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

August 19, 2010

MEMORANDUM TO: ACRS Members

FROM: Sherry Meador **/RA/**
Technical Secretary, ACRS

SUBJECT: CERTIFICATION OF THE MEETING MINUTES FROM THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS 561st FULL COMMITTEE MEETING HELD ON APRIL 2-4, 2009 IN ROCKVILLE, MARYLAND

The minutes of the subject meeting were certified on April 22, 2009 as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment:
As stated

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

April 22, 2009

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

FROM: Cayetano Santos, Chief ***/RA/***
Reactor Safety Branch
Advisory Committee on Reactor Safeguards

SUBJECT: MINUTES OF THE 561st MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), APRIL 2-4, 2009

I certify that based on my review of the minutes from the 561st 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.

OFFICE	ACRS	ACRS:RSB
NAME	SMeador	CSantos/sam
DATE	04/ 22 /09	04/22/09

OFFICIAL RECORD COPY

CERTIFIED Date Certified: 04/22/2009

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A. Reconciliation of ACRS Comments and Recommendations

B. Report on the Meeting of the Planning and Procedures Subcommittee Held on Wednesday April 1, 2009

During its 561st meeting, April 2-4, 2009, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letters, and memoranda:

REPORTS

Reports to Dale E. Klein, Chairman, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Report on the Safety Aspects of the License Renewal Application for the Vogtle Electric Generating Plant, Units 1 and 2, dated April 10, 2009

- Digital I&C Interim Staff Guidances 5, "Highly-Integrated Control Room-Human Factors Issues," and 6, "Licensing Process," dated April 21, 2009

LETTERS

Letters to R. W. Borchardt, Executive Director for Operations, NRC, from Mario V. Bonaca, Chairman, ACRS:

- Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants," dated April 21, 2009

- Draft Final Revision 2 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," dated April 9, 2009

MEMORANDA

Memoranda to R. W. Borchardt, Executive Director for Operations, NRC, from Edwin M. Hackett, Executive Director, ACRS:

- Proposed Revision 1 to Regulatory Guide 1.65 (DG-1211), "Materials and Inspections for Reactor Vessel Closure Studs," dated April 9, 2009

- Draft Final Regulatory Guides 3.16, 1.93, and 1.213, dated April 9, 2009

MINUTES OF THE 561st MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ROCKVILLE, MARYLAND

The 561st meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on April 2-4, 2009. Notice of this meeting was published in the *Federal Register* on March 16, 2009 (72 FR 11139-11140). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting agenda. The meeting was open to public attendance.

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: Dr. Mario Bonaca (Chairman), Dr. Said Abdel-Khalik (Vice-Chairman), Mr. J. Sam Armijo (Member-at-Large), Dr. George E. Apostolakis, Dr. Sanjoy Banerjee, Dr. Dennis Bley, Mr. Charles Brown, Dr. Michael Corradini, Mr. Otto L. Maynard, Dr. Dana A. Powers, Mr. Harold Ray, Dr. Michael Ryan, Dr. William Shack, Mr. John Sieber, and Mr. John Stetkar. Other attendees can be found at the sign-in sheets.

I. Chairman's Report (Open)

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

Dr. Mario Bonaca, Committee Chairman, convened the meeting at 8:30 a.m. In his opening remarks he announced that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. He reviewed the agenda items for discussion and noted that no written comments or requests for time to make oral statements from members of the public had been received. Dr. Bonaca also noted that a transcript of the open portions of the meeting was being kept and speakers were requested to identify themselves and speak with clarity and volume.

II. License Renewal Application and Final Safety Evaluation Report (SER) for the Vogtle Nuclear Plant

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

The Committee met with representatives of the NRC staff and Southern Nuclear Operating Company Inc. (SNC), the applicant, to discuss the Final Safety Evaluation Report (SER) related to the license renewal application for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2.

The applicant discussed the Boric Acid Corrosion Control Program. During the Region II site inspection, a deposit was identified in containment. The applicant stated that the deposit was not boric acid corrosion but it could potentially mask boric acid corrosion. Aggressive inspection, cleaning, and repainting have been performed, and measures are being taken to eliminate the cause of the residue. The applicant discussed water in underground cable pull boxes (manholes). The cables are routed in manholes located below ground water level and submerged in water. Since the cables are not qualified for continuous wetting and drying cycles, water could induce cable failure. Inspections are conducted more frequently and pull boxes will be equipped with sump pumps in the near future. The applicant also discussed the degradation (blisters) of Boral. The applicant stated that industry operating experience has not shown any evidence of reduced neutron absorption efficacy in Boral that has blistered. However, SNC has committed to perform a baseline and a follow-up inspection to confirm the neutron-absorbing capacity of Boral panels.

The NRC staff provided an overview of the VEGP license renewal review and the license renewal audit and inspection. The scoping and screening review, aging management review, and time-limited aging analyses were discussed. The staff also discussed flooding within the medium voltage non-safety related cable manholes. The staff has identified this as a generic current operating plant issue and will address it through the Reactor Oversight Process. The staff is also preparing a Regulatory Guide to further address this issue. The material condition inside of containment and Boral blistering were also discussed.

The VEGP final SER contained no open or confirmatory Items. Based on its review, the staff concluded that the requirements 10 CFR 54.29(a) have been met.

The Committee issued a report to the NRC Chairman on this matter, dated April 10, 2009, recommending that the SNC application for renewal of the operating licenses of VEGP, Units 1 and 2 be approved.

III. Digital Instrumentation and Control (I&C) Interim Staff Guidances (ISGs)

[Note: Mrs. Christina Antonescu was the Designated Federal Office for this portion of the meeting.]

The Committee met with representatives of the NRC staff to discuss DI&C-ISG-05, “Highly-Integrated Control Room–Human Factors Issues,” and draft DI&C-ISG-06, ”Licensing Process.”

DI&C-ISG-05 provides guidance on how to demonstrate through suitable human factors engineering (HFE) analysis that manual operator actions that can be performed inside the control room are acceptable in lieu of automated backup functions. Also, this guidance can be used to demonstrate the acceptability of operator actions required in less than thirty minutes. Specifically, a new Section 3 of this ISG was developed to provide an alternative process to the thirty-minute criterion to determine the conditions under which operator actions can be credited.

DI&C-ISG-06 clarifies the licensing criteria that the staff will use for nuclear plant license amendments in confirming that a proposed design meets applicable requirements. Specifically, the industry and vendors have requested clarification regarding what documents need to be provided to the staff for each phase of its review, which documents need to be on the docket, and which documents do not need to be docketed but should be available for staff review during the audit. The draft DI&C-ISG-06 incorporates the lessons learned from the Oconee and Wolf Creek DI&C system upgrades. Specifically, the staff has incorporated several issues in the ISG, including the need to: 1) interface early with the licensee on key technical issues such as defense-in-depth and diversity; 2) provide for more frequent feedback on the progress of the review; and 3) provide for a phased approach to the submittal of important documents for staff review.

The Committee issued a report to the NRC Chairman on this matter, dated April 21, 2009, recommending that DI&C-ISG-05, Section 3, "Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses," be revised to incorporate additional guidance on the estimation methods of the time required for operator action. The Committee also recommended that draft DI&C-ISG-06 not be issued until Sections C and Section D are revised to require sufficient design detail to ensure deterministic behavior and independence of each DI&C safety train.

IV. License Renewal Application and Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor

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

The Committee met with representatives of the NRC staff and NIST to discuss the license renewal application for the National Bureau of Standards Reactor (NBSR) and the associated NRC staff's final SER.

The staff and NIST addressed several questions raised during the Subcommittee meeting on February 4, 2009, including: monitoring of groundwater for tritium releases, adequacy of seismic analyses, relaxation of surveillance requirements, review process for in-core experiments, adequacy of natural circulation, void and moderator temperature coefficients, Class B and C wastes, single failure criterion to the accident analyses, spent fuel pool safety calculation, and Confinement Building wind and snow loads. On March 30, 2009, NIST self-identified an error in the flow coastdown data used to benchmark the RELAP5 model used to analyze a loss-of-coolant-flow accident. During the meeting, NIST explained the error in the data and outlined a plan to update the accident analysis. The staff will review the updated analysis, revise the final SER, if needed, and present its findings on this newly identified open item to the ACRS during a future meeting.

The Committee plans to review the revised final SER, which includes the resolution of the issue identified by NIST, during its June 2009 meeting.

V. Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

[Note: Mrs. Christina Antonescu was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and industry to discuss Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants."

The staff stated that this Regulatory Guide endorses the Institute of Electrical and Electronics Engineers, Inc., Standard 383-2003 with some minor clarifications and/or exceptions. One of the key regulatory positions taken by the staff in this Guide is the need for programs that monitor the environment and condition of safety-related power, instrument, and control cables. Periodic monitoring of parameters such as temperature, radiation, and other key environmental parameters is important to ensure that the environment is consistent with that used as the basis for qualification. Condition monitoring of the cables includes the use of one or more techniques to assess the actual condition of the cable.

The industry questioned the need for condition monitoring of the cables. They argued that the current methods available to monitor the condition of cables provide marginal information and benefit. They further stated that industry experience did not support the need for condition monitoring. The staff believes that industry experience justifies condition monitoring and that techniques are available to provide meaningful data for use in assessing the condition of the cables.

The Committee agrees with the staff that condition monitoring is needed and is useful as long as it can detect unexpected degradation that may require further evaluation. However, there may be situations in which condition monitoring may not provide meaningful data. For these cases, the staff should approve exceptions where reasonable testing techniques would not provide useful information or would be detrimental to the cable or other equipment. In addition, Draft Final Regulatory Guide 1.211 states that condition monitoring may be limited to safety-related cables within the scope of the Maintenance Rule. The staff agreed to revise the Guide to clarify that the scope is for safety-related cables and remove the reference to the Maintenance Rule.

The Committee issued a letter to the Executive Director for Operations on this matter, dated April 21, 2009, recommending that the staff issue Regulatory Guide 1.211 after clarifying the scope for condition monitoring. The Committee encouraged the staff to remain cognizant of advances in condition monitoring techniques and industry experience so it can refine future guidance in this area.

VI. Risk Metrics for New Light-Water Reactor Risk-Informed Applications

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

The Committee met with representatives of the NRC staff and the Nuclear Energy Institute (NEI) to discuss risk metrics for new light-water reactor risk-informed applications. The existing risk goals for new reactors are a core damage frequency (CDF) < 10^{-4}/year, a large release frequency (LRF) < 10^{-6}/year, and a conditional containment failure probability < 0.1. These numeric goals are not completely consistent with each other. Moreover, LRF has never been explicitly defined and differs from the large early release frequency (LERF) parameter used in risk metrics for current operating reactors. The current basis is that increases should be limited to small increments.

The new reactor designs pose the following questions because the risk estimates for these designs are much lower than those of the current operating plants:

(1) Should the principle of a "small" increase in ΔCDF, ΔLERF, or ΔLRF be based on its absolute value or as a certain fraction of the unit's overall CDF, LERF, or LRF?

(2) Should there be an alternate or additional ΔLRF guideline (in contrast to the existing ΔLERF guideline) for new reactors?

The staff described 6 options it is considering to address these questions.

Representatives of NEI provided an industry perspective on the various options. They stated that new risk metrics would penalize new plants and the Commission has been consistent in maintaining that new reactors would not be measured against a more stringent risk requirement. NEI also discussed the relationship of the LRF and LERF parameters. LERF is defined as any large unscrubbed release of radionuclide inventory. Although LRF has never been explicitly defined by the agency, the early uses of LRF were based on a release of radionuclides large enough to cause an offsite fatality. By this definition, "LRF" is a frequency of events resulting in a fatality, not a frequency of large releases. NEI examined the accident sequences in the 5 NUREG-1150 probabilistic risk assessments (PRAs) and showed that, using these definitions, a radioactive release large enough to be counted in LERF would often not be large enough to be counted in LRF. NEI also found in these PRAs that the LERF was more than a factor of 10 greater than LRF. This is counter-intuitive in that the normal expectation would be that the LERF would be a subset of LRF. The reason is that, according to NEI, the two parameters use different definitions of "large." This was an information briefing. No Committee action was necessary.

VII. Subcommittee Reports

Plant License Renewal Subcommittee Report (Three Mile Island (TMI) Nuclear Station, Unit 1 License Renewal Application)

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the April 1, 2009, meeting, with the NRC staff and representatives of

Exelon Generation Group, LLC, (EGC) and the Committee on Health Aspects & Management of Nuclear Power, to review the draft SER related to license renewal application for the TMI Nuclear Station Unit 1. The NRC staff's draft SER, issued in March 2009, contained no open items. EGC submitted an application to the NRC in January 2008 to extend the TMI Unit 1 license by 20 years.

During the meeting, EGC representatives described the operating history, the license renewal review methodology, and the aging management programs. In addition, one confirmatory item was discussed by the applicant concerning the bounding assumption of less than 0.050 ppm for reactor coolant dissolved oxygen used in fatigue calculations. The Boral Surveillance Program, reactor building liner, and medium voltage cables were also discussed by both the staff and applicant. The Chairman of the Subcommittee noted that EGC takes credit for an Electric Power Research Institute report (Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools) as a surrogate for a review of plant-specific operating experience. The representative of the Committee on Health Aspects & Management of Nuclear Power provided comments concerning environmental and health issues relating to the renewal of the TMI, Unit 1 license. The Committee plans to review the final SER related to the license renewal application for the TMI, Unit 1 in September 2009.

Plant License Renewal Subcommittee Report (Susquehanna Steam Electric Station License Renewal Application)

The Chairman of the Plant License Renewal Subcommittee provided a report to the Committee summarizing the results of the April 1, 2009, meeting with representatives of the NRC staff and PPL Susquehanna, LLC (PPL) to review the draft SER with Open Items related to the license renewal application for the Susquehanna Steam Electric Station (SSES), Units 1 and 2.

The current operating licenses for the two SSES units expire on July 17, 2022, and March 23, 2024, respectively. The staff's draft SER, issued in March 2009, contained no open items. During the meeting, PPL described the plant, its operating history, the license renewal review methodology, the aging management programs, and its commitment tracking system. The staff discussed its review and inspection results, which include scoping and screening, license renewal inspections, aging management programs, and time-limited aging analyses. The staff informed the Subcommittee that it is still working with the applicant on two emerging issues: Boral surveillance and cycle counting in the fatigue monitoring program. The Committee plans to review the final SER related to the license renewal application for the SSES, Units 1 and 2 in October, 2009.

Reliability and PRA Subcommittee Report (NUREG-1855 Appendix A)

The Chairman of the Reliability and PRA Subcommittee provided a report to the Committee summarizing the results of the March 27, 2009, Subcommittee meeting with representative of the NRC staff, the Electric Power Research Institute, Brookhaven National Laboratory, Sandia National Laboratories, and ERIN Engineering to review the staff's revisions to NUREG-1855 Appendix A, "Example Implementation of the Process for the Treatment of PRA Uncertainty in a Risk-Informed Regulatory Application." The Committee had previously issued a letter approving the publication of NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRA in Risk-Informed Decisionmaking." However, the Committee had recommended that Appendix A to this NUREG not be published until it is revised. The staff has added numerous footnotes, explanatory text, and additional discussion to address the Committee's concerns. Moreover, the staff plans to hold a workshop with representatives of the industry and other interested stakeholders in May 2009. The Subcommittee plans to meet with the staff again to discuss the results of the workshop and any further modifications to the Appendix.

VI. Executive Session

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

A. Reconciliation of ACRS Comments and Recommendations/EDO Commitments

- The Committee considered the EDO's response of February 13, 2009, to comments and recommendations included in the December 22, 2008, ACRS letter on Chapters 7 and 14 of the NRC staff's SER with Open items related to the certification of the ESBWR design. The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of March 19, 2009, to comments and recommendations included in the February 18, 2009, letter on SECY-08-0917, "Review of Options to Revise Radiation Protection Regulations and Guidance with Respect to the 2007 Recommendations of the International Commission on Radiological Protection." The Committee decided that it was satisfied with the EDO's response.

- The Committee considered the EDO's response of March 25, 2009, to comments and recommendations included in the February 18, 2009, ACRS letter on the Draft Final Regulatory Guide DG-5021, "Managing the Safety/Security Interface." The Committee decided that it was satisfied with the EDO's response.

B. Report of the Planning and Procedures Subcommittee Meeting

Review of the Member Assignments and Priorities for ACRS Reports and Letters for the April ACRS Meeting

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through June 2009 were discussed and the objectives were 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

Annual Visit to a Nuclear Plant and Meeting with the Regional Administrator

During the March 2009 ACRS meeting, the members decided to tour Watts Bar and meet with the Regional Administrator in Region II. Subsequent to the March meeting, Dr. Cecil Parks, ORNL, contacted the ACRS Executive Director to extend an invitation to the ACRS members to visit selected ORNL facilities. In addition, a representative of INPO agreed to accommodate the members' visit to INPO. ORNL has recommended a list of facilities for the members' tour and ORNL is preparing a list of discussion topics for the visit. This list will be provided to the members during the May meeting.

