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February 1 - 3, 2007

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2. **Report on the Safety Aspects of the License Renewal Application for the Oyster Creek Generating Station**, dated February 8, 2007.

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The following letters to Luis A. Reyes, Executive Director for Operations, NRC, from William J. Shack, Chairman, ACRS:

1. **Draft Final Revision 1 to Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants,"** dated February 14, 2007.

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The following memoranda to Luis A. Reyes, Executive Director for Operations, NRC, from, Frank P. Gillespie, Executive Director, ACRS:

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MINUTES OF THE 539th MEETING OF THE
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
February 1 - 3, 2007
ROCKVILLE, MARYLAND

The **539th** meeting of the Advisory Committee on Reactor Safeguards (ACRS) was held in Conference Room 2B3, Two White Flint North Building, Rockville, Maryland, on **February 1 - 3, 2007**. Notice of this meeting was published in the *Federal Register* on **December 29, 2006** (71 FR **78470**) (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 (Member-at-large), Dr. Said Abdel-Khalik, Dr. George E. Apostolakis, Dr. J. Sam Armijo, Dr. Sanjoy Banerjee, Dr. Michael Corradini, Dr. Thomas S. Kress, Mr. Otto L. Maynard, and Dr. Dana A. Powers. For a list of other attendees, see Appendix III.

I. Chairman's Report (Open)

[Note: Mr. Frank P. Gillespie 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. He announced in his opening remarks that the meeting was being conducted in accordance with the provisions of the Federal Advisory Committee Act. In addition, he reviewed the agenda for the meeting 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. He discussed the items of current interest and administrative details for consideration by the full Committee.

II. Final Review of the Power Uprate Application for the Browns Ferry Nuclear Plant, Unit 1

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

Opening and Licensee Presentation

The Committee met with representatives of the Tennessee Valley Authority and the NRC staff to discuss the proposed 5-percent uprate application for Browns Ferry Nuclear Plant, Unit 1. Dr. Bonaca opened the session with a report of the Power Uprate Subcommittee on

January 16-17, 2007. He noted that much of the analyses performed to support this uprate were performed at 120% of the Original Licensed Thermal Power (OLTP). He also invited the members to carefully consider the request for containment overpressure credit because it involves some scenarios that the Committee has not considered before for other plants.

Mr. McGinty, Office of Nuclear Reactor Regulation (NRR), opened the staff presentation with a history of the Browns Ferry (BF) plants, and their current status. He recalled the extended shutdown period, and the program used by the Tennessee Valley Authority (TVA) to restart the 3 BF units. He noted that the original plan was to restart Unit 1 at 120% OLTP, but because of a lack of necessary information to complete the review of the steam dryers, TVA decided to request a 5% uprate for Unit 1. The information for the steam dryer analyses should be provided to the staff by April 2, 2007 and the staff should then proceed to complete its Extended Power Uprate (EPU) review.

Ms. Brown, NRR, described the staff review process for the uprate, which used the guidance in Review Standard (RS) 001, and also followed the plan laid out in the ELTR1 and ELTR2 Topical Reports. The staff used RS-001 to evaluate synergistic effects, safety margin reductions, and operating procedures. This guidance was endorsed by the ACRS in September 2003. She described how the staff evaluates the effects of the EPU on equipment operation and analysis results. She presented example results of the review for both low pressure injections systems and the control rod drive (CRD) systems.

Mr. Bhatagrat, TVA, described the status of the uprate efforts. The work is nearly complete, with the focus of the project shifting to the balance-of-plant (BOP) systems, where testing is in progress. He noted that they were able to move up the refueling outage for Unit 2 so that both plants will not be involved in a restart/startup test program at the same time. At this time, the operations department has full control of the entire plant, and they are proceeding as scheduled.

Mr. Crouch, TVA, noted that after the 5% uprate, Unit 1 will be operating very similarly to Units 2 and 3. Dr. Bonaca asked whether Units 2 and 3 would be replacing their condensate pumps like Unit 1, and Mr. Crouch replied that they would. Mr. Bhagarat commented that they would not be modifying the high-pressure (HP) turbines for Units 2 and 3 until the EPU applications are approved. Dr. Armijo asked about the water chemistry controls for the plants, and Mr. Phillips, TVA, explained that they would eventually be operating with the same chemistry, after the noble metal chemistry is fully implemented.

Mr. Crouch emphasized that TVA has generally used the 120% analyses to support the 105% operating conditions, except for the core analyses, and a few other specialized situations. After implementation of the 105% uprate, Unit 1 will have effectively the same licensing basis as Units 2 and 3. Dr. Bonaca asked about the different fuel in the plants and whether TVA is performing the calculations required by ELTR1. Mr. Crouch replied that they had not done the SAFER/GESTR analyses at 100%. Ms. Brown commented that the staff has issued a letter requiring the performance of a plant-specific core analysis. The staff did a plant-specific review of the core analyses. Mr. Thomas, NRR, explained that the staff did independent calculations of the loss-of-coolant accident (LOCA) scenario for Unit 1. Dr. Bonaca asked whether the staff was happy about the change in the methodology. Mr. Crouch commented that TVA looked at this change in methodology when they uprated Units 2 and 3. Mr. Sieber asked whether TVA or General Electric (GE) had performed the reload analyses, and Mr. Crouch explained that the fuel vendors did the analyses. Dr. Armijo asked what was special about operation at 105% compared to 120% for the core that has been loaded, and Mr. Storey, TVA, replied that they have a special operating strategy for the 105% condition.

