BWRs with power uprates to monitor moisture carryover and RFO inspections.
- Feb 2004: BWROG assumes industry lead for EPU vibration issue.
- Mar 2004: Exelon evaluated Dresden EPU operation with RFOs for Unit 2 in Nov 2005 and Unit 3 in Nov 2004.

INDUSTRY ACTION (continued)

- Exelon Commitments - April 2, 2004:

  Limit QC 1 and 2 to pre-EPU power except for 72-hour testing.

  Modify QC 1 electromatic relief valves before long-term EPU operation.

  Provide specific commitments on

  obtaining NRC acceptance of QC 1 and 2 EPU operation;
  monitoring steam dryers and other components;
  criteria for prompt corrective action if needed;
  description of steam dryer loads;
  evaluation of QC 2 steam dryer repairs;
  independent review;
  reevaluation of flow-induced vibration assessments;
  EPU vulnerability team effort; and
  future steam dryer inspection plans.

PLANNED INDUSTRY ACTION

- May 2004: BWROG to submit plan and GE/Exelon to complete operational improvement recommendations.
- June 2004: GE to complete review of steam dryer and steam/feedwater components.
- Sept 2004: BWR Vessel and Internals Project to complete steam dryer inspection guidance.

NRC STAFF ACTION

- Sept 2002: Information Notice 2002-26 on QC 2 steam dryer cover plate failure.
- July 2003: NRC Special Inspection Team and Supplement 1 to IN 2002-26 in response to QC 2 steam dryer hood failure.
- Sept 2003: NRC letter (9/26) to BWROG with comments on SIL 644 (Supplement 1).
- Nov 2003: Public meeting (11/5) with BWROG.
- Nov 2003: NRC discussions with Exelon on QC 1 steam dryer repair and lost parts.

NRC STAFF ACTION (continued)

- Jan 2004: Supplement 2 to IN 2002-26 on QC 1 steam dryer and additional component failures.
- Feb/Mar: Public meetings (2/3 and 3/4) with BWROG.
- Mar: IN 2004-06 on loss of FW sampling probes at Dresden 2 and 3.
- Mar/Apr: NRR/RES meetings to discuss research support on adverse flow effects from power uprates.

NRC STAFF ACTION (continued)

- April: Acknowledgement letter (4/20) to Exelon:

  No problem with proceeding as described in April 2 letter.

  Concerns with plans to justify long-term EPU operation at Quad Cities and Dresden. Examples include:

  Licensee did not indicate that loads (forcing function) causing steam dryer damage will be identified.

  Quad Cities test plan not clear that sufficient data will be collected to assess dynamic loading on steam dryer and other components.

  Dresden EPU basis did not provide quantitative technical assessment of loadings and stresses that could fail steam dryer or other components.

FUTURE PLANS

- Review of Exelon information supporting Quad Cities and Dresden EPU operation.
- Regulatory communications being considered:
  - Regulatory Issue Summary on potential adverse flow effects from power uprates.
  - Generic regulatory action for other BWRs with power uprates.
- Review of Vermont Yankee power uprate request.
- Revision to power uprate review standard.

Draft Research Plan to Assess Potential Adverse Flow Effects During BWR Power Uprates

Shah Malik, MEB/DET/RES
Don Helton, SMSAB/DSARE/RES

ACRS Briefing
May 7, 2004
US NRC

Research Program Objectives

- Due to recent events at Quad Cities 1 & 2 and Dresden 2 & 3 plants, a research program is being planned in RES to address adverse flow effects due to power uprates in BWRs
- **Objectives of Research Program:**
  - Identify and determine relative significance of phenomena that cause adverse flow effects in steam dryers and other components in steam and feedwater flow paths leading to degradation and potential failures due to flow induced vibration (FIV) and high cycle fatigue
  - Apply these phenomena to characterize failures observed in BWR plants under power uprate conditions
  - Determine if there are any generic implications that can be drawn on the extent of the adverse flow effects
  - Assess feasibility of developing a screening tool that NRR can use to review submittals on BWR power uprates
  - Support NRR in evaluating BWR power uprate submittals