Software PRA Workshop

The Office of Nuclear Regulatory Research (RES) has invited the ACRS members to observe a 2-day workshop scheduled for May 5-6, 2009, at the Brookhaven National Laboratory to discuss issues related to addressing software failures in a nuclear plant PRA. This Workshop will address issues of likely interest to the ACRS Subcommittees on Digital I&C Systems and on Reliability and PRA. The primary purpose of this Workshop is to establish and document a philosophical basis to including software failures in a PRA in terms of failure rates or failure probabilities.

Draft Final Regulatory Guides

The staff plans to issue the following Draft Final Regulatory Guides and would like to know whether the Committee wants to review these Guides prior to being issued final.

- Draft Final Revision 1 to Regulatory Guide 3.16, "General Fire Protection Guide for Plutonium Processing and Fuel Fabrication Plants"

Revision 1 to Regulatory Guide 3.16 endorses the guidance in Chapter 7, "Fire Protection" of NUREG-1718, "Standard Review Plan for the Review of an Application for Mixed Oxide (MOX) Fuel Fabrication Facility," as a process acceptable to the staff for reviewing the fire protection portion of a licensing application for MOX or plutonium processing facilities. This version of this Guide reflects incorporation of public comments. No significant changes have been made as a result of public comments.

Based on his review of this Guide, Dr. Powers recommends that the Committee not review this Guide.

- Draft Final Regulatory Guide 1.213, "Qualification of Motor Control Centers for Nuclear Power Plants"

Regulatory Guide 1.213 endorses, with some modifications, the qualification guidelines of the Institute of Electrical and Electronic Engineers (IEEE) Standard 649-2006 , "IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations," as an acceptable method of demonstrating that the motor control centers are capable of performing their required safety functions with no failure mechanisms that could lead to common mode failures under the postulated conditions stated in the equipment specifications. This version of this Guide reflects incorporation of public comments. No significant changes have been made as a result of public comments.

Based on his review of this Guide, Mr. Maynard recommends that the Committee not review this Guide.

- Draft Final Revision 1 to Regulatory Guide 1.93, "Availability of Electric Power Sources"

Revision 1 of Regulatory Guide 1.93 describes the operating procedures and restrictions the staff believes should be implemented if the number of available electric power sources is less than the limiting conditions for operation in the nuclear plant's technical specifications. This version of this Guide reflects incorporation of public comments. No significant changes have been made as a result of public comments.

Based on his review of this Guide, Mr. Sieber recommends that the Committee not review this Guide.

Proposed Regulatory Guide

The staff plans to issue the following proposed Regulatory Guide (DG) for public comment and would like to know whether the Committee wants to review this document prior of being issued for public comment.

- Proposed Revision 1 to Regulatory Guide 1.65 (DG-1211), "Materials and Inspections for Reactor Vessel Closure Studs"

The staff issued the initial RG 1.65 in October 1973. Changes have been made to DG-1211, first to add a staff position previously established for license renewal and provided in NUREG/CR-1801, "Generic Aging Lessons Learned," regarding the design conservatism exercised in determining the sizing of the reactor vessel closure studs. The staff position is that the measured ultimate tensile strength should not reach a level that would make the studs susceptible to stress corrosion cracking. Second, the updated guidance reflects provisions in NUREG-0800, "Standard Review Plan," Section 3.13, "Threaded Fasteners – ASME Code Classes 1, 2, and 3", addressing the operating experience regarding the types of lubricant that are acceptable and those that have been found to be detrimental.

Webstreaming of ACRS Meetings

In a March 6, 2009 Staff Requirements Memorandum (SRM) related to Webstreaming of ASLBP Adjudicatory proceedings, the Commission stated the following:

If the ACRS decides that it wishes to pursue webstreaming of ACRS meetings, the ACRS should prepare a proposed plan reflecting their interests, in consultation with the Office of Administration. An ACRS plan could include the results of any further discussions between the ACRS staff and the ASLBP, and should be submitted to the Office of Administration, with a copy to the Commission.

Third Quadripartite Working Group Meeting

Japan's Nuclear Safety Commission (NSC) will host the third Quadripartite Working Group (WG) Meeting in Tokyo either the week of October 5, 2009 or the week of October 12, 2009 on the general topic of "Digital Instrumentation & Control" (Digital I&C). A tentative agenda was discussed.

NRC Budget for FY-2009

NRC's FY-2009 budget was approved on March 11, 2009 at $1.04 billion. This represents an increase of $28.5 million over that originally requested by the NRC and 13% more than the FY-2008 funding level. Approval of the NRC budget eliminates the restrictions imposed when operating under the Continuing Resolution.

Miscellaneous

- ACRS letters dating back to 1985 and ACNW&M letters from 1988 are now available on the ACRS website.

- Demolition of the ACRS Subcommittee room is expected to begin next week.

- Construction of the new conference room is expected to be completed in September 2009.

The meeting was adjourned at 7:00 p.m. on April 3, 2009.

Further information regarding this meeting can be obtained by contacting the Designated Federal Officer between 7:30 a.m. and 4 p.m. (ET). 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: March 10, 2009.

Cayetano Santos,

Chief, Reactor Safety Branch A, Advisory Committee on Reactor Safeguards.

[FR Doc. E9–5613 Filed 3–13–09; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards (ACRS); Meeting of the Subcommittee on Plant License Renewal

Notice of Meeting

The ACRS Subcommittee on Plant License Renewal will hold a meeting on April 1, 2009, Room T–2B3, 11545 Rockville Pike, Rockville, MD.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Wednesday, April 1, 2009—1:30 p.m. until 5 p.m.

The Subcommittee will discuss the Three Mile Island License Renewal Application and the associated Safety Evaluation Report (SER) with Open Items prepared by the staff. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Exelon Generation Company, LLC, and other interested persons regarding this matter. 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. Christopher Brown (telephone 301–415–7111) five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted. Detailed procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 6, 2008 (73 FR 58268–58269).

Further information regarding this meeting can be obtained by contacting the Designated Federal Official between 6:45 a.m. and 3:30 p.m. (ET). 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 to the agenda.

Dated: March 10, 2009.

Cayetano Santos,

Chief, Reactor Safety Branch A, Advisory Committee on Reactor Safeguards.

[FR Doc. E9–5622 Filed 3–13–09; 8:45 am]

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NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards

In accordance with the purposes of Sections 29 and 182b of the Atomic Energy Act (42 U.S.C. 2039, 2232b), the Advisory Committee on Reactor Safeguards (ACRS) will hold a meeting on April 2–4, 2009, 11545 Rockville Pike, Rockville, Maryland. The date of this meeting was previously published in the **Federal Register** on Monday, October 6, 2008, (73 FR 58268–58269).

Thursday, April 2, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10 a.m.: License Renewal Application and Final Safety Evaluation Report (SER) for the Vogtle Nuclear Plant (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and Southern Nuclear Operating Company (SNC) regarding the Vogtle Nuclear Plant License Renewal Application, the associated NRC staff's final SER, and related matters.

10:15 a.m.–12 p.m.: Digital Instrumentation and Control (I&C) Interim Staff Guidances (ISGs) (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding Digital I&C ISGs on: Highly Integrated Control Room—Human Factors, Licensing Process Issues, and related matters.

1 p.m.–2:30 p.m.: License Renewal Application and Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and NIST regarding the License Renewal Application for the NIST Reactor, the associated NRC staff's final SER, and related matters.

2:45 p.m.–4:15 p.m.: Draft Final Regulatory Guide 1.211—''Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants'' (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff and the Nuclear Utility Group on Equipment Qualification (NUGEQ) regarding Draft Final Regulatory Guide 1.211 and related matters.

4:30 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports on matters discussed during this meeting, as well as a proposed report on the Draft Final Revision 2 to Regulatory Guide 1.200, ''An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.''

Friday, April 3, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

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

8:35 a.m.–10:30 a.m.: Risk Metrics for New Light-Water Reactor Risk-Informed Applications (Open)—The Committee will hear presentations by and hold discussions with representatives of the NRC staff regarding risk metrics for new light-water reactor risk-informed applications and related matters.

10:45 a.m.–11 a.m.: Subcommittee Reports (Open)—The Committee will hear reports by and hold discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim reviews of the Three Mile Island Unit 1 and the Susquehanna Steam Electric Station License Renewal Applications, and the associated NRC staff's Safety Evaluation Reports with Open Items. The Committee will also hear a report by and hold discussions with the Chairman of the Reliability and PRA Subcommittee regarding revisions to NUREG–1855, Appendix A, ''Example Implementation of the Process for the Treatment of PRA Uncertainty in a Risk-Informed Regulatory Application.''

11 a.m.–11:45 a.m.: Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed)—The Committee will discuss the recommendations of the Planning and Procedures Subcommittee regarding items proposed for consideration by the full Committee during future ACRS meetings and other matters related to the conduct of the ACRS.

[*Note:* A portion of this session may be closed pursuant to 5 U.S.C. 552b (c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.]

11:45 a.m.–12 p.m.: Reconciliation of ACRS Comments and Recommendations (Open)—The Committee will discuss the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

1 p.m.–7 p.m.: Preparation of ACRS Reports (Open)—The Committee will discuss proposed ACRS reports.

Saturday, April 4, 2009, Conference Room T–2B3, Two White Flint North, Rockville, Maryland

8:30 a.m.–1 p.m.: Preparation of ACRS Reports (Open)—The Committee will continue its discussion of proposed ACRS reports.

1 p.m.–1:15 p.m.: Miscellaneous (Open)—The Committee will discuss matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACRS meetings were published in the **Federal Register** on October 6, 2008, (73 FR 58268–58269). In accordance with those procedures, oral or written views may be presented by members of the public, including representatives of the nuclear industry. Electronic recordings will be permitted only during the open portions of the meeting. Persons desiring to make oral statements should notify the Cognizant ACRS staff named below five days before the meeting, if possible, so that appropriate arrangements can be made to allow necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during the meeting may be limited to selected portions of the meeting as determined by the Chairman. Information regarding the time to be set aside for this purpose may be obtained by contacting the Cognizant ACRS staff prior to the meeting. In view of the possibility that the schedule for ACRS meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should check with the Cognizant ACRS staff if such rescheduling would result in major inconvenience.

In accordance with Subsection 10(d) Public Law 92–463, I have determined that it may be necessary to close a portion of this meeting noted above to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which constitute a clearly unwarranted invasion of personal privacy pursuant to 5 U.S.C. 552b(c)(2) and (6).

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, as well as the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by contacting Girija Shukla, Cognizant ACRS staff (301–415–6855), between 7:15 a.m. and 5 p.m. (ET). ACRS meeting agenda, meeting transcripts, and letter reports are available through the NRC Public Document Room at *pdr.resource@nrc.gov,* or by calling the PDR at 1–800–397–4209, or from the Publicly Available Records System (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at *http://www.nrc.gov/reading-rm/adams.html* or *http://www.nrc.gov/reading-rm/doc-collections/ACRS/.*

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

Dated: March 10, 2009.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. E9–5610 Filed 3–13–09; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

[Docket No. 50–315; License No. DPR–58; NRC–2009–0111]

Indiana Michigan Power Company; Receipt of Request for Action Under 10 CFR 2.206

Notice is hereby given that by petition dated December 16, 2008, Mr. David Lochbaum (petitioner) has requested that the U.S. Nuclear Regulatory Commission (NRC) take action with regard to the licensee for the Donald C. Cook Nuclear Power Plant, Unit 1. The petitioner requests that the NRC issue a demand for information requiring the licensee to docket information regarding the following four issues at least 30 days before restarting the reactor from the current outage:

(1) The vibration levels experienced in the control room, turbine building, and other structures during the September 20, 2008, event;

(2) The vibration levels assumed in these locations during the safe-shutdown earthquake (SSE);

(3) In locations where the vibration levels during the September 20, 2008, event exceeded the vibration levels assumed for an SSE, the extent to which piping and pipe supports were replaced or repaired because of potential stress damage and the bases for not replacing other structures, systems, and components exposed to greater than SSE loading; and

(4) In locations where the vibration levels during the September 20, 2008, event did not exceed the vibration levels assumed for an SSE, the extent of the measures taken to protect against spurious equipment operation and the bases for concluding that the as-left configuration will not pose a public health hazard in event of an SSE.

As the basis for this request, the petitioner states that the event caused significant vibration levels that caused spurious operation of standby equipment and may have contributed to a breach that seriously impaired the fire protection system. The petitioner further states that the information being sought in the petition is needed to apply the proper lessons from the event to the future operation of the Donald C. Cook Nuclear Plant, Unit 1. The petitioner states that without this information, the NRC cannot be assured of, and the public is therefore not adequately protected from, significant adverse safety implications of an SSE causing spurious actuation of equipment.

The NRC is treating the request pursuant to Title 10, Section 2.206, ''Requests for Action under This Subpart,'' of the *Code of Federal Regulations* (10 CFR 2.206). The request has been referred to the Director of the Office of Nuclear Reactor Regulation. As provided by 10 CFR 2.206, the NRC will take appropriate action on this petition within a reasonable time. A copy of the petition is available for inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville,

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

March 10, 2009

AGENDA
561st ACRS MEETING
APRIL 2-4, 2009

THURSDAY, APRIL 2, 2009, 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/EMH/SD)
   1.1) Opening statement
   1.2) Items of current interest

2) 8:35 – 10:00 A.M. License Renewal Application and Final Safety Evaluation Report (SER) for the Vogtle Nuclear Plant (Open) (JDS/CLB)
   2.1) Remarks by the Subcommittee Chairman
   2.2) Briefing by and discussions with representatives of the NRC staff and Southern Nuclear Operating Company (SNC) regarding the Vogtle Nuclear Plant License Renewal Application, the associated NRC staff's final SER, and related matters.

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

10:00 – 10:15 A.M. ***** BREAK *****

3) 10:15 – 12:00 P.M. Digital Instrumentation and Control (I&C) Interim Staff Guidances (ISGs) (Open) (CB/CEA)
   3.1) Remarks by the Subcommittee Chairman
   3.2) Briefing by and discussions with representatives of the NRC staff regarding Digital I&C ISGs on Highly Integrated Control Room - Human Factors, Licensing Process Issues, and related matters.

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

12:00 – 1:00 P.M. ***** LUNCH *****

4) 1:00 – 2:30 P.M. License Renewal Application and Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor (Open) (JDS/PW)
   4.1) Remarks by the Subcommittee Chairman
   4.2) Briefing by and discussions with representatives of the

NRC staff and NIST regarding the License Renewal Application for the NIST Reactor, the associated NRC staff's final SER, and related matters.

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

2:30 – 2:45 P.M. ***** BREAK**

5) 2:45 – 4:15 P.M. Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants" (Open) (OLM/CEA)

5.1) Remarks by the Subcommittee Chairman

5.2) Briefing by and discussions with representatives of the NRC staff and the Nuclear Utility Group on Equipment Qualification (NUGEQ) regarding draft final Regulatory Guide 1.211 and related matters.

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

4:15 – 4:30 P.M. ***** BREAK**

6) 4:30 – 7:00 P.M. Preparation of ACRS Reports (Open)

Discussion of proposed ACRS reports on:

6.1) License Renewal Application and Final SER for the Vogtle Nuclear Plant (JDS/CLB)

6.2) Digital I&C Interim Staff Guidances on Highly Integrated Control Room - Human Factors and Licensing Process Issues (CB/CEA)

6.3) License Renewal Application and Final SER for the National Institute of Standards and Technology Reactor (JDS/PW)

6.4) Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants" (OLM/CEA)

6.5) Draft Final Revision 2 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (DCB/HPN)

FRIDAY, APRIL 3, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

7) 8:30 – 8:35 A.M. Opening Remarks by the ACRS Chairman (Open) (MVB/CS/SD)

8) 8:35 – 10:30 A.M. Risk Metrics for New Light-Water Reactor Risk-Informed Applications (Open) (GEA/HJV/HPN)

8.1) Remarks by the Subcommittee Chairman

8.2) Briefing by and discussions with representatives of the NRC staff regarding risk metrics for new light-water reactor risk-informed applications and related matters.

