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

September 11, 2008

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 554th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS), July 9-11, 2008

I certify that based on my review of the minutes from the 554th 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	08/11/08	08/11/08

OFFICIAL RECORD COPY

CERTIFIED

Date Issued: 9/3/08
Date Certified: 9/11/08

TABLE OF CONTENTS
MINUTES OF THE 554th ACRS MEETING

July 9-11, 2008

APPENDICES

During its 554th meeting, July 9-11, 2008, the Advisory Committee on Reactor Safeguards (ACRS) discussed several matters and completed the following reports, letter, and memorandum.

REPORTS

Reports to Dale E. Klein, Chairman, NRC, from William J. Shack, Chairman, ACRS:

- Security and Aircraft Impact Rulemaking for Nuclear Power Plants, dated July 18, 2008
- Stretch Power Uprate Application for the Millstone Power Station, Unit 3, dated July 23, 2008

LETTER

Letter to R. W. Borchardt, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

- Interim Letter 4: Chapter 3 of the NRC Staff's Safety Evaluation Report with Open Items Related to the Certification of the ESBWR Design, dated July 21, 2008

MEMORANDUM

Memorandum to R. W. Borchardt, Executive Director for Operations, NRC, from Frank P. Gillespie, former Executive Director, ACRS:

- Draft Regulatory Guides 1149 and 1189, dated July 15, 2008

MINUTES OF THE 554th MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
July 9-11, 2008
ROCKVILLE, MARYLAND

The 554th meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on July 9-11, 2008. Notice of this meeting was published in the *Federal Register* on June 20, 2008 (72 FR 35172-35173) (Appendix I). The purpose of this meeting was to discuss and take appropriate action on the items listed in the meeting schedule and outline (Appendix II). The meeting was open to public attendance.

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. William J. Shack (Chairman), Dr. Mario V. Bonaca (Vice-Chairman), Dr. Said Abdel-Khalik (Member-at-Large), Dr. George E. Apostolakis, Dr. Sam Armijo, 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, Mr. John Sieber, and Mr. John Stetkar. For a list of other attendees, see Appendix III.

I. <u>Chairman's Report</u> (Open)

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

Dr. William J. Shack, 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. Shack 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. Dr. Shack welcomed Mr. Harold Ray and Dr. Michael Ryan as new official members and stated that the Committee was now at its statutory strength of 15 members.

II. Stretch Power Uprate Application for Millstone Power Station

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

The Committee met with representatives of the NRC staff, Dominion Nuclear Connecticut, Inc, and members of the public to discuss Dominion's license amendment request to increase the power level of Millstone Unit 3 by 7%. Topics of discussion included fuel system and nuclear design as well as containment and design basis accident analyses. Also discussed were proposed modifications to the plant to support the increased power level such as changes to the core design and disabling automatic control rod withdrawal. The plant safety analyses have been performed in many cases, with more modern methods than when the plant was initially licensed. These analyses show substantial margins to licensing limits for containment design pressure, peak cladding temperature, and departure from nucleate boiling. Mr. Gunderson of the Citizens Against Millstone described a concern that the containment design pressure could be exceeded. The Committee issued a report to the NRC Chairman on this matter dated July 23, 2008, recommending that the application for power uprate at Millstone Unit 3 be approved.

III. Selected Chapters of the Safety Evaluation Report (SER) Associated with the Economic Simplified Boiling Water Reactor (ESBWR) Design Certification Application

[Note: Mr. Harold Vandermolen was the Designated Federal Office for this portion of the meeting.]

The Committee met with representatives of the NRC staff and General Electric-Hitachi Nuclear Energy to discuss Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the NRC Staff's SER with Open Items related to the ESBWR Design Certification Application. The discussion focused on classification of Structures, Systems, and Components (SSCs) and the seismic analysis.

The SSCs are classified as Safety Class 1, 2, 3, or N depending on whether the SSC is needed to preserve the integrity of the reactor coolant pressure boundary, to shut down the reactor and maintain it in a safe shutdown condition, or to prevent or mitigate potential offsite exposures. Thus, the reactor coolant pressure boundary components and supports are classified as Safety Class 1 whereas nonsafety-related SSCs are classified as Class N. Safety Classes 1 through 3 are very closely related to Quality Groups A through C. The quality groups are defined in terms of their pressure retaining functions. Pressure retaining components of the reactor coolant pressure boundary are Quality Group A. Finally, there is a seismic classification; all safety-related SSCs are placed in Seismic Category I, which means they must remain functional in the event of a design basis earthquake. Nonsafety-related SSCs may be placed in Seismic Category II, which means that they need not remain functional, but must not fail in such a way as to interfere with safety-related SSCs. The remaining nonsafety-related SSCs may be assigned to Seismic Category NS, which means that they must conform to the International Building Code but have no further seismic design requirements.