Research Plan (Draft)

- **2-Phase approach to understand and evaluate the adverse flow effects**
- **<u>Phase 1</u>:**
  - **With the assistance of NRR, acquire detailed plant data**
    - Affected components drawings and vibration monitoring data
    - Scaled-model test data, in-plant test data
    - Analytical modeling information (fluid and structural evaluations)
    - Licensee inspection information
  - **Procure tech. consultants in flow induced vibration (FIV)**
    - Computational fluid dynamics (CFD),
    - Fluid-structure interaction (FSI),
    - FIV computational structural dynamics analyses (FEA)
  - **Perform CFD feasibility studies to predict vortex shedding**
  - **Perform FEA structural dynamics studies (natural frequencies, mode shapes, ...)**

Research Plan (Contd.)

- **Phase 2**:
  - Determine what FIV mechanisms are of concern
    - Turbulent loading
    - Vortex shedding
    - Acoustic excitation
    - Any other mechanism
  - Predict FIV loadings via thermal-hydraulic models
  - Determine significance of fluid-structure interaction (FSI)
  - Apply FIV loadings on finite element structural dynamic models and perform analyses
  - Predict components' failure modes
  - Infer generic implications
  - Develop potential screening tools for NRR's use in review of submittals on power uprates

Research Plan Schedule (Draft)

- **Phase 1:**
  - Acquire detailed plant & analysis data: 05/2004 - 06/2004
  - Procure FIV technical consultants: 05/2004 - 09/2004
  - CFD feasibility study: 07/2004 - 02/2005
  - FEA structural dynamics studies: 07/2004 - 03/2005
- **Phase 2: (Tentative)**
  - FIV mechanisms determination: 10/2004 - 12/2004
  - Predict FIV loadings: FY05
  - Determine significance of fluid-structure interaction (FSI): FY05
  - Develop FEA structural dynamics models and perform analyses: FY05
  - Predict operating conditions and potential issues: FY05-FY06
  - Infer generic implications: FY06
  - Develop potential screening tools for assessing power uprate submittals: FY06
- **Continue providing additional guidance to NRR in reviewing submittals as soon as research information becomes available**

Fire Protection Subcommittee Report

- Resolution of Post-Fire Safe Shutdown Circuit Analysis Issues
- Revised Fire Protection SDP
- Fire Risk Re-Quantification
- Operator Manual Action Rulemaking
- 10 CFR 50.48 - NFPA 805 Rulemaking

(15)

G:PPHandout

ACRS MEETING HANDOUT

Meeting No.	Agenda Item	Handout No.:
512	12	12.1

Title **PLANNING & PROCEDURES/ FUTURE ACRS ACTIVITIES**

Authors
JOHN T. LARKINS

List of Documents Attached

PLANNING & PROCEDURES MINUTES

12

Instructions to Preparer
1. Paginate Attachments
2. Punch holes
3. Place Copy in file box

From Staff Person
JOHN T. LARKINS

May 7, 2004

G:PlanPro(ACRS):ppmins.512

INTERNAL USE ONLY

SUMMARY MINUTES OF THE
ACRS PLANNING AND PROCEDURES SUBCOMMITTEE MEETING
May 5, 2004

The ACRS Subcommittee on Planning and Procedures held a meeting on May 5, 2004, in Room T-2B1, 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 8:30 a.m. and adjourned at 10:30 a.m. A portion of this meeting was closed to discuss organizational and personnel matters that relate solely to the internal personnel rules and practices of the ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

ATTENDEES
M. Bonaca
S. Rosen

ACRS Staff
J. T. Larkins
H. Larson
R. P. Savio
S. Duraiswamy
J. Gallo
S. Steele
M. Sykes
M. Snodderly
R. Caruso
M. El-Zeftawy
M. Weston
S. Meador

NRC Staff
D. Weaver

1) Review of the Member Assignments and Priorities for ACRS Reports and Letters for the May ACRS meeting

Member assignments and priorities for ACRS reports and letters for the May ACRS meeting are attached (pp. 8-10). Reports and letters that would benefit from additional consideration at a future ACRS meeting were discussed.