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

10:30 – 10:45 A.M. * BREAK ***

9) 10:45 – 11:00 A.M. Subcommittee Reports (Open)

9.1) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the Three Mile Island Unit 1 License Renewal Application and the SER with Open Items that were discussed during the Subcommittee meeting on April 1, 2009 (JWS/CLB)

9.2) Report by and discussions with the Chairman of the Plant License Renewal Subcommittee regarding interim review of the Susquehanna Steam Electric Station License Renewal Application and the SER with Open Items that were discussed during the Subcommittee meeting on April 1, 2009 (WJS/PW)

9.3) Report by and discussions with the Chairman of the Reliability and PRA Subcommittee regarding revisions to NUREG-1855, Appendix A, "Example Implementation of the Process for the Treatment of PRA Uncertainty in a Risk-Informed Regulatory Application," that were discussed during the Subcommittee meeting on March 27, 2009 (GEA/HJV)

10) 11:00 – 11:45 A.M. Future ACRS Activities/Report of the Planning and Procedures Subcommittee (Open/Closed) (MVB/EMH)

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

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

[NOTE: A portion of this session may be closed pursuant to 5 U.S.C. 552b (c)(2) and (6) to discuss organizational and personnel matters that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy]

11) 11:45 – 12:00 P.M. Reconciliation of ACRS Comments and Recommendations (Open) (MVB/CS/AFD)
Discussion of the responses from the NRC Executive Director for Operations to comments and recommendations included in recent ACRS reports and letters.

12:00 – 1:00 P.M. * LUNCH ***

12) 1:00 – 7:00 P.M. Preparation of ACRS Reports (Open)
Discussion of proposed ACRS reports on:

12.1) License Renewal Application and Final SER for the Vogtle Nuclear Plant (JDS/CLB)

12.2) Digital I&C Interim Staff Guidances on Highly Integrated Control Room - Human Factors and Licensing Process Issues (CB/CEA)

12.3) License Renewal Application and Final SER for the National Institute of Standards and Technology Reactor (JDS/PW)

12.4) Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants" (OLM/CEA)

12.5) Draft Final Revision 2 to Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (DCB/HPN)

SATURDAY, APRIL 4, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

13) 8:30 – 1:00 P.M. Preparation of ACRS Reports (Open)
(10:30-10:45 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 12.

14) 1:00 – 1:15 P.M. Miscellaneous (Open) (MVB/EMH)
Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

- During the days of the meeting, phone number 301-415-7360 should be used in order to access anyone in the ACRS Office.

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

- Thirty five (35) hard copies and one (1) electronic copy of the presentation materials should be provided to the ACRS in advance of the briefing.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

April 2-4, 2009

PLEASE PRINT

TODAY'S DATE: April 2, 2009

	NAME	NRC ORGANIZATION
1	Donnie J. Ashley	NRR/DLR/
2	John Daily	NRR/DLR
3	EDWARD SMITH	NRR/DSS
4	Seung Min	NRR/DLR
5	SAMSON LEE	NRR/DLR
6	Bill Rogers	NRR/DLR
7	Jim Davis	NRR/DLR
8	DAVID ALLEY	NRR/DLR
9	Duc Nguyen	NRR/DLR
10	Emma Wong	NRR/DCI/CSGB
11	~~Jan~~ ~~Hang~~	NRR/DCR/REB2
12	Roy MATHEW	NRR/DE
13	SHEILA RAY	NRR/DE
14	Glenn Meyer	NRC Region I
15	JAMES MEDOFF	NRC NRR/DSR
16	ALBERT WONG	NRC NRR/DLR
17	Jerry Dozier	NRC NRR/DLR/RER/
18	Raj Anand	NRC/NRR/DLR
19	CLIFF DOUTT	NRC/NRR/DLR/REB2
20	Andrew Johnson	NRR/DCI/CSGB
21	Chuangyen Yang	NRC NRR/DLR
22	Edward Andruszkiewicz	NRC/NRR/DCI
23	JAY COLLINS	NRC/NRR/DCI
24	Matthew A. McLeod	NRR/DCI
25	ANDREW PRINARIS	NRR/DLR
26	Jim Shaffern	ACRS/DRMCP
27	Barrett Brady	NRR/DLR
28	BRIAN HOLIAN	NRR/DLR

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

April 2-4, 2009

PLEASE PRINT

TODAY'S DATE: April 2, 2009

	NAME	NRC ORGANIZATION
1	Liliana Ramadan	NRC/RES/DE
2	John Burke	NRC/RES/DE
3	GARY WANG	NRC/RES/DE
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

April 2-4, 2009

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TODAY'S DATE: April 2, 2009

	NAME	AFFILIATION
1	JACK STRINGFELLOW	SOUTHERN NUCLEAR
2	Cary Martin	Southern Nuclear
3	TOM TYNAN	SOUTHERN NUCLEAR
4	TODD YOUNGBLOOD	SOUTHERN NUCLEAR
5	Jon Hornbuckle	Southern Nuclear
6	Mark Ajluni	Southern Nuclear
7	Lee Mansfield	Southern Nuclear
8	Louis Bohn	Southern Nuclear
9	Greg Halnon	FirstEnergy
10	CHRIS WILSON	EXELON
11	Joe Ruether	Xcel Energy
12	MIKE GALLAGHER	EXECON
13	STAFFORD M. MISHOE	SAVANNAH RIVER NAT'L LABORATORY
14	Spyros Traiforos	LINK
15	GORDON CLEFTON	NEI
16	Melanie Wong	NRC
17	Ed Miller	NRC
18	JIM RILEY	NEI
19	~~David Desaulniers~~	
20	Glenn Tracy	NRC
21	Dan Hughes	NIST CNR
22	J. M. ROWE	NIST
23	DAVID BROWN	NIST
24	DANIEL FLYNN	NIST
25	Sy Weiss	NIST
26	John Crosby	EnviroTech Sensors, Inc
27	Paul BRAND	NIST
28	Robert Williams	NIST

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

April 2-4, 2009

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TODAY'S DATE: April 2, 2009

	NAME	AFFILIATION
1	George Wilson	NRC
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

April 2-4, 2009

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TODAY'S DATE: April 3, 2009

	NAME	NRC ORGANIZATION
1	DONALD DUBE	NRO/DSRA
2	GARETH PARRY	NRR/DRA
3	Mary Drouin	NRC/RES
4	Bob TJADER	NRC/NRO/CTSB
5	Lynn Mrowca	NRO/SPLA
6	Jeff Ciocco	NRO/DNRL
7	HIEN LE	NRO/AGIP
8	Jocelyn Mitchell	RES/DSA
9	Theresa Clark	NRO/DSRA/SPLA
10	John Manning	NRO/DSRA
11	Charles Ader	NRO/DSRA
12	~~Jim~~	
13	Taesuk Hwang	NRR (foreign assignee)
14	~~Hash~~ Phan	NRO/DSRA
15	DON HELTON	RES/DRA
16	John Lai	NRO/DSRA
17	KRISTY BUCHOLTZ	NRR/DIRS/ITSB
18	Tina Ghosh	NRR/DRA
19	Paul Gallon	NRO/NMIP.
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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
561st FULL COMMITTEE MEETING

April 2-4, 2009

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TODAY'S DATE: April 3, 2009

NAME	AFFILIATION
SPYROS TRAIFOROS	LINK
Patricia Campbell	GE Hitachi
Keiko Chitose	MNES
JOSEPH TAPIA	MNES
Russell Bywater	MNES
RICHARD BRABEC	LUMINANT
Bill McCONAGHY	TOSHIBA/TANE
Futoshi Tanaka	MHI
Eugene Hughes	ETRANCO
C. Keith Paulson	MNES
Andrew Johnson	MNES
Shinji Otani	MNES
BOB LUTZ	WESTINGHOUSE
Don Woodlan	Luminant
STANLEY LEWINSON	AREVA NP
DOUG WALTERS	NEI
CHRIS WILSON	EXELON
Biff Bradley	NEI
Hossein Hamzehee	NRC/NRO
CAREY BICKETT	NRC/NRO
Steve FRANTZ	MHI
Etsuro Saji	MHI
Jim Chapman	Scientech
Vesna DIMITRIJEVIC	AREVA NP
RUSS BELL	NEI
Patrick O'Regan	EPRI

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

March 10, 2009

AGENDA
561st ACRS MEETING
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5.1) Remarks by the Subcommittee Chairman

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SATURDAY, APRIL 4, 2009, CONFERENCE ROOM T-2B3, TWO WHITE FLINT NORTH, ROCKVILLE, MARYLAND

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(10:30-10:45 A.M. BREAK) Continue discussion of the proposed ACRS reports listed under Item 12.

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Discussion of matters related to the conduct of Committee activities and specific issues that were not completed during previous meetings, as time and availability of information permit.

NOTES:

- During the days of the meeting, phone number 301-415-7360 should be used in order to access anyone in the ACRS Office.

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

- Thirty five (35) hard copies and one (1) electronic copy of the presentation materials should be provided to the ACRS in advance of the briefing.

Appendix V

LIST OF DOCUMENTS FROM THE
561ST ACRS MEETING APRIL 2-4, 2009

Agenda Item 2:
License Renewal Application and Final Safety Evaluation Report (SER) for the Vogtle Nuclear Plant

1. Proposed Schedule
2. Status Report
3. Attachment:
   - Slides from meeting on November 5, 2008
   - Subcommittee Meeting Minutes

Agenda Item 3:
Digital Instrumentation and Control (I&C) Interim Staff Guidances (ISGs)

4. Table of Contents
5. Proposed Agenda
6. Status Report
7. References:
   - ISG - 5 Rev 1, Highly-Integrated Control Room - Human Factors, Rev.1
   - Draft ISG - 6, DI&C Licensing Process

Agenda Item 4:
License Renewal Application and Final Safety Evaluation Report for the National Institute of Standards and Technology (NIST) Reactor

8. Proposed Schedule
9. Status Report

Agenda Item 5:
Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

10. Table of Contents
11. Proposed Agenda
12. Status Report
13. References
    - Final Draft Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants", March 2009

Appendix IV

LIST OF DOCUMENTS FROM THE
561ST ACRS MEETING APRIL 2-4, 2009

Agenda Item 8:
Risk Metrics for New Light-Water Reactor Risk-Informed Applications

14. Proposed Schedule
15. Status Report
16. Attachments

- Memorandum to C. Santos, Chief, Reactor Safety Branch – A, ACRS from L. Mrowca, Chief, PRA Licensing, Operating & Maintenance Support Branch 1, NRO, dated February 26, 2009, Subject: Advisory Committee on Reactor Safeguards April 2009 Tentative Meeting Agenda Item
- White Paper on Options for Risk Metrics for New Reactors

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Protecting People and the Environment

Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee Vogtle Electric Generating Plant (VEGP)

Safety Evaluation Report (SER)

April 2, 2009

Donnie J. Ashley, Project Manager
Office of Nuclear Reactor Regulation

Introduction

- Overview of VEGP license renewal review
- License Renewal Audit and Inspection
- SER Section 2: Scoping and Screening review results
- SER Section 3: Aging Management review results
- SER Section 4: Time-Limited Aging Analyses (TLAAs)

Overview

- License Renewal Application (LRA) submitted June 2007

Recap of November 2008 Subcommittee Meeting

- 87 RAI items issued
- 173 Audit Questions
- 40 Commitments
- Additional Components Brought Into Scope
- Draft SER issued November 19, 2008
- Final SER issued March 13, 2009
- No Open Items (OIs)
- No Confirmatory Items
- Three (3) License Conditions [(1)]

Overview

Recap of November 2008 Subcommittee Meeting (cond't)

- Material Condition in Containment
  - Identified during June Regional Inspection
  - Region has evaluated past inspections and will continue to monitor
- Boral
  - Questions about adequacy of AMP
  - Telecon with Applicant and Tech Staff
  - Revised AMP commitment
- Water in Cable Pull-Box
  - Applicant revised monitoring program
  - Region will continue to monitor

Overview (cond't)

Subsequent to sub-committee meeting

- 7 additional RAIs issued on Boral
  - Resulted in a revised commitment to provide for inspection of panels in addition to water chemistry program.

- 94 total RAIs

- Applicant provided update on Commitments
  - Added one new commitment on debris screens.

Audits and Inspections

- Scoping and Screening Methodology Audit
  9/17 – 9/21, 2007
- Aging Management Program (AMP) Audit
  10/15 – 10/19, 2007
- Aging Management Review (AMR) Audit
  12/9 – 12/14, 2007
- Region II Inspection (Scoping and Screening & AMP)
  5/19/ – 06/06, 2008

Audit and Review

- Audit Summary (ADAMS Accession No. ML080430373)
  - Publicly Available, Issued on September 30, 2008
  - Audit Summary Includes :
    - Audit and Review Results
    - Audit and Review Q&A Database
    - List of Documents Reviewed by the Audit and Review Team

Regional Inspection

- Two Weeks on Site
- AMP inspection May 19 – June 6, 2008
- 10 CFR 54.2(a)
- License renewal chapter - MC 2516
- License renewal inspection procedure IP 71002

Regional Inspection

- Inspectors identified enhancements
  - Manhole flooding with Medium Voltage Non-Safety Related Cable
  - Condition inside containments
- Region II will follow up on these issues during a future IP 71003 inspection

Regional Inspection

The inspection team concluded that the scoping and screening of non-safety related systems, structures, and components, was implemented as required by the rule and the aging management portions of the license renewal activities were conducted as described in the application.

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SER Section 2: Structures and Components Subject to Aging Management Review

<u>Section 2.1</u> Scoping and Screening Methodology

<u>Section 2.2</u> Plant-Level Scoping Results

<u>Section 2.3</u> Mechanical Systems Results

<u>Section 2.4</u> Structures Results

<u>Section 2.5</u> Electrical and I&C Control Systems

Section 2 – Conclusion

Based on its review of the LRA, the onsite audit results, and additional information submitted as the result of RAIs, the staff concluded that:

- The applicant's scoping and screening methodology meets the requirements of 10 CFR 54.4 and 54.21(a)(1), and
- That the applicant adequately identified those SSCs within the scope of license renewal in accordance with 10 CFR 54.4(a), and those SCs subject to an AMR in accordance with 10 CFR 54.21(a)(1)

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Section 3: Aging Management Review Results

Aging Management Programs (AMPs)

- 38 AMPs
  - 14 are NEW programs
  - 24 are EXISTING programs
- 20 programs with exceptions and/or enhancements
- 11 are plant specific
- 7 are consistent with GALL

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Section 3 – Conclusion

Based on its review of the LRA and additional information submitted as the result of RAIs, the staff concluded that:

- Aging effects will be managed so that the intended functions will be maintained consistent with the CLB for the period of extended operation, per 10 CFR 54.21(a)(3), and
- That activities authorized by the renewed license would continue to be conducted in accordance with the CLB, and any changes made to the CLB in order to comply with 10 CFR 54.21(a)(3), would be in accordance with the NRC's regulations

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SER Section 4: Time-Limited Aging Analyses

- 4.1 TLAA Process
- 4.2 Reactor Vessel Neutron Embrittlement
- 4.3 Metal Fatigue
- 4.4 Environmental Qualification of Electrical Equipment
- 4.5 Concrete Containment Tendon Prestress
- 4.6 Containment Liner Plate Metal Containments and Penetration Fatigue
- 4.7 Other Plant Specific TLAA

4.7 Other Plant Specific TLAAs

- 4.7.1 Leak Before Break Analysis
- 4.7.2 Fuel Oil Storage Tank Corrosion Allowance
- 4.7.3 Steam Generator Tube, Loss of Material
- 4.7.4 Cold Overpressure Protection System
- 4.7.5 Underclad Cracking of Reactor Pressure Vessel

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Section 4 - Conclusion

Based on its review of the LRA and additional information submitted as the result of RAIs, the staff concluded that the applicant provided an adequate list of TLAAs, per 10 CFR 54.3 and that the:

- TLAAs will remain valid for the period of extended operation, per 10 CFR 54.21(c)(1)(i)
- TLAAs have been projected to the end of the period of extended operation, per 10 CFR 54.21(c)(1)(ii)
- Aging effects will be managed for the period of extended operation, per 10 CFR 54.21(c)(1)(iii)