Regarding the seismic design, the Combined Seismic Design Response Spectra (CSDRS) are based on Regulatory Guide 1.60 spectra with the addition of the North Anna site-specific spectra at high frequencies, i.e., the CSDRS is the envelope of the generic and North Anna spectra.

North Anna is representative of most severe rock sites in the eastern US, and thus the CSDRS envelopes most candidate sites with considerable conservatism.

The fluids in the reactor building pools are modeled as a mass-spring (sloshing) component and an impulsive (rigid) component. However, for conservatism, the entire water mass of each pool is considered as an impulsive mass in the seismic stick model for predicting overall building response. The sloshing component generally responds at very low frequencies (below 0.5 Hz), where no structural modes of vibration exist. The seismic loads used in the stress analysis of pool structures include both the global loads calculated from the seismic response analysis and local hydrodynamic pressure loading on the pool boundaries.

The Committee issued a letter to the EDO on this matter, dated July 21, 2008, stating that the evolving nature of the ESBWR design makes it difficult to perform an effective review, and that additional information is needed to demonstrate that dynamic forces from seismic events are treated properly in the analyses of heat exchangers immersed in elevated water pools.

IV. Safeguards and Security Matters

[Note: Ms. Maitri Banerjee was the Designated Federal Official for this portion of the meeting.]

The Committee met with representatives of the NRC staff and a member of the pubic to discuss the draft final rules on security and aircraft impact assessment. Consistent with the Commission direction in the October 31, 2003, Staff Requirements Memorandum, the Committee did not review the elements of the security rule that dealt with physical security. The ACRS review was limited to three parts of the rule: (i) 10 CFR 50.54(hh), “Mitigative Strategies and Response Procedures for Potential or Actual Aircraft Attack;” (ii) 10 CFR 73.54 “Protection of Digital Computer and Communication Systems and Networks;” and (iii) 10 CFR 73.58, “Safety/Security Interface Requirements for Nuclear Power Reactors.” The Committee also reviewed the draft final rule, “Consideration of Aircraft Impacts for New Nuclear Power Reactor Designs.” The staff discussed the essential elements of each rule, how comments from the public were addressed, and the status of the associated regulatory guidance. Mr. James Riccio of Greenpeace stated that the aircraft impact rule, in his opinion, lacks substantive acceptance criteria, and the requirements that the containment remains intact may not be sufficient.

The Committee issued a report to the NRC Chairman on this matter, dated July 18, 2008, recommending that the draft final rules be approved. The committee agreed with the staff that it is appropriate to treat aircraft attacks as beyond-design-basis events.

V. Status of NRC Staff Activities Associated with Seismic Design Issues at Nuclear Power Plants

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

The Committee was briefed on four topics under this agenda item. These briefings were intended to provide the Committee with an overview of ongoing and future staff activities related to the evaluation of earthquakes and how that information will be used to evaluate the safety of commercial nuclear power reactor designs. Dr. William Hinze (Professor emeritus of geophysics at Purdue University) was an invited Committee consultant.

(i). NRC Seismic Research Program Plan – FY 2008-2011:

To aid in future decision-making, NRC's Office of Nuclear Regulatory Research (RES) staff determined that there is a need for the agency to invest heavily in seismic-oriented research to ensure that the most up-to-date science and engineering is available to the staff. To this end, RES issued a Seismic Research Program Plan in January 2008. Speaking on behalf of the RES staff, Dr. Annie Kammerer provided an overview of the plan. As part of that overview, she noted that the research plan identifies approximately 40 research topics (projects) distributed among the following four subject areas:

- Earth Science and Natural Hazards Research (10 projects)
- Earthquake Engineering Analysis and Earthquake Resistant Design (14 projects)
- Cooperation in Ongoing International Research Activities (7 projects)
- Updates to NRC Regulatory Guides (6 projects)

Dr. Kammerer also noted that the scope of this plan includes certain briefing topics such as when to update a probabilistic seismic hazard analysis (PSHA) and how to consider that updated information in regulatory decision-making, generic seismic design issues effecting the next generation of advanced nuclear power reactors, and potential lessons-learned from the July 2007 earthquake at the Kashiwazaki-Kariwa Nuclear Power Plant site. This plan is intended to be implemented over the next three years, as resources permit. The plan has also been coordinated with NRR and NRO.