RECOMMENDATION

The Subcommittee recommends that the assignments and priorities for the May ACRS meeting be as shown in the attachment (pp. 8-10).

2) Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through July 2004 is attached (pp. 8-10). 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

RECOMMENDATION

The Subcommittee recommends that the members provide comments on the anticipated workload. Changes will be made, as appropriate.

3) ACRS Meeting with the NRC Commissioners

The ACRS was previously scheduled to meet with the NRC Commissioners between 1:30 and 3:30 p.m. on Thursday, May 6, 2004, to discuss items of mutual interest. Due to the unavailability of the NRC Chairman, this meeting has been postponed to June 2, 2004, between 1:30 and 3:30 p.m. The following topics have been approved by the Commission:

1. Overview (MVB)
2. PWR Sump Performance (JDS)
3. PRA Quality (for decisionmaking) (GEA)
4. Risk-Informing 10 CFR 50.46 (WJS)
5. NRC Safety Research Program Report (DAP)
6. ESBWR Pre-Application Review (TSK)
7. Interim Review of the AP1000 Design (TSK)

Even though not scheduled as a main topic, Commissioner McGaffigan may ask for ACRS views on the Mitigating System Performance Index (MSPI) Program. The Reliability and Probabilistic Risk Assessment and Plant Operations Subcommittees held a meeting on April 14, 2004, to hear the results of the pilot program on MSPI. The Commission has recently issued an SRM (pp. 11) on this topic. During the April 2004 ACRS meeting, the Committee assigned the lead responsibility to Mr. Sieber to answer any questions on MSPI. During the April meeting and subsequent to the meeting, the members reviewed and provided comments on the presentation slides.

RECOMMENDATION

The Subcommittee recommends that the Committee discuss and approve the slides during the May ACRS meeting.

4) Revision to ACRS Action Plan

As agreed to by the Committee during its January 29-30, 2004, retreat, the ACRS Action Plan that was issued in 2001 is being revised. A proposed revision to the Action Plan includes a discussion of planned pro-active initiatives of the ACRS. A copy of the revised Action Plan will be sent to the members following the May ACRS meeting. Members are requested to provide their comments to Mrs. Weston by May 24, 2004.

RECOMMENDATION

The Subcommittee recommends that the ACRS members provide comments on the revised Action Plan to Mrs. Weston by May 24, 2004, and that Mrs. Weston prepare another revision incorporating the members' comments and submit it to the Committee for consideration during the June meeting.

5) Visit to a Nuclear Plant and Regional Office

Each year the members visit a nuclear plant and the NRC Regional Office and meet with the licensee and the Regional staff to discuss items of mutual interest. As suggested by the Committee during the April ACRS meeting, Mrs. Weston made arrangements to visit the D.C. Cook Nuclear Plant and the Region III Office on June 9-10, 2004.

RECOMMENDATION

The Subcommittee recommends that Mrs. Weston provide additional details on this matter, including an agenda for the meeting with the Region III personnel.

6) Tour of Test Facilities Used for the ACR-700 Design

The ACRS Executive Director previously suggested that some ACRS members and ACRS staff tour selected test facilities in Canada that were used for the ACR-700 design. In consultation with AECL, the NRR staff suggested that the ACRS Subcommittees on Future Plant Designs and on Materials and Metallurgy tour the Chalk River facility and hold a meeting in Canada between July 25-30, 2004, to discuss various aspects of the ACR-700 design, including materials issues.

During the April 2004 ACRS meeting, Drs. Apostolakis, Ford, Kress, Ransom, and Wallis expressed interest in touring the Chalk River Facility and participating in the meeting. Dr. El-Zeftawy is in the process of selecting specific dates in coordination with the staff.