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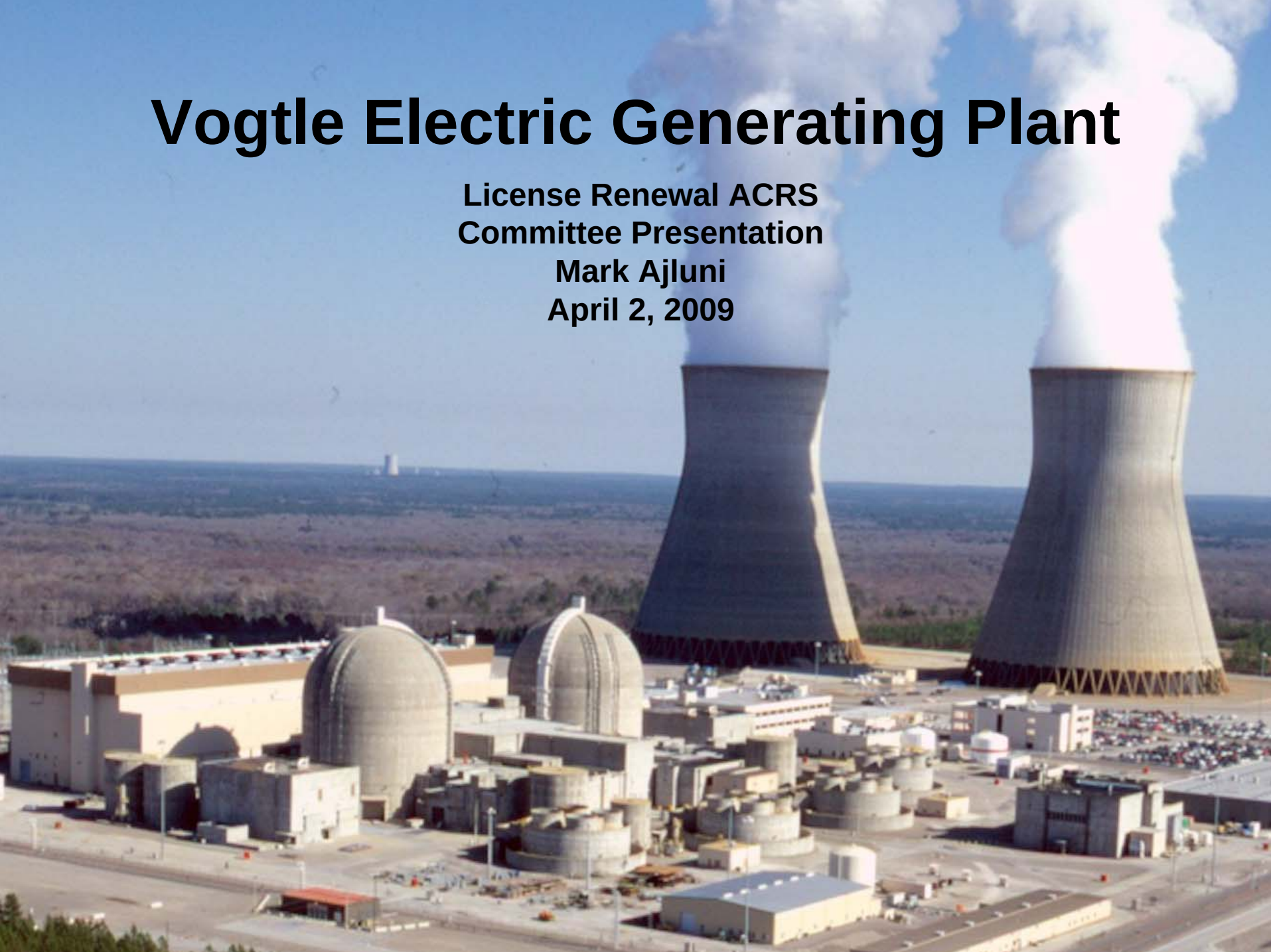
Overall Conclusion

The staff has concluded there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB, and that any changes made to the VEGP CLB in order to comply with 10 CFR 54.29(a) are in accordance with the Act and the Commission's regulations.

Vogtle Electric Generating Plant

License Renewal ACRS
Committee Presentation
Mark Ajluni
April 2, 2009



Agenda

- Introductions
- VEGP Site Description
- Project Key Points
- Open Items
- Subcommittee Follow-Up Items
- Summation and Questions

VEGP Representatives

- Tom Tynan, Site Vice President
- Lee Mansfield, Engineering Support Manager
- Mark Ajluni, Licensing Manager

VEGP Site Description

- Westinghouse (NSSS), Bechtel (AE)
- Two 4 Loop PWR Units
  - 3625 MWt
  - 1250 MWe
- Ultimate Heat Sink – NSCW Forced Draft Cooling Towers and Basins
- Turbine Cycle Cooling Provided By Natural Draft Towers

VEGP License Renewal Project Key Points

- 10 CFR 54.17(c) Exemption
  - NRC granted VEGP an exemption to 10 CFR 54.17(c) to submit Vogtle Unit 2 License Renewal Application prior to reaching 20 years remaining on the operating license
    - Basis: Unit 2 is the same design and construction as Unit 1

VEGP License Renewal Project Key Points

- As a result of the aging management reviews, VEGP credited the following:
  - 38 Aging Management Programs
    - 9 existing programs with no change
    - 15 existing programs with enhancements
    - 14 new programs
  - 27 of the 38 are GALL Programs
    - Only minor exceptions
    - 11 plant specific programs incorporated GALL attributes where possible

Open Item Resolution

- The VEGP Final SER for License Renewal has no Open or Confirmatory Items

Subcommittee Follow-Up Items
Containment Residue

- Region II Site Inspection performed walkdown of Boric Acid Corrosion Control Program during 1R14 (April 2008)
- Inspection concluded BACC Program would adequately manage boric acid corrosion
- However inspector noted non-boric acid deposits from NSCW leakage that could mask boric acid corrosion

Subcommittee Follow-Up Items
Containment Residue

Subcommittee Follow-Up Items
CNMT Residue Corrective Actions

- BACC Program procedure changes, training, and enhanced communication have been implemented to heighten awareness.
- Aggressive inspection, cleaning and repainting program is in progress – at power and during refueling outages.
- Tools developed to promptly identify and address leakage in containment.

Subcommittee Follow-Up Items
Containment Residue

016
NSCW
CNMT ELR
OUTLET
TEST CONN
A

Subcommittee Follow-Up Items
Containment Residue

- The valve on the preceding slide was discussed at the Subcommittee meeting
- The corrosion visible in the picture had been evaluated as minor surface corrosion that did not require remediation
- VEGP inspected and cleaned the valve. The initial evaluation was confirmed as correct.

Subcommittee Follow-Up Items
Water in Pullboxes

- Medium Voltage Cables
  - In scope medium voltage cables at VEGP are located in tunnels and not subject to submergence, with one exception
    - Non-safety related 4kV feeders to high voltage switchyard switch house
  - Region Inspection found water in pull box near switch house
  - Corrective action – accelerated implementation of the license renewal aging management program inspections
    - Quarterly inspection (later increased to monthly based on first two inspections)
    - Trending of results
  - Design changes initiated to add sump pumps to two boxes. Planned for installation in 2010.

Subcommittee Follow-Up Items
Aging Management of Boral

- VEGP Unit 1 spent fuel racks contain Boral™ neutron-absorbing panels. Unit 2 racks do not use Boral™.
- Our strategy monitors spent fuel pool aluminum concentration and monitors industry operating experience.
- To date, industry operating experience has not shown evidence of loss of neutron absorption in Boral™.
- SNC has committed to perform a baseline and a follow-up inspection to confirm the neutron-absorbing capacity of the Boral™ panels.

Summary

- Experienced team created a high quality application
- Extensive use made of GALL
- Thorough audits and inspection of the application and programs
- VEGP responsive to NRC throughout review
- No Open or Confirmatory Items
- VEGP is confident that we are prepared to manage aging and continue to operate safely beyond 40 years

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License Conditions (1)

- The first license condition requires the applicant to include the UFSAR supplement required by 10 CFR 54.21(d) in the next UFSAR update, as required by 10 CFR 50.71(e), following the issuance of the renewed license.

- The second license condition requires future activities identified in the UFSAR supplement to be completed prior to the period of extended operation.

- The third license condition requires that all capsules in the reactor vessel that are removed and tested meet the requirements of American Society for Testing and Materials (ASTM) E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule insertion and withdrawal schedule, including use of spare capsules, must be approved by the staff prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the staff, as required by 10 CFR Part 50, Appendix H.

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Digital Instrumentation Control Interim Staff Guidance on Highly Integrated Control Room - Human Factors, Licensing Process Issues, and Related Matters

April 2, 2009

Steven Arndt
Office of Nuclear Reactor Regulation

David Desaulniers
Office of New Reactors

G. Edward Miller
Office of Nuclear Reactor Regulation

Agenda

- Background on Steering Committee
  - Digital I&C Project Products
  - Path Forward
- Overview of ISG-5, Revision 1
  - Manual Operator Action Background
  - Overview of ISG for crediting manual operator actions
- Overview of ISG-6
  - Process Overview
  - Format of ISG-6
  - Tiers of Complexity
  - Phases of Process
  - Areas of Review
  - Path Forward

Background on Steering Committee

- Digital I&C Steering Committee formed January 2007
- 7 Task Working Groups formed to address specific issues
  - TWG-1 Cyber Security
  - TWG-2 Diversity and Defense in Depth (D3)
  - TWG-3 Risk-Informing Digital I&C
  - TWG-4 Highly-Integrated Control Room – Communications
  - TWG-5 Highly-Integrated Control Room – Human Factors
  - TWG-6 Licensing Process
  - TWG-7 Fuel Cycle Facilities
- Industry established counterpart groups
- Over 100 public meetings to define, discuss and resolve issues

Status of Digital I&C Products

- TWG-1: Cyber Security - ISG-1 Issued 12/2007
  - ACRS Review: Letter dated April 29, 2008
  - Next Steps: Update SRP and RG 1.152 following rulemaking and RG 5.71

- TWG-2: Diversity and Defense in Depth (D3) - ISG-2 Issued 9/2007
  - Industry Concern: Credit defensive measures and component level actuation
  - ACRS Review: Letter dated October 16, 2007
    - Recommends Process to Evaluate < 30 Minute Operator Action
  - Next Steps: Complete NUREG on diversity attributes, update SRP

Status of Digital I&C Products (cont'd)

- TWG-3: Risk Informing Digital I&C - ISG-3 Issued 8/2008
  - Industry Concern: Risk informing guidance on D3 and policy on diverse actuation systems
  - ACRS Review: Letter dated April 29, 2008
    - Recommends emphasize failure modes vs. sensitivity studies
    - ISG-3 was revised to incorporate recommendations
  - Next Steps: Methods for obtaining risk insights and risk informing digital I&C to be addressed in 5-year research plan

- TWG-4: Highly Integrated Control Room Communications ISG-4 Issued 9/2007
  - ACRS Review: Letter dated October 16, 2007
  - Next Steps: Update SRP, RG 1.152, and IEEE 7.4.3.2

Status of Digital I&C Products (cont'd)

- TWG-5: Highly Integrated Control Room Human Factors ISGs issued 9/2007 and 11/2008
  - ACRS Review: Letter dated October 16, 2007
    - Present ISG on Manual Action today
  - Next Steps: Rulemaking on Safety Parameter Display System, and update NUREG, Reg. Guide, and SRP

- TWG-6: Licensing Process - ISG-6 under development, inspection procedure issued 10/2008, audit procedure issued 12/2008
  - Industry Concern: Level of staff review and slow progress on ISG
  - ACRS Review: Letter Dated April 29, 2008
    - Present update today
  - Next Steps: Bi-weekly public meetings, workshop on licensing experiences in Spring 2009, add cyber security, issue final ISG

Status of Digital I&C Products (Cont.)

- TWG –7: Fuel Cycle Facilities - ISG-7 under development
  - ACRS Review: Request Review End of Summer
  - Next Steps: Issue ISG, update and issue appropriate NUREGs
- Ongoing Work
  - Updates to Regulatory Documents
  - Operational Program Updates

Path Forward

- Complete ISG for Licensing and Fuel Cycle Facilities
- Update Regulatory Documents
- Address Operational Issues
- Continue to use ISGs in Ongoing Reviews
- International Cooperation
  - MDEP
  - Bilateral Work
  - COMPSIS
  - IAEA and Other Interactions
- Ongoing Research Activities

ISG-05, Revision 1

- Manual operator actions
- Computer-based procedures
- Minimum inventory

Background

ISG-02, Interim Staff Guidance on Diversity and Defense-in-Depth Issues, September 26, 2007

- Provided acceptable methods for implementing diversity and defense-in-depth (D3) in digital I&C system designs
- Clarified use of operator action as a diverse defensive measure and established corresponding operator action times

Background

ISG-02 states:

"Manual operator actions may be credited for responding to events in which the protective action subject to a CCF is not required for at least the first 30 minutes and the plant response is bounded by BTP 7-19 recommended acceptance criteria."

Background

- Industry sought flexibility and guidance to credit manual operator actions in less than 30 minutes
- ACRS letter (10/16/07) recommended development of an alternative process to the 30-minute criterion
- Scope of TWG-5 action plan expanded to develop guidance for crediting manual operator actions
- Public interactions between TWG-5 and industry counterparts
- Industry developed white paper methodology for crediting manual operator actions
- Staff considered and incorporated, as appropriate, white paper methods in developing an amendment to ISG-05

ISG-05 (Revision 1)

Manual Operator Action ISG

- Scope
- Staff Position
- 4-Phase Methodology

Scope

- Manual actions credited in D3 analyses for coping with abnormal operational occurrences and postulated accidents (AOO/PAs) concurrent with software CCF of the digital protection system
- New and existing reactors

Staff Position

Credited actions should be:

- Included in emergency operating procedures (EOPs)
- Executed from within the main control room
- Demonstrated to be feasible and reliable
- Addressed in the human factors engineering (HFE) program consistent with NUREG-0711

Overview of ISG-5

Method

4-Phases:

- Analysis
- Preliminary Validation
- Integrated System Validation
- Long-term Monitoring

Analysis

Objective

- Estimate *time available* and *time required*
- Identify critical assumptions and credible operator errors
- Establish adequate *margin*

Analysis

Method

Time Available

- Use methods and realistic assumptions consistent with BTP 7-19.

Time Required

- Use a documented sequence of actions (from task analysis, EPGs, EOPs)
- Use one of several acceptable methods for developing estimates of time required to perform action sequence

Margin

- Time to recover from credible errors

Analysis

Examples of Acceptable Methods

- Operator interviews and surveys
- Operating experience reviews
- Software models of human behavior, such as task network modeling
- Use of control/display mockups
- Expert panel elicitation
- ANSI/ANS 58.8, *Time Response Design Criteria for Safety-Related Operator Actions (task decomposition)*

Analysis

Review Criteria Topics

- Time required
- Time available
- Use of alarms, controls, and displays
- Use of symptom/function-based EOPs
- Staff size, composition and augmentation
- Level of detail
- Identification of credible operator errors

Preliminary Validation

Objective

- Independent confirmation of analysis results

Applicability

- Only required for those vendors/applicants who are using the 10 CFR Part 52 process

Method

- Use diverse methods that are as realistic as maturity of design allows
- Submit analysis and results for NRC review as part of D3 submittal(s)

Preliminary Validation

Examples of Acceptable Methods

- Tabletop analysis
- Walkthrough/talkthrough analysis
- Software models of human behavior, such as task network modeling
- Use of control/display mockups
- Man-in-the-loop prototype testing
- Real-time validation using part-task simulator

Preliminary Validation

Review Criteria Topics

- Independence from Phase 1
- Validation team qualifications
- Use of two or more methods
- Validation of time required

Preliminary Validation

Results

- Shall be documented in the D3 analysis for NRC review
- Should support high confidence that the time required for manual operator actions will satisfy the success criteria for the integrated system validation

Integrated System Validation

Objective

- Confirm operators are able to perform credited actions in real-time using as-built design

Method

- Use plant-referenced simulator capable of realistically representing AOO/PA with CCF
- Validate time required using both nominal and TS minimum crews
- Accomplish as part of HFE program activities per NUREG-0711

Integrated System Validation

Review Criteria Topics

- Integration with HFE program
- Simulator
- Personnel
- Operational Conditions
- Performance Times

Integrated System Validation

Performance Time Criteria

- For each AOO/PA, the *mean performance times* of the crews is less than or equal to the estimated *time required* derived from the analysis phase.
- For each AOO/PA, the *performance time* for each crew, *including margin* determined in the time required analysis, is less than the analyzed *time available*.