(ii). Status of Generic Safety Issue (GSI) 199:

A presentation on Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern U.S. on Existing Nuclear Power Plants was led by Mr. Marty Stutzke and Dr. Jon Ake, both representing RES. As background, in a 2004 (unpublished) analysis prepared for RES, the U.S. Geological Survey (USGS) reported that the peak ground acceleration (PGA) reference probability for the 29 Central Eastern United States (CEUS) nuclear power reactor sites had been recalculated and found to have increased annually to somewhere in the range of 6 to 7e-5. This revised estimate was developed taking into account new geologic information acquired by Survey researchers in the years following completion of the earlier Lawrence Livermore National Laboratory (LLNL) and Electric Power Research Institute (EPRI) PSHAs. Nevertheless, it was noted that in light of the revised PGA estimates for CEUS sites, the probability of exceeding the SSE at some nuclear power plants east of the Mississippi River is now believed to be generally higher than previously understood. The impact of updated geologic information on earlier NRC licensing decisions is not new and in many ways similar to situation that occurred once before, in the 1980s, when USGS researchers reported that new geologic information had led it to reinterpret the source for the Charleston 1886 earthquake. This reinterpretation ultimately compelled the NRC to undertake the Seismic Margins Program (SMP) which included individual plant examinations (IPEs) in the form of walk-downs. Inasmuch as there will continue to be the collection of new geologic information through basic field studies and a continuing interpretation of that information against existing seismic paradigms, the challenge for nuclear power plant designers will be the selection of a sufficiently robust safe shutdown earthquake (SSE) capable of meeting potentially greater earthquake hazards in the future. In association with this GSI, the RES staff representatives noted that they are reviewing documentation associated with the earlier SMP/IPEs to better understand what seismic margin currently exists for the current fleet of nuclear power plants and then evaluate that new PGA estimates against those margins.

In a related area, it was acknowledged that in early 2008, the Nuclear Energy Institute (NEI) submitted three seismicity-related White Papers to the NRC staff. Collectively, the trio of White Papers concern the robustness of the earlier LLNL and EPRI PSHAs, and how those analyses, currently referenced by the staff in various NRC regulatory guides, would be suitable for use in future seismic design basis decision-making. EPRI also has a task underway to update its PSHA for the CEUS. Absent an updated PSHA for CEUS sites, the NEI White Papers provide the staff with some constructive advice on how one might factor the existing PSHA knowledge base to NRC's ESP/COL decision-making currently taking place. Staff is reviewing these White Papers, and they are expected to produce RAIs in the near future.

(iii). Final Interim Staff Guidance (ISG) on Seismic Issues Associated with High Frequency (HF) Ground Motion:

This presentation was led by Dr. Manas Chakravorty representing the Office of New Reactors (NRO). By way of introduction, Dr. Chakravorty noted that through its review of seismic information associated with new nuclear power reactor licensing actions, the NRC staff recently learned that for some current and future nuclear power plant sites, the site specific ground motion may exceed the ground motion derived from a LLNL- or EPRI-based PSHA. For CEUS sites, this "exceedance" is generally in the HF range of ground motion. In addition to the HF issue, other seismic-related issues bearing on nuclear power plant design and performance have emerged. They include:

- Definitions of the various ground motions used in the design and site-specific analyses
- Definition/specification of the operating basis earthquake (OBE_ and SSE)
- Reaching a clear understanding of ground motions to be used in the certified design portion, site-specific design portion, and operability considerations
- OBE exceedance and location of seismic instrumentation
- Development and justification of incoherency functions,
- Use and validation of computer codes for seismic incoherency analysis
- Scope of the analyses and approaches to be used to address high-frequency responses for structures, systems, and components

In an attempt to address all of these issues in a manner that is both orderly and timely, the staff undertook development of an ISG. This ISG, designated DC/COL-ISG-01, was released for public comment in August 2007. Following stakeholder meetings in late 2007 and early 2008, DC/COL-ISG-01 was issued in final form in May 2008. The ACRS was not asked to review this ISG prior to its release.

(iv). July 2007 Earthquake at the Kashiwazaki-Kariwa (KK) Nuclear Power Plant Site:

The last presentation was conducted by Dr. Yong Li representing NRO. The Tokyo Electric Power Company (TEPCO) owns 17 of the approximately 50 commercial nuclear power reactors in operation in Japan. Seven are at one location – the Kashiwazaki-Kariwa (KK) site on the south-west shoreline of the Niigata Prefecture, on the Sea of Japan. The TEPCO KK site contains the largest complex of operating nuclear power reactors in the world. The combined generating capacity of the seven independent power reactors there is about 8200 MW. The first power reactor at the TEPCO KK site went on-line in 1985. The last reactor unit, TEPCO KK-7, went on-line about 1996.