RECOMMENDATION

The Subcommittee recommends that the ACRS staff provide additional information on this matter, including proposed topics for discussion at the meeting.

11) LINK Technologies, Inc. Report

At the request of Mr. Rosen, LINK Technologies, Inc. has prepared a report that includes recommendations for enhancing the NRC training materials for inspecting a licensee's corrective action program and explores the possibility of implementing Performance Indicators in the Reactor Oversight Process for addressing the corrective action programs. During the April 2004 meeting, the members had agreed to hear a presentation on this matter from a representative of the LINK Technologies, Inc. at the May 2004 ACRS meeting.

RECOMMENDATION

The Subcommittee recommends that Mr. Rosen propose a course of action subsequent to the briefing by LINK at the May 2004 ACRS meeting.

8) Effectiveness of Implementing Commitments Made During the ACRS Retreat

During the January 29-30, 2004 ACRS retreat, the members made several commitments. It is worthwhile for the Planning and Procedures Subcommittee to periodically assess the effectiveness of the Committee's implementation of these commitments. At this time, the following commitments have been chosen for this assessment:

- **Commitment**

  - **The members should allow uninterrupted presentations for about 10 minutes**

Effectiveness

Implementation of this commitment is ineffective. Attempt was made by the members at the February meeting to adopt this practice. Since then, it is not being followed.

RECOMMENDATION

The Subcommittee recommends that the members not interrupt the presenters for 10 minutes. The Cognizant Subcommittee Chairman and the full Committee Chairman should remind the members who deviate from this practice during the meetings.

- **Commitments**

  - **Members should identify "High Level" issues for discussion on Saturdays of the ACRS meetings.**

  - **Members should identify "Proactive" issues for discussion by the ACRS.**

Effectiveness

Very few members have responded to these commitments.

RECOMMENDATION

The Subcommittee recommends that the members periodically identify a list of "High Level" and "Proactive" issues for consideration by the Planning and Procedures Subcommittee and the full Committee and to maintain a list of such issues for use by the Committee, as warranted.

9) ACRS Review of License Renewal Applications

During the review of the license renewal applications, especially those related to SEP plants, some members raise issues that are not within the scope of 10 CFR Part 54, The License Renewal Rule. In addition, it appears that they raise questions regarding the adequacy of the current licensing basis. We need to make sure that the Committee's review is in conformance with the License Renewal Rule.

RECOMMENDATION

The Subcommittee recommends that the Committee ensure that the review of license renewal applications are in conformance with the License Renewal Rule.

10) NRC's International Council Meeting

Mr. Snodderly, ACRS Senior Staff Engineer, attended a meeting of the NRC's International Council on April 28, 2004. It was mentioned at the meeting that China appears to be serious about ordering an AP1000 reactor. The NRC Chairman has agreed to support a four day workshop in China during July 2004 to discuss design certification of AP1000. Mr. Thadani has the lead for this workshop. The workshop may have some impact on the staff review activities associated with the future plant designs.

The Committee is scheduled to review the final SER in July 2004 and issue its final report to the Commission. The Committee should consider sending Dr. Kress, Chairman of the Future Plant Designs Subcommittee, or some other member to the workshop in China.

RECOMMENDATION

The Subcommittee recommends the following:

- The ACRS staff should keep the Committee informed of any delay in schedule for ACRS review of the AP1000 final SER.

- The Committee should send Dr. Kress or some other member to the workshop in China if there is no conflict with the dates for this workshop and those for the ACRS members' visit to the Chalk River facility in Canada.

- The ACRS staff should provide a list of foreign trip reports prepared quarterly by the Office of International Programs to the Department of State.

11) Staff Requirements Memorandum on RES Activities

An April 28, 2004 SRM (pp. 12) resulting from the RES briefing to the Commission on April 13, 2004 states the following:

"The staff should inform the Commission through the budget process about how specific recommendations in the Advisory Committee on Reactor Safeguards (ACRS) report, NUREG-1635, Volume 6, 'Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program,' dated March 2004, were dispositioned by the staff."