Long-term Monitoring

Objective

- Ensure credited actions remain feasible and reliable

Method

- Design and configuration controls ensure discrepancies from D3 assumptions and constraints are identified and corrected
- Training keeps performance within assumptions of the analysis

Long-term Monitoring

Review Criteria

- A long-term monitoring strategy is capable of tracking performance of the manual operator actions to demonstrate that performance continues to support the associated D3 analysis
- The program is structured such that corrective actions are formal, effective, and timely

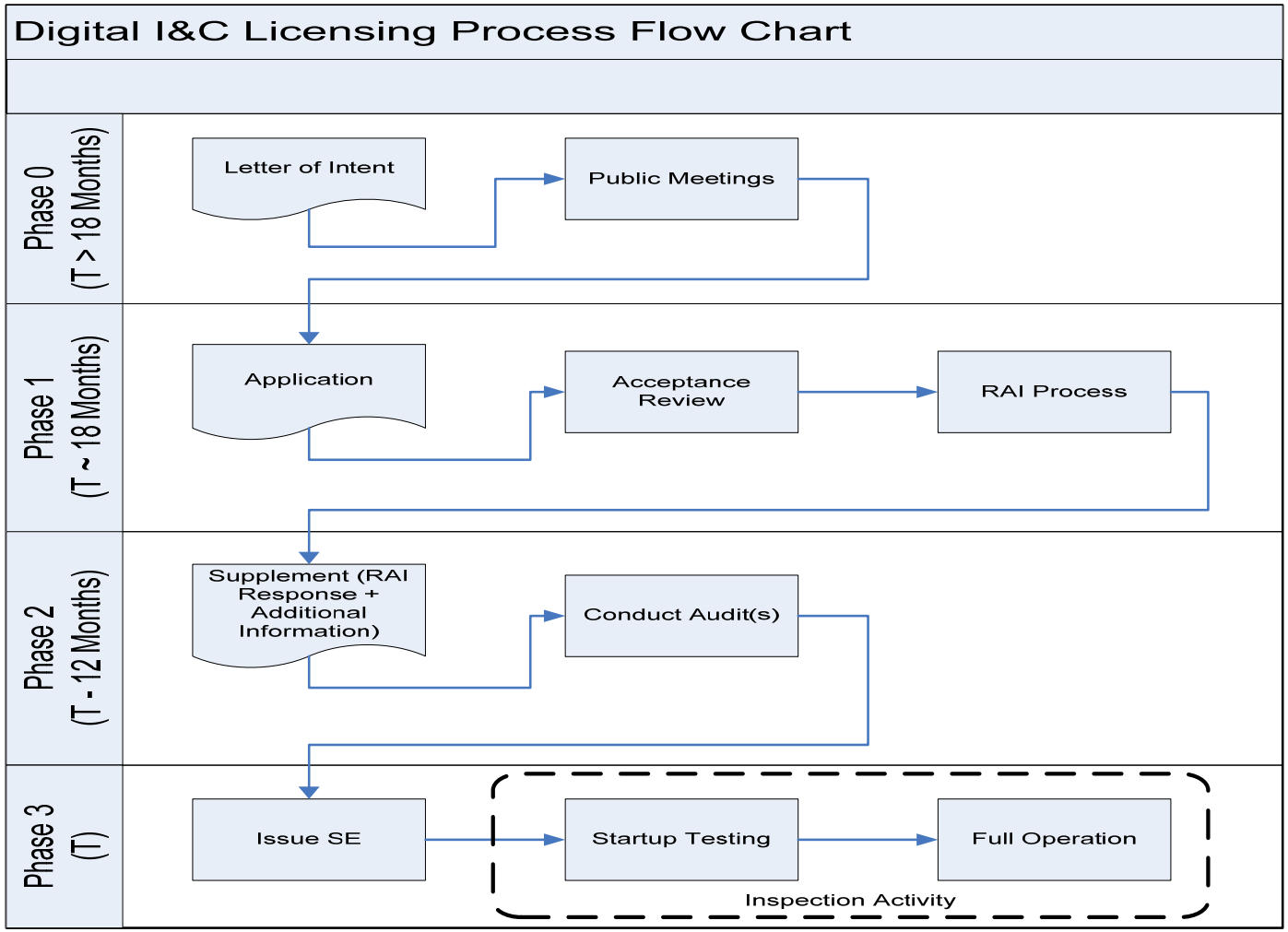
ISG-5 Path Forward

- Develop BTP and revise SRP
- Support future development and revision of ANSI/ANS 58.8, Time Response Design Criteria for Safety-Related Operator Actions

Introduction

- Purpose of ISG-6
  - Refined licensing process
  - Expectations for documentation
  - Knowledge management
- Lessons learned from recent I&C amendment reviews

Process Overview



Format of ISG-6

- Introduction
- Purpose
- Licensing Process
  - Process Overview
  - Pre-Application Meetings
  - Initial Application
  - Continued Review and Audit
  - Implementation and Inspection
  - Review Areas
    - Scope of Review
    - Information to be Provided
    - Regulatory Evaluation
    - Technical Evaluation
    - Conclusion
- Appendices (Example Formats)

Tiers of Review

- Each Tier corresponds to an expected review complexity:
  - Tier 1: Previously approved system, no deviations from topical report, review to focus on plant specific aspects, least review effort expected.
  - Tier 2: Previously approved system, with deviations, moderate review effort expected.
  - Tier 3: Totally new system, extensive review effort expected. Thorough review of all technical areas.

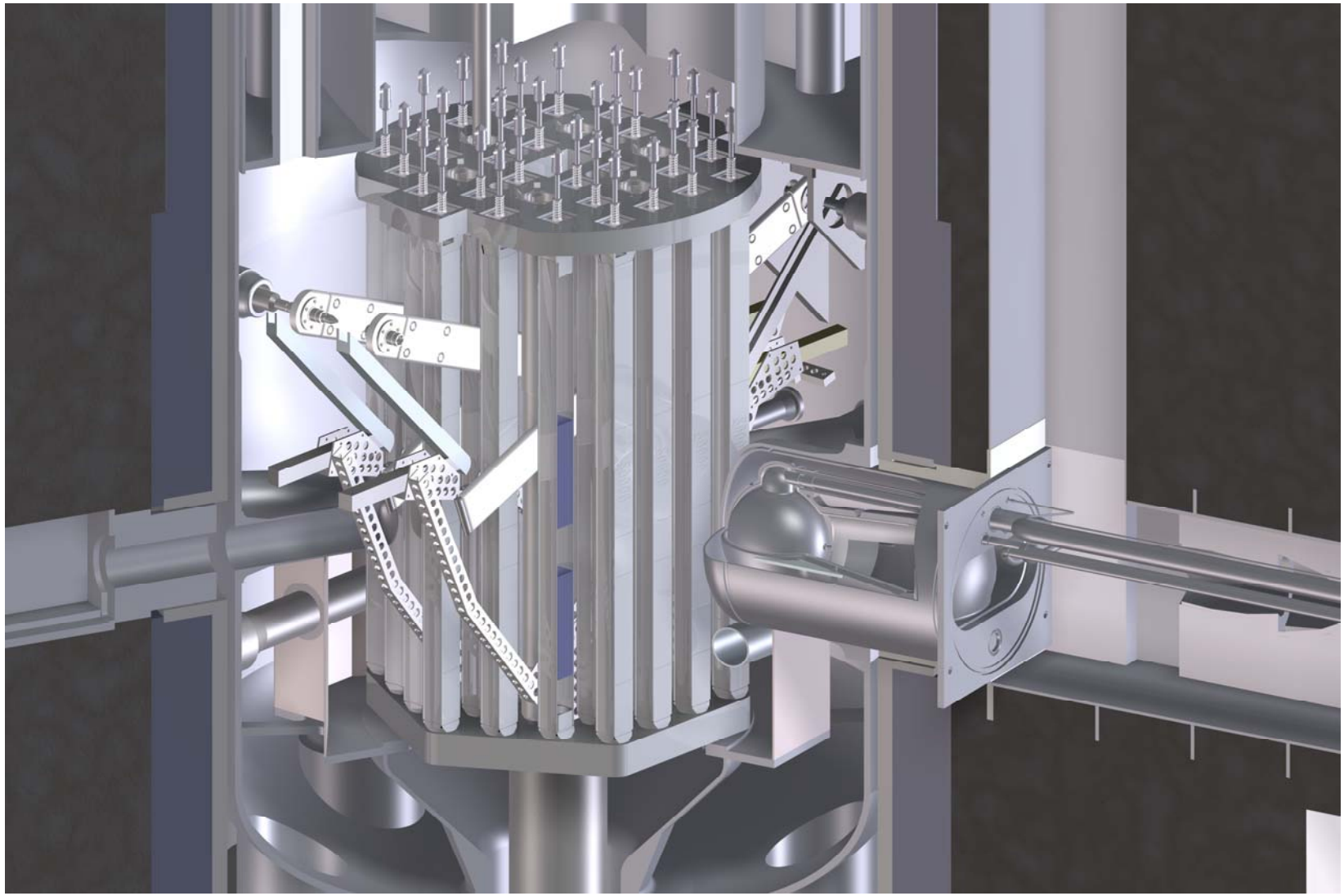
Draft Review Areas

- Working List of Review Areas
  - Defense-in-depth & Diversity
  - Hardware Architecture
  - Hardware Design Process and Quality Control
  - Communications
  - Software Architecture
  - Software Design Process
  - System Qualifications
  - System, Hardware, Software, and Methodology Modifications
  - Technical Specifications
  - IEEE 603 Compliance
  - IEEE 7-4.3.2 Compliance
  - Cyber Security

Path Forward

- Monthly Public Meetings
  - Next meeting is April 28, 2009
- Monthly conference calls on status
- Full Draft of ISG for Public Comment
  - Summer, 2009
- ISG-6 Issued
  - Fall, 2009
- Pilot application encouraged

NIST Center for Neutron Research
ACRS Meeting April 2, 2009



ACRS Meeting April 2, 2009

- **<u>NIST Participants</u>**
- Dr. Robert Dimeo, Director NCNR
- Dr. Michael Rowe, Special Advisor to the NCNR Director
- Dr. Wade Richards, Chief Reactor Operations & Eng.
- Dr. Paul Brand, Chief Reactor Engineering
- Dr. Robert Williams, Section Head Nuclear Analysis
- Mr. David Brown, Supervisor Health Physics
- Mr. Daniel Flynn, Senior Reactor Operator
- Mr. John Crosby, Enviro Tech Sensors Inc.

ACRS Meeting April 2, 2009

- Description of NCNR
- Description of the NBSR
- Licensing Basis Accidents
- Subcommittee Follow-up Items

ACRS Meeting April 2, 2009

Description of the NCNR

The NIST Center for Neutron Research
A National User Facility

The mission of the NIST Center for Neutron Research is to assure the availability of neutron measurement capabilities to meet the needs of U.S. researchers from industry, university and other Government agencies.

- 23 instruments with access based on technical merit
- Highly interdisciplinary: basic/applied materials science & fundamental physics
- More than 2200 research participants per year
- Over 300 scientific publications per year
- Numerous partnerships with other agencies, industry, and academia (e.g. NSF, ExxonMobil, FDA, Smithsonian, Johns Hopkins, UMD,…)

Largest user program and highest productivity of any neutron facility in the US [1]

[1] The American Physical Society: *Access to Major International X-Ray and Neutron Facilities*, November 2008.

The National Context

"The NIST facility is the only U.S. facility which currently provides a broad range of world-class capability." [1]

"…the NIST Center for Neutron Research (NCNR) currently has the largest number of users in the United States, largely because of its modern suite of cold neutron instruments." [2]

"The highest priority for federal investments in neutron scattering is to fully exploit the best U.S. neutron source capabilities...for the benefit of the broadest scientific community." [1]

"To improve access and to enable the user community to grow it is critically important to increase the number of beamlines and instruments at major facilities in the US." [2]

[2] The American Physical Society: *Access to Major International X-Ray and Neutron Facilities*, November 2008.

[1] The Office of Science and Technology Policy Interagency Working Group on Neutron Science: *Report on the Status and Needs of Major Neutron Scattering Facilities and Instruments in the United States*, June 2002.

Strong Management Support
NIST and DoC

NIST and DoC leadership have been extremely supportive of and remain committed to the safe, effective, and reliable operation of the NIST Reactor as a critical component of the NIST measurement mission.

<u>Past</u>

- Upgrade from 10 MW to 20 MW (1984)
- Cold neutron source/guide hall (1994)
- Upgraded cold source installed (2002)
- Cooling tower w/plume abatement installed (2002)
- Initiative to expand access by supporting more instruments and developing new capabilities (2004)

<u>Present/future</u>

- NCNR Expansion and Reliability Enhancements Project (2007-2011)

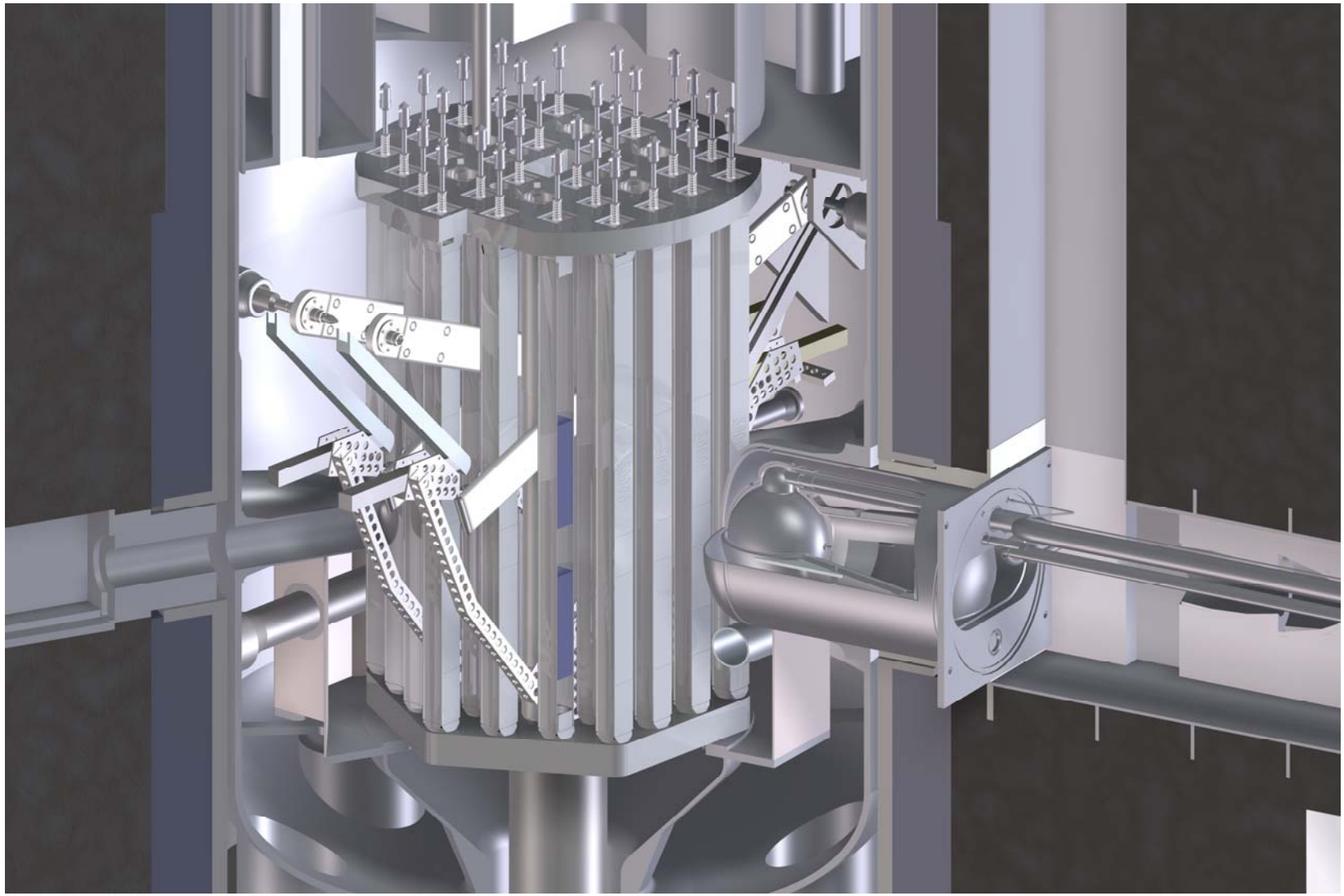
ACRS Meeting April 2, 2009

Description of the NBSR

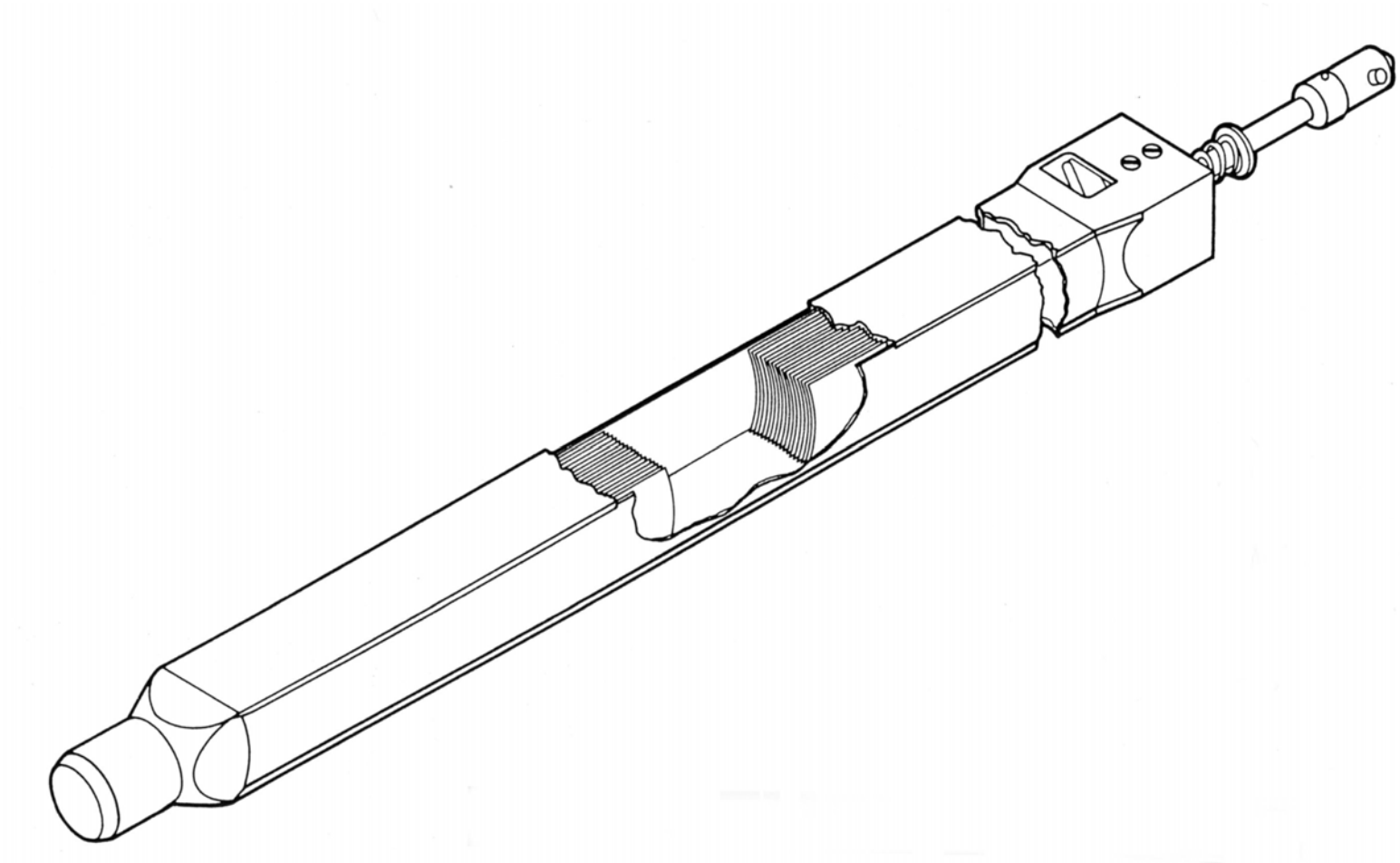
ACRS Meeting April 2, 2009

- The National Bureau of Standards Reactor (NBSR) is a heavy water moderated and cooled, enriched fuel, tank type reactor designed to operate at 20Mw.
- It is a custom designed variation of the Argonne CP-5 class reactor.