Dr. Shack asked when the piping at Units 2 and 3 were replaced, and Mr. Crouch replied that this was done either during the restart efforts, or during subsequent outages.

Mr. Crouch also described the large number of related licensing activities that were needed to support the restart of the plant, and how they related to both license renewal and the uprate. He also described the plant modifications performed to support the uprate. Many of these were done to add margin to plant operations, as well as to support the uprate. Some of these modifications have not yet been performed on Units 2 and 3.

NRC Staff Review

Ms. Brown then described the application for the 105% uprate, which was submitted on September 22, 2006. She also described the staff review process, which built on the 120% review process, and used the guidance in RS-001, ELTR1, and ELTR2. The staff also reviewed a number of independent licensing changes to support the uprate and restart. Almost all of the licensee's analyses to support the 105% uprate were performed at 120% and were found to be bounding for 105%. The staff found that the Unit 2 and 3 flow-accelerated corrosion and stress corrosion cracking programs are applicable to Unit 1.

Dr. Shack asked about extraction steam erosion issues. Mr. Crouch replied that they have replaced all of the susceptible piping and based on the pre-startup inspections, they have confidence that the piping will not drop below minimum wall thicknesses before the next inspection.

Dr. Kress asked what constituted an acceptable margin, and Ms. Brown replied that this meant that the analyses met the acceptance limits.

Dr. Powers asked about the alternate source term (AST), and Ms. Hart, NRR, explained that the AST proposal was reviewed by the staff for 120% well before the uprates were proposed. She reported that the staff verified that none of the assumptions had changed. Dr. Powers noted that the source term varies with the use of high-burnup fuel, and he asked Ms. Hart if the staff had considered this. Ms. Hart replied that she did not have any details of the high burnup source term, but the review did consider the use of both GE and AREVA fuel types.

Ms. Brown described the TVA change in operating strategy, and the extra analyses that the staff asked TVA to perform at 105%. As a result of these analyses, the staff concluded that the analyses performed at 120% envelope and operation at 105%. Dr. Razzaque, NRR, commented that the staff performed independent calculations of the LOCA at 120%, and they believe that this is bounding. Dr. Kress asked why they did not perform a station blackout (SBO) calculation. Mr. Rubin, NRR, explained that SBO is a required calculation but is not a licensing basis calculation. Dr. Bonaca commented that this brings into question whether the SBO and Appendix R calculations are part of the licensing basis. Mr. Lobel, NRR, explained that these calculations are part of the licensing basis, but are not part of the design basis for the plant, and they have different acceptance criteria. Design basis is defined in 10 CFR 50.2, and license basis is defined in part 10 CFR 54.

Ms. Brown noted that the licenses for BF were renewed before the uprate review was completed. As a result, there is a license renewal component in the uprate review. Dr. Corradini asked whether this was done for 105% or 120% operation, and Ms. Brown explained that it was done at 120%. Dr. Bonaca noted that although the Committee may determine that the analyses are acceptable for 105%, it will not conclude that they are acceptable for 120%. Ms. Brown agreed with this comment.

Ms. Brown also described the proposed test program that includes component, system, and integrated testing. This test program is similar to that done for the Unit 3 restart. She also described the detailed tests to be performed at increments between 100% and 120% power, and the steam dryer monitoring that will be performed, which will be similar to the program at Vermont Yankee (VY). The testing program is consistent with Standard Review Plant (SRP) Section 14.2.1, and Appendix L of ELTR1. The staff believes that integrated testing is necessary only for Unit 1.

The Unit 1 steam dryer is similar to ones at Units 2 and 3, so the staff and TVA believe that the Unit 1 steam dryer is acceptable for operation at 105%. Dr. Abdel-khalik asked what sort of program TVA has to monitor low-frequency (<30Hz) vibrations. TVA replied that they have a program to do this, and they will be discussing this with the staff in the spring. Dr. Kress asked what could be seen during walk-downs, and Mr. Fuente, TVA, replied that they are useful to detect unusual vibrations and failure of hangers and fasteners.

Risk Evaluation

Mr. Stutzke, NRR, noted that this is not a risk-informed application, but TVA and the staff did perform risk evaluations of the proposed uprate. Dr. Powers asked why there was any consideration of risk if there was no consideration of the increase in fission product inventory. Mr. Stutzke replied that the increase in inventory risk is directly related to the power level, so the Level 1 evaluation is useful in considering the increase in the level of risk.