RECOMMENDATION

The Subcommittee recommends that Dr. Powers review the adequacy of the RES response, when made available, and propose a course of action for dealing with areas of disagreement, if any.

12) Topics for Discussion on Saturday, May 8, 2004

During the retreat, the members agreed to discuss selected "high-level" issues on Saturdays of the full Committee meetings, if the scheduled work is completed ahead of time.

RECOMMENDATION

The Subcommittee recommends that time permitting, the Committee discuss the following on Saturday, May 8, 2004:

- Manual action rulemaking
- Issues associated with core power uprates
- ACRS review of license renewal applications

5) Subcommittee Meetings/Annual Plant Visit

The Subcommittee discussed the purpose, expected outcome, and appropriateness of the dates of the meetings scheduled through June 2004.

RECOMMENDATION

The Subcommittee recommends the following:

- In the future, the Subcommittee Chairmen and the ACRS staff engineers should try not to schedule a Subcommittee meeting the day after a Government holiday unless the meeting is essential to discuss a significant issue and gather information for use by the Committee in preparing a report to the Commission or the EDO at the ensuing ACRS meeting.

- The half-a-day meeting of the Future Plant Designs Subcommittee scheduled for May 25, 2004, is not an effective use of resources and should be postponed to June 24, 2004.

- The Committee should decide whether it would be more appropriate and beneficial to visit plants annually in connection with its review of license renewal and/or power uprate applications.

14) Interview of Candidates for Potential Membership on the ACRS (Closed)

The ACRS Member Candidate Screening Panel screened several applications and selected five candidates for interview during the June meeting. The Members should discuss and decide if they would like to add any additional names to the interview list. The schedule for interviewing the candidates along with their resumes will be provided to the members during the June 2004 ACRS meeting.

ANTICIPATED WORKLOAD
MAY 5-8, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly	Status of the pilot/trial use of Reg. Guide 1.200 (Formerly DG-1122) regarding PRA Quality - **Information Briefing**	--	--	--
	--	Weston	Risk Management Technical Specifications	B	To provide feedback to the staff	--
		Jain	Document on Good Practices for Human Reliability Analysis	A	To support the staff schedule for issuing this document for public document	--
Bonaca	All Members	Savio/Major	Safeguards and Security Matters (May 5, 11:00 a.m. - 6:00 p.m.) **(Closed)**	--	--	--
Ford	--	Jain	Resolution of certain items identified by the ACRS in NUREG-1740 related to DPO on Steam Generator tube integrity	A	To provide feedback to the staff	**Draft**
Powers	--	Caruso	Use of MOX Lead Test Assemblies at the Catawba Plant	A	To support the staff schedule	
	--	Nourbakhsh/ Duraiswamy	Response to SRM on divergence in regulatory approaches between U.S. and other countries	A	To respond to the Commission SRM	--
Rosen		Sykes	Subcommittee Report on Fire Protection Issues - Subc Mtg. 4/23/04	--	--	--
Sieber	--	Jain	Potential adverse flow effects from power uprates- **Information Briefing**	--	--	--

ANTICIPATED WORKLOAD
JUNE 2-4, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Apostolakis	--	Snodderly	Draft Final 10 CFR 50.69. Risk-Informed Categorization and treatment of Structures, Systems, and Components	A	To support the staff schedule	--
		Nourbakhsh	Metrics for quantitative assessment of the effectiveness (quality) of the research projects	--	--	--
Bonaca	Leitch	Sykes	Revised License Renewal Review Process - **Information Briefing**	--	--	--
	All Members	Larkins	Meeting with the NRC Commissioners (June 2, 2004, 1:30pm -3:30pm)	--	--	--
Ford	--	Duraiswamy	Update to SRP Sections (5.2.3, 5.3.1, and 5.3.3)	A	To support the staff schedule	--
		Weston	Vessel Head Penetration Degradation	--	--	--
Kress	--	El-Zeftawy	Proposed response to the March 17, 2004 ACRS Report on AP1000	--	--	--
Siber	Apostolakis	Sykes	Digital I&C Systems matters	A	To provide Committee's views	--