NIST Center for Neutron Research



ACRS Meeting April 2, 2009



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- The NBSR includes many inherent, passive safety features
- The prompt neutron lifetime is long due to the heavy water
- The reactivity coefficients of void and temperature are negative
- Reactor operates in a low temperature, unpressurized condition and has no large stored energy content.

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Licensing Basis Accidents

Accidents Analyzed

- **Startup Accident**
  - 100 kW → 130% at $\Delta\rho/\Delta t = 5x10^{-4}\ \Delta\rho/s$
- **Maximum Reactivity Insertion**
  - Experiment removed $\Delta\rho$ =.5% in .5 sec
- **Loss of Coolant**
  - Inlet pipe break in process room
  - Offsite tritium doses within 10 CFR 20 limits
- **Misloading of Fuel**
  - Fresh element in all locations

Accidents Analyzed II

- **Loss of Flow**
  1. Loss of Offsite Power *
  2. Single Pump (1 of 3) Seizure *
  3. Throttling of Inlet Valves to either Plenum
  4. Spurious signal closing DWV-19 (Hot Leg)
  5. Loss of Both Shutdown Coolant Pumps
- **Maximum Hypothetical Accident**

  - Postulates complete melting of fuel element

  - Satisfies 10 CFR 100 limits

* Open Item

Open Item

- Flow coastdown data out-of-date.
- New data will be measured this week.
- BNL will analyze loss of flow accidents again with updated coastdown data.

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Subcommittee Follow-up Items

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- Discuss the methodology used to monitor the groundwater at the reactor site for contamination by releases of radioactive material from the facility.

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- Provide a discussion of seismically-induced damage to any structures and components that have been installed since the analysis in 1966 whose failure could impact proper operation of safety-related structures and components.

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- The most recent revision of the proposed Technical Specifications contains some surveillance requirements that are less conservative than those specified in the current TSs. Provide a discussion that explains why the recommended surveillance requirements contained in ANS 15.1-2007 are inappropriate or overly conservative for the NBSR. Provide a discussion that explains how the decrease in conservatism maintains the current level of safety.

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- Provide a detailed step-by-step explanation of the experiment review, approval, and implementation process. Include discussions of the applicable requirements specified in the proposed Tss and administrative requirements that ensure no experiment will have an adverse impact on reactor safety or the health and safety of the public and staff personnel.

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- Provide a discussion and analysis of the adequacy of natural circulation cooling in the event that DWV-19 is isolated during extended full-power operation. Include a discussion of the heat sinks, flow paths, peak vessel temperature, and ability of the pressure relief valve to perform its intended function. Also discuss any actions or preventative measures to mitigate the consequences of such a occurrence or measures taken to prevent it altogether.

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- The moderator temperature coefficient and moderator void coefficient appeared to be inconsistent with each other. Discuss how these coefficients were determined including methods of calculation. Discuss whether these coefficients necessarily need to be consistent given the methods of calculation. Discuss how these coefficients represent the most limiting conditions in the coolant and/or how the values chosen for these coefficients provide adequate conservatism.

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- Discuss the storage and disposal of Class B and Class C waste that may be generated at the NBSR during the period of the renewed license.

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- Proposed TS 3.9.1,"Fuel Storage", specified that "all fuel elements shall be stored and handled in a geometry such that the calculated k_{eff} shall not exceed 0.90 under optimum conditions of water moderation and reflection". Provide a summary and discussion of this calculation for storage in the spent fuel pool.

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- Clarify whether the confinement building design is based on loading from 100-mph sustained winds or a 100-mph wind gust.

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- Discuss the derivation of the 50-year and 100-year numerical scaling factors for wind speeds. If Caribbean hurricane wind data was used to estimate the 100-year maximum wind gust speed at the NBSR site, provide a discussion of the relevance of that date to winds at the NBSR site. Otherwise, provide a discussion of the wind data used to estimate the 100-year maximum wind gust speed for the NBSR site.

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- In March 2008, storms in the Washington DC metro area generated wind gust speeds of 74 mph and 60 mph measured at Regan National Airport and 66 mph measured at Dulles International Airport. Discuss how this storm data affects the projection of the 100-year maximum wind gust speed.

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- Discuss the technical basis for the applied additional rain-on-snow loading including historical data to justify the assumed 50% rain fraction.

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The End

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Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee

National Institute of Standards and Technology National Bureau of Standards Test Reactor License Renewal

April 2, 2009

William B. Kennedy, Project Manager
Office of Nuclear Reactor Regulation

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Introduction

- National Institute of Standards and Technology (NIST) National Bureau of Standards Reactor (NBSR) License Renewal
- Topics:
  - Licensing History
  - Staff Review Criteria
  - Overview of the Safety Evaluation Report (SER)
  - Resolution of Open Items in the draft SER
  - Additional Open Item in the final SER

Licensing History

- 1961
  - The National Bureau of Standards applied for a construction permit and operating license for a 10 Megawatt thermal (MW(t)) heavy-water-cooled-and-moderated reactor
- 1963
  - The Atomic Energy Commission (AEC) issued a construction permit
- 1967
  - The AEC issued Provisional Operating License No. TR-5

Licensing History

- 1970
  - The AEC issued Facility License No. TR-5 with a term of 15 years
- 1980
  - The National Bureau of Standards applied for a 20-year renewal and an increase in the maximum licensed power level to 20 MW(t)
- 1984
  - The NRC issued the renewed license at the increased power level for a period of 20 years
- NIST filed an application for license renewal April 9, 2004

Regulatory Review Criteria

- In accordance with Section 104 of the Atomic Energy Act of 1954, as amended (the Act), the NRC must “impose the minimum amount of regulation consistent with its obligations under this Act...”
- 10 CFR Part 54 does not apply to license renewal for non-power reactors

Regulatory Review Criteria

- Non-power reactor license renewal is primarily conducted in accordance with 10 CFR Parts 20, 50, 51, and 100 (in part, for test reactors)
  - Part 100 guidelines for radiological consequences of accidents apply to test reactors
  - Part 100, Appendix A, “Seismic and Geologic Siting Criteria for Nuclear Power Plants,” does not apply

Staff Review Guidance

- NUREG-1537, Part II, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria,” dated February 1996, provides the staff with review criteria, including:
  - NUREGs pertinent to special areas of the review, e.g., emergency planning
  - Regulatory Guides, division 2
  - American National Standards Institute/American Nuclear Society ANSI/ANS-15 series standards

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Areas of Review

- Facility siting criteria
- Structures, systems, and components
- Reactor characteristics
- Electrical power systems
- Experiment program
- Radiation protection
- Accident analyses
- Technical specifications
- Prior use of components

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Safety Conclusions

Based on its safety evaluation, the staff concludes:

- The design, testing, and performance of SSC important to safety during normal operation are acceptable; safe operation can reasonably be expected to continue
- The licensee's management organization is acceptable to maintain and safely operate the reactor

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Safety Conclusions

- The licensee's research activities and programs, including experiment malfunctions, will not pose a significant risk to continued safe operation of the facility
- Exposures from and releases of radioactive effluents and waste from the facility are not expected to result in doses or concentrations in excess of the limits specified in 10 CFR Part 20, and are consistent with as-low-as-reasonably-achievable principles

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Safety Conclusions

- The licensee has conservatively considered the consequences of a bounding maximum hypothetical accident and shown the radiological consequences to be a small fraction of those specified in 10 CFR Part 100
- The licensee has conservatively considered an appropriate range of postulated credible accidents using appropriate initiating and mitigating assumptions

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Safety Conclusions

- The renewed Facility Operating License and TSs provide reasonable assurance that the licensee will operate the facility in accordance with the assumptions in the SAR
- No significant degradation of SSC has occurred, and the TSs will continue to provide reasonable assurance that no significant degradation of SSC will occur

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Safety Conclusions

- The licensee's physical security plan continues to be acceptable to protect its special nuclear material
- The licensee's emergency plan provides acceptable assurance that the licensee will continue to be prepared to assess and respond to emergency events

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Safety Conclusions

- Continued operation of the NBSR during the period of the renewed license poses no significant radiological risk to the health and safety of the public, facility personnel, or the environment

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Resolution of Open Item

- Draft SER contained one open item regarding a timing requirement in the operator training and requalification program
- NRC issued an RAI on February 18, 2009
- The licensee responded March 3, 2009, stating that the requalification "program shall be administered over a period not to exceed 24 months, followed by successive 24 month periods."

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Resolution of Open Item

- The RAI response resolved the open item by changing the timing requirement in the operator training and requalification program to meet the regulatory requirement
- Final SER concludes that the program meets all applicable regulatory requirements and is consistent with standard ANSI/ANS-15.4, “Selection and Training of Personnel for Research Reactors,” issued 1988

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Additional Open Item

- In addressing the concerns of the ACRS subcommittee members, the licensee identified an unrelated inaccuracy in the pump coastdown data used in two loss-of-flow accident analyses
- The licensee will perform additional analyses to determine if the inaccuracy is safety-significant
- The staff will review the analyses, verify that the conclusions in the final safety evaluation report remain valid, and present the findings to the ACRS full committee later this year

Backup Material

- Responses to ACRS subcommittee member concerns

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Seismic Hazard Analysis

- NRC staff reviewed the seismic design basis for the facility (0.1 g peak ground acceleration)
- NRC staff reviewed past AEC and NRC approvals of the seismic design for the facility
- The staff concludes that the seismic design is adequate and conservative

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Seismic Hazard Analysis

- The licensee performed a seismic walk-down that identified a wall that, if it collapsed, could affect emergency core cooling system piping
- NRC staff visited the facility to review the walk-down
- The licensee will perform a seismic analysis for a 0.1 g earthquake and reinforce the wall if necessary
- The staff finds this acceptable

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Groundwater Monitoring

- The licensee performs groundwater monitoring by sampling surface water fed by groundwater for tritium and other nuclides
- This is consistent with analysis performed by the U.S. Geological Survey during initial licensing
- A 2006 report confirmed the original analysis
- The staff finds the current groundwater monitoring program acceptable

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Natural Circulation Cooling

- The licensee performed an analysis of natural circulation cooling following full power operation
- The safety margin is greater than 2
- There is adequate time (more than 2 hours) to manually open a valve to reestablish flow
- Even with no action, the reactor vessel will not over-pressurize
- The staff finds the analysis acceptable and the consequences are bounded by the MHA

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Single Failure Criterion

- The staff reevaluated redundancy in systems
- Electrical power is not required for maintaining safe shutdown, so redundancy only provides defense-in-depth
- Multiple monitoring devices for a single variable have diverse designs and are electrically and physically separated
- The staff finds the level of redundancy acceptable to prevent a single failure from causing an unanalyzed accident

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Surveillance Requirements

- The licensee provided additional justification for relaxing surveillance requirements
- Annual emergency sump pump testing is justified by the redundancy in emergency coolant sources
- Quarterly start tests of the diesel generators is consistent with the approved standard
- Annual surveillance of the station battery is supported by operational data and experience
- The staff finds the surveillances acceptable

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Experiment Review Process

- The licensee provided additional clarification of the extent of the experiment review and approval process
- The experiment review process meets the guidance found in the accepted standards
- The staff found that the technical specifications ensure that the licensee's experiment program will not pose a significant risk to the health and safety of the public

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Moderator Coefficients

- The licensee provided additional clarification of the consistency of the moderator void and temperature reactivity coefficients
- The coefficients are consistent when calculated for the entire coolant volume
- The coefficients are always negative
- The accident analyses do not take credit for the coefficients and are conservative
- The staff finds the licensee's treatment of the moderator coefficients acceptable

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Waste Disposal

- The licensee provided additional information on plans for storage and disposal of Class B and Class C wastes
- The NBSR license TR-5 provides possession limits for these wastes and requires appropriate storage and disposal
- The licensee plans to construct an interim storage facility with 40 years of capacity
- The staff finds the licensee's storage and disposal of these wastes acceptable

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Spent Fuel Pool Safety

- The licensee provided results of calculations of criticality safety for the spent fuel storage pool
- The licensee used conservative assumptions to show that the stored fuel cannot achieve criticality
- The effective multiplication factor remains below the limit of 0.9 in the technical specifications
- The staff finds the spent fuel pool storage facility acceptable for storage of spent fuel in accordance with license TR-5

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Site Meteorology - Wind

- The licensee provided additional information on the confinement building design for wind loads
- The building was design and built to the accepted standards at the time of construction
- The licensee's derivation of the maximum 100-year return wind speed was conservative
- The American Society of Civil Engineers (ASCE) standard gives a wind speed of 96 mph which is below the design limit of 100 mph

Site Meteorology – Snow

- The licensee provided additional information on potential roof snow loading
- The licensee's analysis is conservative compared to the ASCE standard
- The licensee's calculations and the ASCE standard give maximum snow loads that are below the design load for the confinement building roof

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Advisory Committee on Reactor Safeguards
561st ACRS Meeting
April 2-4, 2009

Draft Final Regulatory Guide 1.211, "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

Comments of the Nuclear Utility Group on Equipment Qualification

Nuclear Utility Group on Equipment Qualification ("NUGEQ")

- ***Formed in 1981 to address issues related to equipment qualification.***
- ***NUGEQ includes utilities operating nuclear power reactors in the United States and Canada.***
- ***NUGEQ represents over 95 operating nuclear power plants in the United States***

William A. Horin – Winston & Strawn, Counsel to NUGEQ
Philip M. Holzman – Strategic Technologies & Resources, Inc., Consultant to NUGEQ

NRC EQ Task Action Plan (EQ TAP) GSI-168, "Environmental Qualification of Low Voltage Instrumentation and Control Cables"

- NUGEQ actively participated in industry interactions with NRC staff – 1993 to 2003:
- Clear industry support for cable CM research
  - NUMARC comments to NRC, "we believe that research in the area of condition monitoring may be useful toward developing information for assessing the actual service life of equipment." (Reprinted in 1993 workshop proceedings NUREG/CP-0135 (pp. H-3 to H-6))
- Clear NRC staff interest in cable CM
  - 11/15/96 memorandum to the Commission – staff states it would be prudent to have some form of cable CM
- NRC undertook a new cable test program to address six issues.
  - Two of these issues involved cable condition monitoring (CM):
    - Issue 5 – whether condition monitoring methods exist that could be used to monitor the condition of cables in situ, and
    - Issue 6 – whether condition monitoring could be used to predict the accident survivability of cables

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Consideration of Cable Conditioning Monitoring

- October 2000 – testing completed; NRC Staff briefs ACRS on GSI-168
  - Subsequent public meetings and interactions including input on the need for monitoring plant environments and cable CM
- June 2002 – NRC presents GSI-168 proposed resolution to ACRS, including results of CM research and estimates of CM risk benefits

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NRC EQ Task Action Plan (EQ TAP) GSI-168, "Environmental Qualification of Low Voltage Instrumentation and Control Cables"

- NRC Staff Conclusions Regarding Cable Condition Monitoring
  - Methods – No single technique was effective and a combination of techniques would be needed.
  - Risk Benefits – the benefits of reducing cable failure probability to zero [an unrealistic but bounding condition that assumed *perfect* condition monitoring techniques exist] were modest at best.