On July 16, 2007, a magnitude 6.8 earthquake occurred about 9 kilometers offshore of the TEPCO KK site. The shallow-depth earthquake, designated the Niigataken Chãetsu-oki earthquake, affected a 100-km-wide portion of the coastline that included the TEPCO KK site. At the time of the earthquake, three of the seven TEPCO reactor units had already been shut down for planned outages. The site is instrumented with 97 strong-motion recorders and this network captured the highest-amplitude ground motions ever recorded at a nuclear power plant site. PGAs as high as 0.69g were recorded at the bases of some of the reactor buildings. PGAs at the tops of some steam turbines and reactor building roofs were reported to be as twice as high. (In the U.S., the largest PGA assumed in some reactor designs is 0.3g.) The differential seismic behavior of structures within the TEPCO KK complex has been attributed to variations in the thickness and character of the soils at the site as well as to the different types of nuclear power plant designs present at the site. In general, the PGA associated with the July 16 earthquake was reported to be two-and-a-half times greater than what was originally assumed to be acceptable for an earthquake design scenario at the TEPCO KK site.

In his presentation, Dr. Li noted that although structures, systems, and components (SSCs) important to safety appeared to have performed well, there were several incidents involving noncritical SSCs at the TEPCO KK site. For example, there was a fire in an electrical transformer located near Unit 3. There were reports of surging and over-topping of water from most of the seven reactors' spent fuel pools. There were failures in some joints in exhaust pipes. About 400 storage drums containing solid low-level radioactive waste capsized; some 40 of those drums lost their lids as a result. Radiation leaks were also reported. In addition, the site and surrounding area experienced geotechnical problems commonly associated with seismic events. These problems included evidence of soil liquefaction, differential ground displacement and settlement (on the order of <40 cm vertically), and the failure of a man-made embankment in the form of a landslide. Overall, TEPCO officials identified about 60 seismically-driven “incidents” at their facility that could be attributed to the July 2007 earthquake.

As of February 2008, TEPCO reported that it had completed visual in-core inspections for all 7 reactor units at the Kashiwazaki-Kariwa site. Officials reported that no abnormalities were found that could impact the functional or structural integrity of the reactor units. All seven reactor units at the TEPCO KK site remain off-line while additional site inspections and assessments take place. It appears that a key feature of the TEPCO assessment is a reevaluation of the seismic hazard at the Kashiwazaki-Kariwa site in the context of existing design basis for the reactor units. Staff representing the NRC, EPRI, the USGS, and the International Atomic Energy Agency have participated in or observed inspections of the TEPCO KK complex.

In June 2008, the NRC staff received a briefing on the Niigataken Chãetsu-oki earthquake during a drop-in meeting with TEPCO representatives. In November 2008, the American Nuclear Society plans to dedicate a session to lessons-learned from the Niigataken Chãetsu-oki 2007 earthquake at its annual meeting.

Dr. Dana Powers, the cognizant ACRS member for this subject area, recommended that an ACRS Subcommittee undertake more detailed reviews of some of the topics and issues discussed. The initial list of topics proposed for review included:

- A detailed briefing on the 2008 NRC Seismic Research Program Plan
- The forthcoming USGS recommendations on when to update a PSHA (i.e., the advice of the Senior Seismic Hazard Analysis Committee found in NUREG/CR-6372)
- The NRC staff review of the seismic margins for currently operating nuclear power plants and their proposal for addressing GSI-199
- The NRC staff views concerning the disposition of three NEI 2008 seismic White Papers
- The 2005 consensus standard developed by the American Society of Civil Engineers concerning the seismic design of nuclear power reactors (i.e., ASCE 3-05[1]) as well as the consideration of other seismic variables that might effect nuclear power plant design and operation
- Non-U.S. approaches to the definition of OBS and SSE

VI. Containment Overpressure Credit

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

The Committee met with representatives of the NRC staff and the Tennessee Valley Authority (TVA) to discuss technical issues related to crediting of containment overpressure during design basis accidents and special events in support of the extended power uprate for Browns Ferry Units 1, 2, and 3.