ANTICIPATED WORKLOAD
JULY 7-9, 2004

LEAD MEMBER	BACKUP	LEAD ENGINEER/ BACKUP	ISSUE	PRIORITY	BASIS FOR REPORT PRIORITY	AVAIL. OF DRAFTS
Kress	–	El-Zeftawy/ Duraiswamy	Draft final 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants"	A	To support the staff schedule	–
		El-Zeftawy	Final SER associated with the AP1000 design certification	A	To support the staff schedule	–
Ransom	Kress	Caruso/Weston	Maximum Extended Load Line Limit Analysis Plus (MELLA +) Licensing Topical Report	B	To provide feedback to the staff	–
Shack	–	Nourbakhsh	PTS technical basis reevaluation	A	To support the staff schedule	–
		Snodderly	Draft NUREG on 10 CFR 50.46 LB LOCA frequency reevaluation	(Report as Needed)	–	–
Wallis	–	Caruso/Sykes	Generic Letter on potential impact of debris blockage on emergency recirculation during design-basis accidents at PWRs	A	To support the staff schedule	–

IN RESPONSE, PLEASE
REFER TO: M040324B

April 8, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette L. Vietti-Cook, Secretary **/RA by J. Samuel Walker**
Acting For/

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON OFFICE OF REACTOR REGULATION (NRR) PROGRAMS, PERFORMANCE, AND PLANS, 9:30 A.M., WEDNESDAY, MARCH 24, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the status of the Office of Nuclear Reactor Regulation (NRR) programs, performance, and plans. The Commission commended the staff for their many accomplishments. The Commission encouraged the staff to bring technical issues, such as the remaining fire protection issues, to closure in a timely and effective manner. The Commission also expressed interest in and support of the staff's ability to adapt to developments in the new reactor area given the uncertainty in the plans for utilization of new reactor technology.

The Commission supports continued evaluation of enhancements to the Performance Indicator Program as a normal part of the Reactor Oversight Process. The Commission believes that resource considerations alone should not prevent a transition to a more risk-informed basis for the reactor oversight process. The staff should continue its effort to evaluate the use of a risk-informed performance indicator to replace the Safety System Unavailability (SSU) Performance Indicator (PI) in a timely manner. The staff should consider creative and practical approaches to achieve the intended purpose of this effort, including removing the "front stop" discussed during the briefing. The staff should also address lessons learned from the Mitigating Systems PI pilot. The Commission encourages the continued involvement of all stakeholders in this effort. The staff should address this issue in the Commission meeting regarding the Agency Action Review Meeting.

The Commission welcomed the staff's creation of a joint NRR and Office of Nuclear Security and Incident Response (NSIR) working group to review the safety-security interface. The working group should review NRR and NSIR processes, including licensing amendments and the 10CFR50.59 process, to ensure safety and security implications are appropriately addressed.

IN RESPONSE, PLEASE
REFER TO: M040413

April 28, 2004

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Annette Vietti-Cook, Secretary /RA/

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON RESEARCH PROGRAMS, PERFORMANCE, AND PLANS, 9:30 A.M., TUESDAY, APRIL 13, 2004, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff from the Office of Nuclear Regulatory Research (RES) on the programs, performance, and plans for the office.

The staff should communicate research results, particularly those involving conservative bounding analyses, to the public using plain English and in a manner to facilitate better understanding of the context and limitations of the information presented. When research reports are misused and quoted out of context, the staff should respond promptly.

The staff should inform the Commission through the budget process about how specific recommendations in the Advisory Committee on Reactor Safeguards (ACRS) report, NUREG-1635, Volume 6, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," dated March 2004, were dispositioned by the staff.

The Commission requested that the staff keep them currently informed on progress in the research on reactor material degradation issues.

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