- ACRS and NRC Conclusion
  - Actual cable condition monitoring benefits, if any, would be even less than modest.

- GSI-168 Contractor Report
  - Condition Monitoring results obtained in laboratory setting
  - Additional testing needed to determine effect of plant conditions on use of techniques in situ

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Regulatory Guide Regarding "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

- Draft Guide DG-1132
  - NRC concluded some form of cable CM needed to establish reasonable assurance of performance.
  - NRC asserted CM is needed in lieu of "doing nothing"
  - NUGEQ perspective
    - "Doing nothing" is mischaracterization
      - Environmental Monitoring
      - Walkdowns/Inspections
      - Corrective Action Programs
    - Efficacy of CM techniques not supported by the record
      - GSI-168 or subsequent information
    - Risk benefit of CM (even assuming perfect techniques) not supported by the record

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Regulatory Guide Regarding "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

- NUGEQ Conclusions and Recommendations
  - Environmental Monitoring & walkdowns/inspections are proven & effective (NRC and industry concur).
  - Condition Monitoring can supplement environmental monitoring and walkdowns/inspections when the method is proven and effective for the cable design and conditions of concern.
  - Cable aging management activities should be focused on risk significant cables.
  - Cable aging management activities should be focused on "at risk" cables (those with the least margin).
  - Regulatory Position 7 in Regulatory Guide should be modified to reflect above perspectives.

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Regulatory Guide Regarding "Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants"

NUGEQ Proposed Language

(7) Programs for monitoring of environmental conditions (such as temperature, radiation levels) coupled with walkdowns to look for visible signs of anomalies attributable to aging with particular emphasis on the identification of localized adverse environments or “hot spots” should be implemented for power, instrumentation, and control cables whose failures could disable risk-significant equipment. Where appropriate these activities should be supplemented with condition monitoring techniques that are proven to be effective for assessing in situ degradation for the cables and conditions of concern. Such activities are most effective when applied to cables with the least margin due to qualification levels, service conditions, or other application considerations. For safety-related power cables, which are inaccessible or installed underground and may be exposed to condensation and wetting, appropriate inspection, testing and monitoring programs should be implemented to detect degradation.

Query: Why is this language unreasonable?

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Regulatory Guide 1. 211
Qualification of Safety-Related Cables and Field Splices

ACRS Meeting: April 2, 2009

Satish Aggarwal
Division of Engineering
Office of Nuclear Regulatory Research
301-251-7627

Regulatory Guide 1. 211

BACKGROUND

Standards Activities:

-IEEE Std 383-1974

-IEEE Std 383-2003 published in June 2004

Regulatory Activities:

-DG-1132 was issued for public comment in June 2007 with 10 exceptions to IEEE Std 383-2003.

-Received comment letters from 5 organizations.

-Staff met with the ACRS on September 4, 2008.

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ACRS Concerns and the Staff Responses

ACRS:

1) Provide the definition of risk-significant safety-related equipment (e.g. cable).

Staff:

The terminology “risk significant” safety-related equipment has been removed from the RG. The condition monitoring of the safety-related cables is limited to those covered by the maintenance rule.

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Regulatory Guide 1.211 (March 2009) C. Regulatory Position (Revised)

(6) Programs for monitoring of environmental conditions (such as temperature, radiation levels), and condition monitoring should be implemented for power, instrumentation, and control cables (***the condition monitoring of safety-related cables may be limited to those cables covered by 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants.").*** Condition monitoring programs may include any appropriate technique(s), supplemented with walkdowns to look for visible signs of anomalies attributable to aging with particular emphasis on the identification of localized adverse environments or "hot spots." For safety-related power cables that are inaccessible or installed underground, appropriate inspection, testing and monitoring programs should be implemented to detect degradation. The condition monitoring and its frequency may be adjusted based on the cable performance.

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ACRS Concerns and the Staff Responses

ACRS:

2) Clarification is needed in the scope of the guide which should be limited to safety-related cables.

Staff:

The scope of the guide is limited to only safety-related cables.

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ACRS Concerns and the Staff Responses

ACRS:

3) The cables under Appendix R (Fire Protection) are not within the scope of the RG.

Staff:

The Appendix R requirements are not covered within this RG, however, they are covered by:

- Regulatory Guide 1.189 “Fire Protection for Nuclear Power Plants”; March 2007; Rev 1.
- NUREG-0800 “Standard Review Plan Sec 9.5.1: Fire Protection System”; October 2003; Rev 4, (Formerly NUREG-75/087 March 1979).
- “Guidelines for Fire Protection for Nuclear Power Plants”; Branch Technical Position ASB 9.5-1; May 1976.
- “Guidelines for Fire Protection for Nuclear Power Plants”; Branch Technical Position ASB 9.5-1; Rev 1; March 1978.
- “Recommended Fire Protection Policy and Program Actions” (GL 85-01); October 26, 1984.
- “NRC Positions on Certain Requirements of Appendix R to 10CFR50” (GL 83-33); October 1983.

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ACRS Concerns and the Staff Responses

ACRS:

4) Justify the need for testing specialty cables with connectors.

Staff:

This regulatory position was removed from the RG.

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ACRS Concerns and the Staff Responses

ACRS:

The RG does not offer any specific condition monitoring techniques to assess physical and operational conditions of the cables. The requirement for condition monitoring is being imposed without any conditions monitoring techniques being endorsed by the staff.

Staff:

The condition monitoring of safety-related cables should be implemented. The licensees need to choose the most applicable technique(s) based on the cable type and environment. There are about 14 condition monitoring techniques available.

Risk Metrics for Operating New Reactors

ACRS Committee Meeting

April 3, 2009

Biff Bradley - NEI

Doug True - ERIN

NEI

NUCLEAR ENERGY INSTITUTE

Overview

- Issue
- Evolution of risk metrics
- PRA & risk metrics in design and licensing
- Risk metrics in the operational phase
- LRF vs. LERF
- Quantitative thresholds
- Problems with different metrics

NEI

Issue to be Addressed

- NRC paper on risk metrics recently provided to ACRS and Commission
  - Perceived differences in risk metrics and thresholds between Part 52 licensing and currently operating plants
  - Concern with existing NRC risk informed processes (CLB changes, reactor oversight process, MSPI)
  - Prior NRC policy decisions called into question
  - Proposes six options for consideration
- Industry supports Option 1 (same metrics as for existing plants)

NEI

Evolution of Risk Metrics

- Part 52 risk metrics preceded Regulatory Guide 1.174 development
- Design Certifications and COLs rely upon:
  - Core Damage Frequency (CDF)
  - Conditional Containment Failure Probability (CCFP)
  - Large Release Frequency (LRF)
- Operating plants rely upon:
  - CDF
  - Large Early Release Frequency (LERF)

NEI

PRA & Risk Metrics in Design and Licensing

- Part 52 requirements and risk goals for new plants have put appropriate focus on:
  - A low and balanced computed CDF,
  - A low computed CCFP (<0.1) for the corresponding computed CDF, and
  - A low LRF for the corresponding computed CDF.
- These metrics have been effective for design and initial licensing purposes

NEI

LRF History

- Safety Goal Policy Statement introduced large release expectation
- Staff initially defined LRF qualitatively:

  "A large release is a release that has a potential for causing an offsite early fatality."
- Commission directed staff to provide more quantitative definition
- In 1993, the Commission directed NRC staff to abandon efforts to quantitatively define LRF
- New reactor vendors have provided their own definitions
  - Unique to Design Certification application
  - Generally very conservative and simplified definitions

NEI

LERF History

- In 1995, EPRI PSA Applications Guide introduced LERF as a risk metric
- Mechanistic definition, aligned to early fatality (EF) QHO
- LERF quantitative threshold of $1x10^{-5}$/ry suggested based on
  - Level 3 PRAs
  - Consistency with LRF goal
- Staff adopted LERF for RG 1.174 as surrogate for EF QHO
- PRA standards developed to address LERF

NEI

Alignment of LERF & LRF

- NUREG-1150 applied staff qualitative definition of LRF
- Recent comparison performed of internal event LERF accident progression bins to computed LRF
- Results show for all NUREG 1150 plants:

$$LERF > 10 * LRF$$

NEI

NUREG-1150 LERF vs. LRF

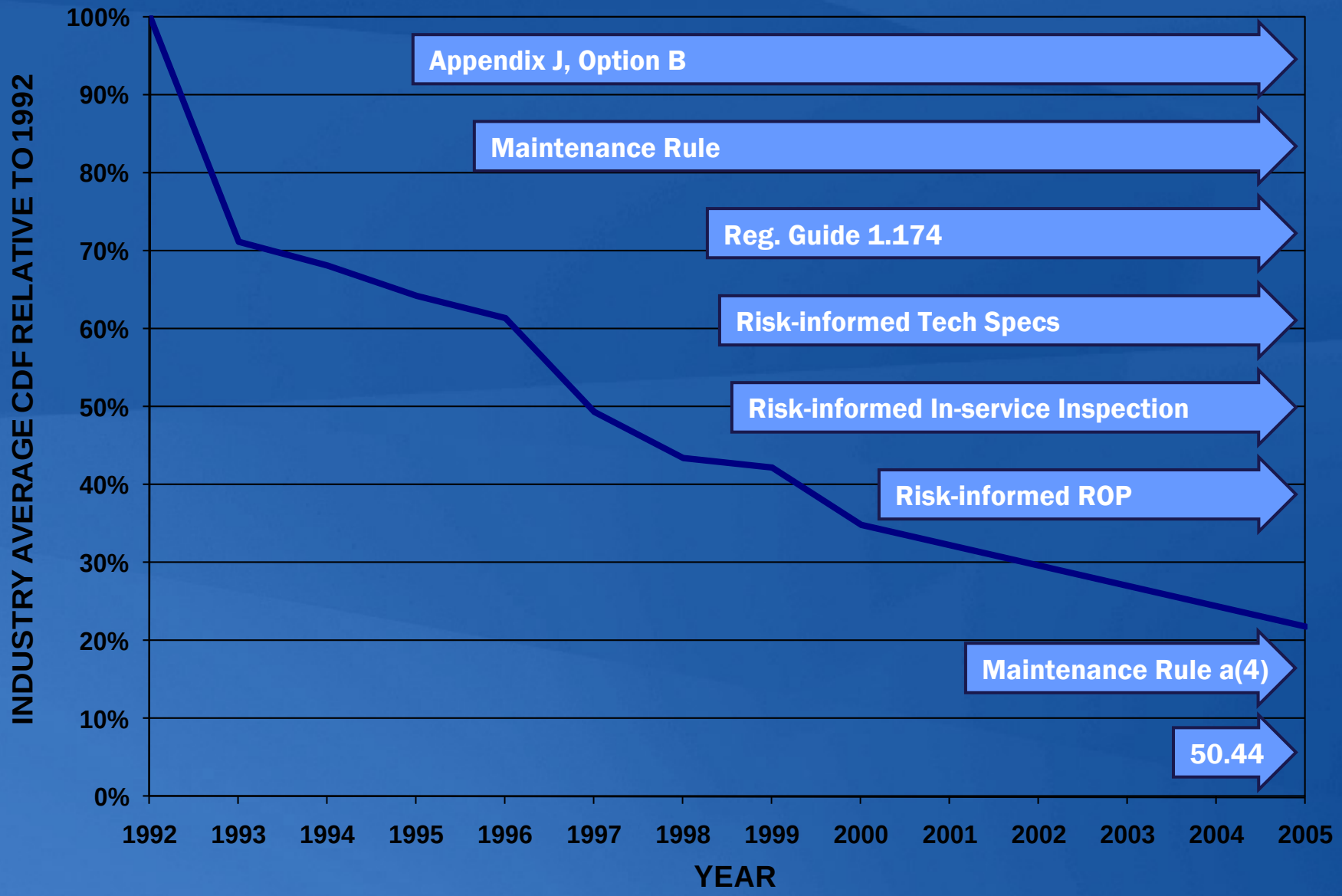
Plant	Estimated LERF (/yr)	Large Release Frequency (/yr)	Conditional Probability of LRF Given LERF
Peach Bottom	2.27E-06	1.0E-09	0.04%
Surry	5.29E-06	2.0E-07	3.8%
Grand Gulf	6.46E-07	3.0E-10	0.05%
Sequoyah	6.75E-06	6.0E-07	8.9%
Zion	7.10E-06	6.0E-07	8.5%

NEI

Risk Applications at Operating Plants

- RG 1.174 applications have not led to increased CDFs or reduction of margin
  - Changes constrained by other elements of risk informed process (DID, margins, etc)
  - Changes mostly granted in “very small” delta risk region of RG 1.174
  - Risk informed applications have led to improved safety, not reduction of safety

NEI


INDUSTRY AVERAGE CDF RELATIVE TO 1992
100%
90%
80%
70%
60%
50%
40%
30%
20%
10%
0%
1992
1993
1994
1995
1996
1997
1998
1999
2000
2001
2002
2003
2004
2005
YEAR
Appendix J, Option B
Maintenance Rule
Reg. Guide 1.174
Risk-informed Tech Specs
Risk-informed In-service Inspection
Risk-informed ROP
Maintenance Rule a(4)
50.44

Risk Metrics in The Operational Phase

- **Commission consistent in maintaining that new reactors would not be measured against a more stringent risk requirement**
- **Reiterating in 2008:**

  "… the policy statement* does not state that advanced reactor designs must be safer than the current generation of reactors, but rather that they must provide the same degree of protection of the environment and public health and safety and the common defense and security that is required for current-generation light-water reactors."

***Policy statement on Advanced Reactor Regulation, October 14, 2008**

NEI

Problems with different or new metrics for Part 52 plants

- Inconsistent with commission policy
  - Safety goal policy statement
  - Advanced reactor policy statement
- New risk metrics would penalize new plants
  - Limit operational flexibility (maintenance rule, Tech Specs)
  - Subject plants to inspection and enforcement at levels not corresponding to public health and safety
- New risk metrics would create public perception problems
  - For example, co-located sites with different thresholds for enforcement actions

NEI

Problems with different or new metrics for Part 52 plants

- The proposed risk metrics values could be well within PRA uncertainty bands
- Proposed new risk metrics could truncate ability of new plants to use risk applications
- Risk profiles for new reactors are not yet complete
  - Internal events, fire, external events (seismic) and possibly shutdown PRA will be required prior to fuel load

NEI

Preservation of Safety

- Risk applications contain deterministic backstops and DID considerations
- Part 52 contains comprehensive change control process that addresses severe accident features

NEI

Summary

- Part 52 licensing process and commission policy effective in enhancing new reactor severe accident prevention and mitigation capability
- New reactors should transition to RG 1.174 risk metrics when operating
- Maintains consistent commission policy and rational regulatory framework

NEI

U.S.NRC

UNITED STATES NUCLEAR REGULATORY COMMISSION

Protecting People and the Environment

Implementation of Risk Metrics for New Light-Water Reactor Risk-Informed Applications

Advisory Committee on Reactor Safeguards

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April 3, 2009

Meeting Purpose

Brief the ACRS regarding the implementation of risk metrics for new light-water reactor risk-informed applications, and identify potential paths forward.