Representatives of TVA described how credit for containment overpressure is part of the current licensing basis for Appendix R and Loss-of-Coolant-Accident requirements. Of the two, the Appendix R is the more limiting. During a postulated fire event in two specific locations, if all of the equipment in these locations is rendered inoperable by the fire, there will not be sufficient net positive suction head for the residual heat removal pumps if the containment overpressure resulting from primary system blowdown is not credited. However, TVA claims that this assumption, when taken together with other licensing basis assumptions, is overly conservative. Moreover, based on discussions with the pump vendor, TVA claimed that there is a high likelihood that the pumps would survive the period of low suction head. Several members expressed an interest in the pump data and how they were obtained, since the pump tests were performed when the pumps were new in the 1970s and the pump rotors were replaced in the 1990s. In addition, the fire hazard analysis is a deterministic analysis; the licensee does not have a fire Probabilistic Risk Assessment. Several Members discussed the lack of any means for quantifying the degree of conservatism claimed by TVA.

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

---

[1] American Society of Civil Engineers, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," Reston, Nuclear Standards Committee, ASCE/SEI 43-05, 2005.

VI. Executive Session

[Note: Mr. Frank Gillespie 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 June 20, 2008, to conclusions and recommendations included in the April 30, 2008, ACRS report on the Draft NUREG-1902, "Next Generation Nuclear Plant Licensing Strategy Report." 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 July ACRS Meeting

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

Anticipated Workload for ACRS Members

The anticipated workload for ACRS members through October 2008 was discussed. The objectives were:

- Review the reasons for the scheduling of each activity and the expected work product and to make changes, as appropriate
- Manage the members' workload for these meetings
- Plan and schedule items for ACRS discussion of topical and emerging issues

Staff Requirements Memorandum Resulting from the ACRS Meeting with the Commission

In a Staff Requirements Memorandum (SRM) dated June 26, 2008 which resulted from the ACRS meeting with the Commission on June 5, 2008, the Commission stated the following:

- At the next Commission briefing on digital I&C, the staff should report the progress made with respect to identifying and analyzing digital I&C failure modes, and discuss the feasibility of applying failure mode analysis to the quantification of risk associated with digital I&C.

- The staff should continue working to address Committee concerns, such as SOARCA, digital I&C, and containment overpressurization, and, as necessary and appropriate, provide timely policy decision papers to the Commission to resolve any disagreements.

- Direction to the staff regarding SOARCA will be provided in the SRM for SECY-08-0029, "State-of-the-Art Reactor Consequence Analysis — Reporting Offsite Health Consequences," which is currently before the Commission for voting.

Status of the Quality Assessment of Selected NRC Research Projects

The Committee is in the process of assessing the quality of the following NRC research projects:

- FRAPCON/FRAPTRAN Code work at the Pacific Northwest National Laboratory and
- NUREG-6948, "Study of Remote Visual Methods to Detect Cracking in Reactor Components"

The Panel Chairmen provided a brief report on their preliminary findings.

Visit to the Braidwood Nuclear Plant and Meeting with the Region III Administrator

Several members of the Committee plan to visit the Braidwood Nuclear Plant on July 23 and meet with the Regional Administrator on July 24, 2008. An itinerary and logistics for the plant visit and meeting with the Regional Administrator were discussed.

Proposed Regulatory Guides,

- DG-1149, Qualification of Safety Related Motor Control Centers for Nuclear Power Plants"

  DG-1149 is a new Regulatory Guide, which describes a method for qualification of safety-related motor control centers for nuclear power plants. This Guide endorses, with certain exceptions, IEEE Standard 649-2006, "Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations." This Standard provides the basic principles, requirements, and methods for qualifying safety-related motor control centers for applications in both harsh and mild environments in nuclear power plants.

  The staff plans to issue DG-1149 for public comment and would like to know whether the Committee wants to review this Guide prior to being issued for public comment.

- Proposed Revision 2 to Regulatory Guide 1.126, (DG-1189), "An Acceptable Model and Related Statistical Methods for Analysis of Fuel Densification"

  This Guide describes an analytical model and related assumptions and procedures for predicting the effects of fuel densification in LWR plants. To meet these objectives, this Guide describes statistical methods related to product sampling that will ensure that this and other approved analytical models will adequately describe the effects of densification for each initial core and reload fuel quantity produced.

  Revision 1 to Regulatory Guide 1.126 was issued in 1978. The proposed revision 2 includes more recent information on in-reactor densification. There are no substantive changes to the existing technical guidance.

  The staff plans to issue DG-1189 for public comment and would like to know whether the Committee wants to review this Guide prior to being issued for public comment.

Visit to the US-APWR Simulation Facility

During its June 2008 meeting, Mitsubishi Heavy Industries, LTD (MHI) provided an overview of the US-Advanced Pressurized Water Reactor (US – APWR) design. During that meeting, MHI invited interested ACRS members to visit a simulation facility related to US-APWR in Pittsburgh, Pennsylvania.

The meeting was adjourned at 1:00 p.m. on July 11, 2008.