Agenda

- Near term and longer term needs
- Background
- Implementation issues
- Options
- Status

Risk-Informed Initiatives for New Reactors

- In the near term, risk-informed applications have been proposed:
  - Risk-Managed Technical Specifications
    - Risk-informed completion times
    - Surveillance frequency control program
- Longer term initiatives (post-COL) may include:
  - EPRI research program on risk-informed inservice inspection of piping
  - Special treatment requirements (10CFR50.69)

Background: Risk Metrics for Operating Reactors

- Core Damage Frequency (CDF) < 10^{-4} /yr
  - Surrogate for latent cancer fatalities in the Commission's quantitative health objective (QHO)
- Large Early Release Frequency (LERF) < 10^{-5} /yr
  - Surrogate for prompt fatalities in QHO

Risk Goals for New Reactors

- Core Damage Frequency (CDF) < 10^{-4} /yr
- Large Release Frequency (LRF) < 10^{-6} /yr
- A deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges
- Conditional containment failure probability (CCFP) less than approximately 0.1

SRM on SECY-90-016, 6/26/90

Commission's Expectations

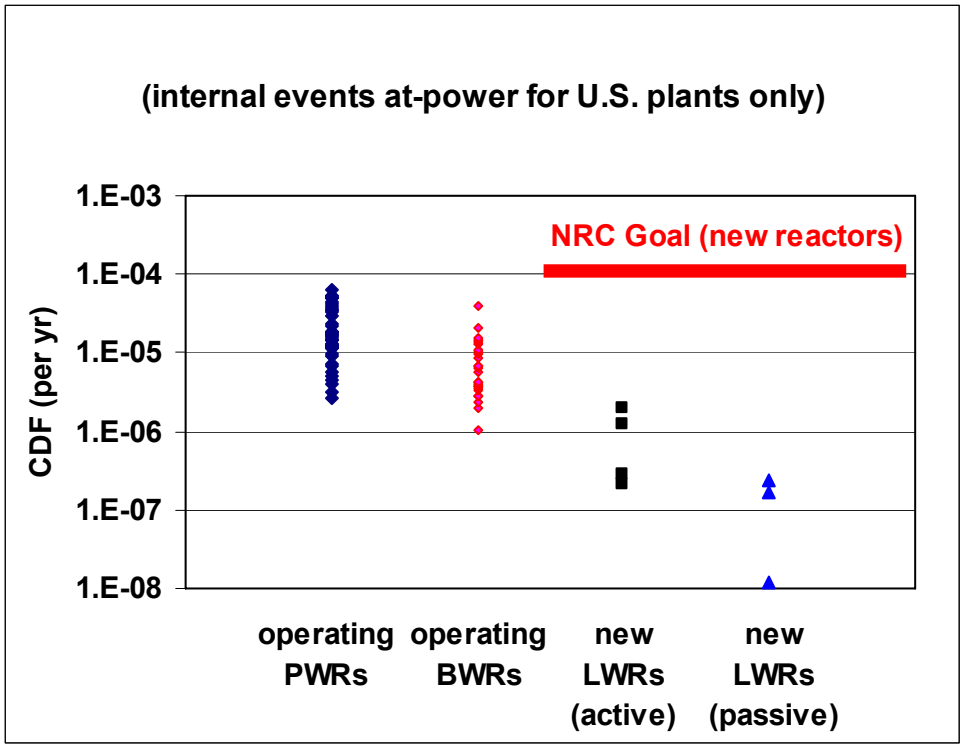
The Commission "fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs."

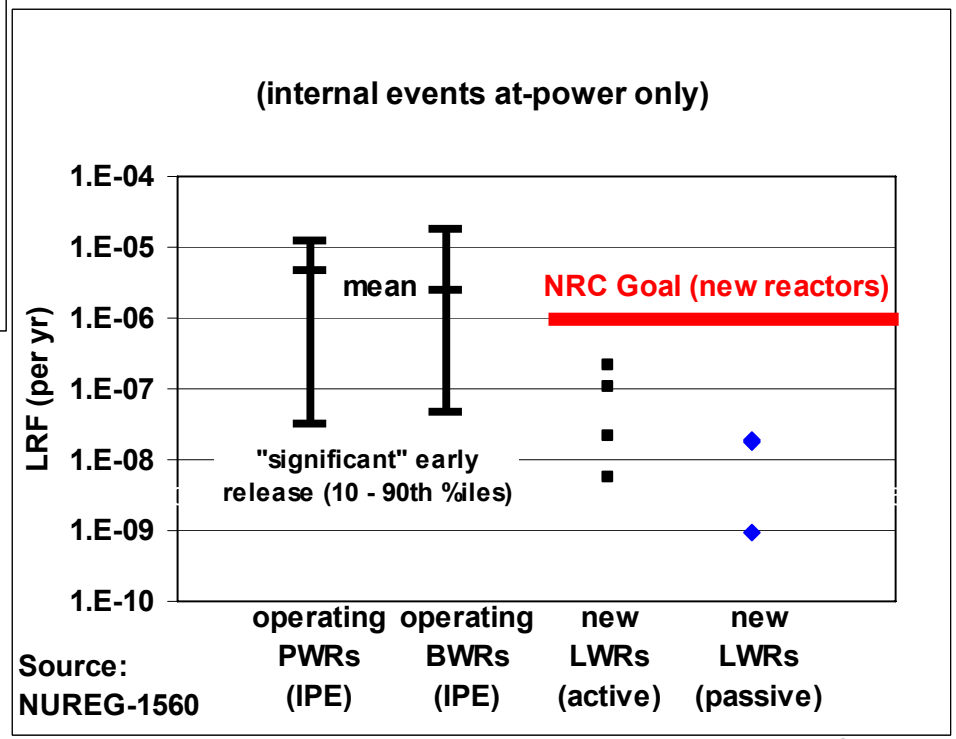
- August 1985

"The Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions."

- July 1994

CDF and LRF by Plant Type





Background (cont.)

- New reactor applicants' PRAs are expected to demonstrate how the design compares against the Commission Goals
  - RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)"
  - Standard Review Plan (SRP) Chapter 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors"

- Regulatory guidance associated with risk-informed initiatives are based on LERF (e.g., RG 1.174, 1.175, 1.177, 1.178, 1.201)

New Reactor Implementation Issues

- Review of these applications has raised questions about risk metric acceptance guidelines for implementation of risk-informed initiatives for new reactors:
  - LERF versus Commission's goal on LRF
  - LRF was not previously finalized in NRC documents.
- Use of current numerical risk metric goals (e.g., LERF) would result in risk-informed applications/amendments being evaluated against less restrictive criteria than those used for the licensing basis of new reactors.

New Reactor Risk Metrics

- **Licensing:**
  - **How should acceptance guidelines for new reactor license applications or amendments proposing to implement risk-informed initiatives consider Commission's expectations:**
    - **CDF?**
    - **LRF?**

- **Operations:**
  - **Reactor Oversight Process thresholds rely on CDF, ΔCDF, conditional core damage probability (CCDP), incremental CCDP, LERF, ΔLERF, etc.**
  - **How should risk metrics for new reactor operations consider Commission's expectations?**

- **Focus on needs for licensing first**

Regulatory Guidance

- **Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"**

- **Risk-Acceptance Guidelines:**
  - **Baseline risk metrics of CDF and LERF**

    AND
  - **ΔCDF and ΔLERF due to change**

- **Basis:**
  - **Increases should be limited to small increments**
  - **CDF threshold related to backfit regulatory analysis guidelines**
  - **ΔCDF limit based on *absolute* change and set close to limit of resolution of PRA models**

From RG 1.174

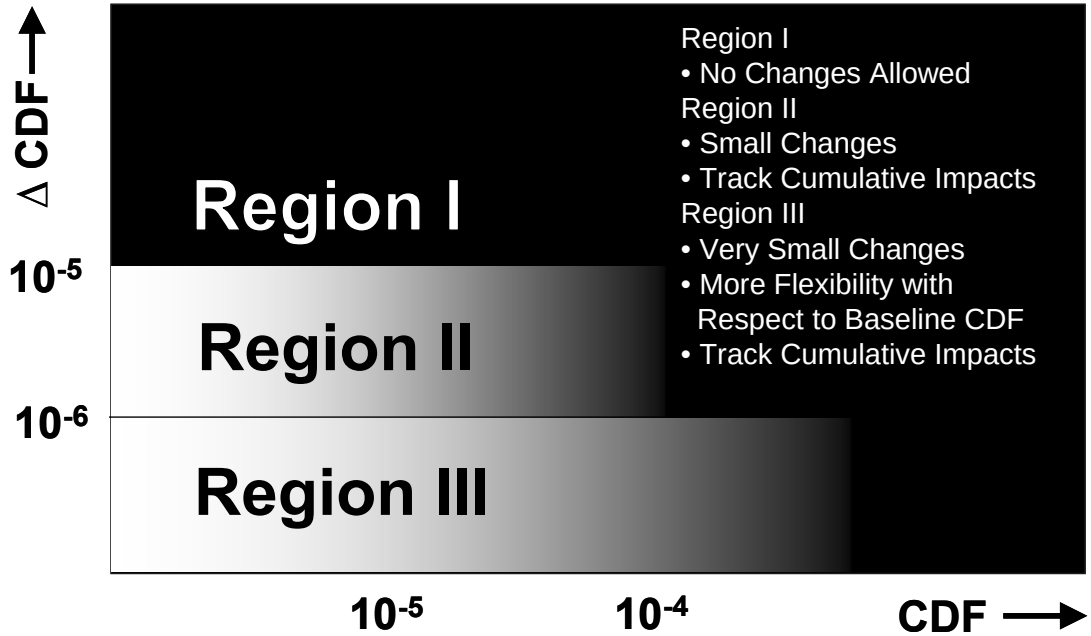


Figure 3. Acceptance Guidelines for Core Damage Frequency (CDF)

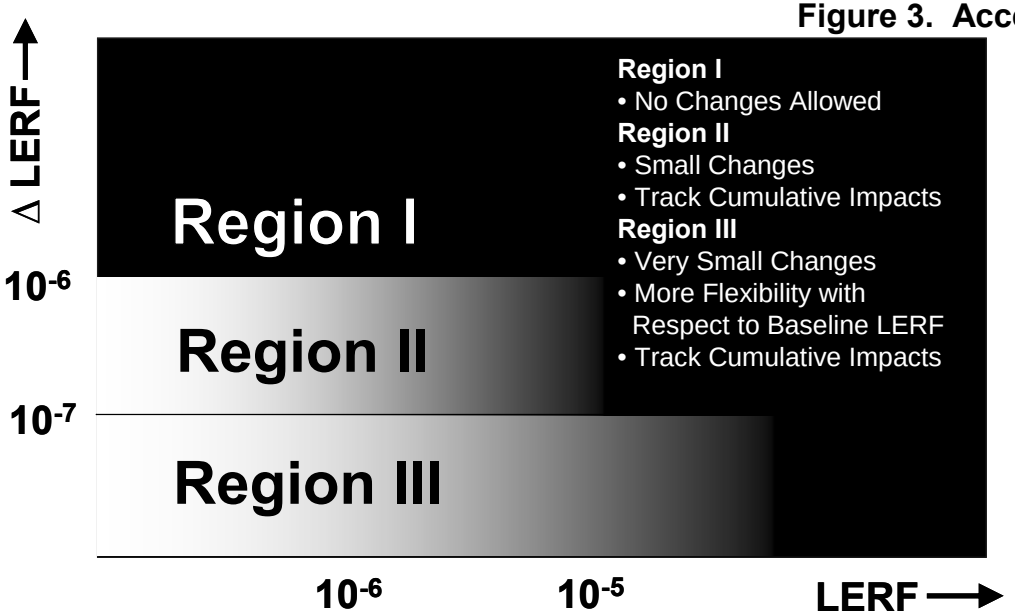


Figure 4. Acceptance Guidelines for Large Early Release Frequency (LERF)

Regulatory Guidance (cont.)

- **For new reactors, should the principle of "small increase" be based on *relative* or *absolute* ΔCDF and ΔLERF or ΔLRF?**

- **Should RG 1.174 include an alternate or additional ΔLRF acceptance guideline for new reactors?**

Regulatory Guidance (Cont.)

- Other programs, processes, and regulations:
  - Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program" (specifically, ILRT test interval extension)
  - Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing"
  - Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications"
  - Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping"
  - Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"
  - Regulatory Guide 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," 10CFR50.69
  - 10CFR50.65, "Maintenance Rule"

Some Possible Options

- **Option 1 Status Quo:** Current acceptance guidelines in RG 1.174 (and associated regulatory guides) would also be applied to new reactors
- **Option 2 Convert to *Relative* Risk Changes:** New and current reactors
- **Option 3 Reduce Acceptance Guidelines for New Reactors:** Acceptance guidelines in RG 1.174 would be lowered by 1 or more orders of magnitude solely for new reactors
- **Option 4 Use a Combination of Existing and New Acceptance Guidelines**
- **Option 5 Use Existing Acceptance Guidelines for Current and New Reactors (*Status Quo*), but Establish an LRF-Based Acceptance Guideline for New Reactors**
- **Option 6 Assess New Reactors on a Case-by-Case Basis**

Status

- White paper issued (Adams # ML090430220)
- EDO Memorandum to Commissioners, February 12, 2009 (ML090160008)
- Public meeting held February 18, 2009
- Presentation at RIC 2009
- Continued engagement of stakeholders

Results of February 18, 2009 Public Meeting

- Broad representation of stakeholders in attendance
- Staff described pending risk-informed applications and implementation issues
- Significant discussion on derivation of LRF and CCFP
- Advantages and disadvantages of each option discussed
- Additional sub-option suggested whereby one might proceed with the current set of risk metrics per RG 1.174 on the first few risk-informed applications, then to assess the need for change based on lessons learned
- Industry to follow-up with white papers regarding its views on the historical perspective of *large release* as well as preferred option(s)

Back-up Slides

Options in Detail

Option 1 Status Quo: Current acceptance guidelines in RG 1.174 (and associated regulatory guides) would also be applied to new reactors

Advantages

- Provides a consistent set of acceptance guidelines for both existing and new reactors
- The bases for RG 1.174 acceptance guidelines are derived from Commission's 1986 Safety Goals
- Acknowledges and gives credit to new reactors for lower risk estimates

Disadvantages

- May not be consistent with Commission 1985 policy statement on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."
- Less restrictive change process than the Commission established for the review of new reactors
- Options could allow large ***relative*** increases in CDF and LERF compared to the baseline CDF and LERF estimates for new reactor designs

Options (cont.)

Option 2 Convert to *Relative* Risk Changes: New and current reactors

Advantages

- Option recognizes that “small increase” is a relative measure and precludes large % change in CDF and/or LERF for new reactors

Disadvantages

- Inconsistent with the underlying technical basis for the current *absolute* thresholds in RG 1.174
- Potential disagreement between industry and staff regarding what constitutes the “baseline” for CDF and LERF changes
- Major changes to current regulatory guides and other processes would be required
- Impacts currently operating reactors
- Results in inconsistency between existing and new reactors, and may be viewed as penalizing new reactors for having lower risk estimates
- Transition from existing *absolute* acceptance guidelines to *relative* (% changes) could be difficult
- Depending on the chosen limits for acceptance guidelines, past conditions that were deemed acceptable might not be found acceptable under the new formulation

Options (cont.)

Option 3 Reduce Acceptance Guidelines for New Reactors: Acceptance guidelines in RG 1.174 (and associated regulatory guides) would be lowered by 1 or more orders of magnitude solely for new reactors

Advantages

- Acknowledges that new reactor CDF/LERF estimates are significantly lower than existing reactors and adjusts acceptance guidelines accordingly
- Consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."

Disadvantages

- Inconsistent with the underlying technical basis for the current ***absolute*** thresholds in RG 1.174
- Penalizes new reactors for having lower risk estimates
- Results in different treatment for new and current reactors of a proposed licensing basis change resulting in a ΔLERF of 4×10^{-8}/yr, for example
- May be inconsistent with the Commission's Safety Goal Policy Statement on acceptable level of risk

Options (cont.)

Option 4 Use a Combination of Existing and New Acceptance Guidelines

Advantages

- Option addresses some of the concerns regarding large ***relative*** changes to risk with new reactors
- Consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."

Disadvantages

- Inconsistent with the underlying technical basis for the current ***absolute*** thresholds in RG 1.174
- Penalizes new reactors for having lower risk estimates
- Results in different treatment at new and current reactors of a proposed licensing basis change resulting in a ΔLERF of 4×10^{-8}/yr
- May be inconsistent with the Commission's Safety Goal Policy Statement on acceptable level of risk

Options (cont.)

Option 5 Use Existing Acceptance Guidelines for Current and New Reactors (*Status Quo*), but Establish an LRF-Based Acceptance Guideline for New Reactors

Advantages

- Option consistent with the goals that the Commission established for the review of new reactors
- Provides a consistent set of acceptance guidelines for both existing and new reactors with regard to ΔCDF
- Consistent with the underlying technical basis for the current absolute thresholds for ΔCDF and ΔLERF in RG 1.174, as modified to reflect Commission policy regarding ΔLRF for new reactors
- Consistent with Commission policy statements on expectations that new reactor designs "will achieve a higher standard of severe accident safety performance…."
- Allows anticipated risk-informed initiatives to move forward

Disadvantages

- Options could allow large *relative* increases in CDF and LERF compared to baseline CDF and LERF estimates for new reactors
- Requires significant changes to regulatory guides

Options (cont.)

Option 6 Assess New Reactors on a Case-by-Case Basis

Advantages

- No changes needed to regulatory guides and related documents for current reactors
- Staff could await the accumulation of sufficient new reactor operating experience before making a decision on the treatment of new reactors

Disadvantages

- Current and new reactors would be treated inconsistently
- New reactor applicants/licensees would not know the acceptance guidelines for risk-informed initiatives
- Defers any decision on the treatment of new reactors