He noted that several success criteria have changed for this plant as a result of the uprate and the plant modifications. These relate to CRD flow rates, main steam relief valve (MSRV) operation during anticipated transients without scram (ATWS), and containment overpressure credit. The overpressure credit issue related to a loss of containment integrity caused by pre-existing leaks or failure to achieve containment isolation. This could cause loss of core spray (CS) and residual heat removal (RHR) pumps as well as loss of pump function.

Mr. Stutzke noted that the need for overpressure credit depends on the number of RHR pumps running for suppression pool cooling - credit is only needed for 1 pump operating, or 2 pumps under certain plant conditions. Dr. Armijo asked whether this was at 105% or 120%, and Mr. Stutzke replied that this was done at 120% - no calculations were done at 105%. Dr. Banerjee asked who had done the calculations, and TVA replied that they had been done by one of their consultants. Mr. Anderson, GE, explained that the calculations were similar to those done for VY, starting with the GE base calculations and varying the parameters to see what sort of combinations required overpressure credit.

Mr. Stutzke also noted that credit is always required for SBO, ATWS, and the Appendix R scenario. The Appendix R scenario is the driving scenario. The risk impact of these scenarios is quite small, with a total core damage frequency (CDF) of 1.7E-7, for the assumption of loss of containment integrity. Dr. Powers asked about seismic events, and Mr. Stutzle replied that they only considered internal events. The staff looks at external events qualitatively, and since the licensee did not identify any seismic vulnerabilities, the seismic events are not considered. Dr. Apostolakis noted that the seismic analyses are quite stylized and may not be applicable. Mr. Rubin replied that there may be some coupling, but the initiating earthquake would be quite low in frequency and that it would be comparable in overall risk. Dr. Powers noted that he thought that the seismic studies extant were providing risk values on the order of 1E-5, so he did not understand how this was consistent with the 1.7E-7 value presented. Mr. Rubin replied that safe shutdown earthquake (SSE) is part of the design basis, and the margins analyses show that the equipment is quite robust for larger earthquake. Dr. Powers commented that the staff is looking at the wrong class of accidents, if it does not include seismic considerations, because of the possibility that seismic events may compromise a large amount of equipment.

Mr. Stutzke described the human reliability evaluation, and he noted that TVA used cause-based decision trees with specific causal factors that were judged to be more likely to drive the probability rather than time constraints. They used human cognitive reliability for time sensitive errors, and this approach is consistent with the HRA good practices document. Dr. Apostolakis commented that the staff has never really reviewed this methodology.

All of the affected human failure events pertain to ATWS, and the human failure events (HFE) that became significant as a result of the EPU include controlling level using HPCI/RCIC, and initiation of depressurization. Other HFEs were modified to address EPU impacts. Overall, the influence of these changes has a small effect on risk frequencies. Dr. Bonaca noted that this information was not provided at the subcommittee meeting, and it is important to know how to evaluate this application. Mr. Rubin replied that this is quite an unusual situation, because the HFEs do not significantly affect the risk, but instead, it is the CRD flow rate that is significant.

Dr. Apostolakis asked whether the staff has captured the effects of HFEs in its deterministic evaluations, given that this is not a risk-informed application. Mr. Rubin replied that changes to the HFE are reflected in the design basis analyses, and therefore they are considered by the staff. These HFEs only consider design basis equipment. The PRA looks at a wider range of equipment.

Mr. Stutzke briefly described the staff review of the Unit 1 PRA, and concluded that the staff had not identified any "special circumstances" that rebut the presumption of adequate protection afforded by compliance with the Commission's regulations.

Mr. Crouch clarified an earlier condition regarding hydrogen injection, and he stated that the plant will run at a low level of hydrogen for about 30 days after the injection of the noble metals.

Mr. Walcott discussed the containment overpressure analyses and the ECCS systems involved. He explained that all of the Unit 1 net positive suction head (NPSH) analyses were performed at 120%, and this bounds operation at 105%. Four events need containment overpressure credit (COP): LOCA, ATWS, SBO, and Appendix R events.

Mr. Walcott presented comparisons of the amount of COP required for other BWR EPUs to that requested for BF1. He then presented the results of several events showing the amount of available pressure and the pump NPSH required. Dr. Abdel-Khalik asked whether they had done these at 105%, and Mr. Wolcott replied that they had not, but a reduction in the power level would affect both the required and available pressures, equally. He also showed how the available containment pressure changed with changes to various analytical assumptions so that the actual amount of COP that is available is greater than the minimum value that results from using the staff-required assumptions. He also showed how the available and required pressure varies when realistic parameters are used for the calculations.

Dr. Banerjee and Dr. Corradini asked about the effects of energy partition on the results, and GE explained that much of the effect is due to differences in the assumption about the location of non-condensables in the containment. Mr. Lobel explained that this analysis is an integrated containment model of the LOCA scenario, where the energy partition is determined by the details of the model and the flow paths. He noted that the calculations are biased towards either higher or lower pressure depending on the intended use of the results. Also, the temperature of the pool is more important for NPSH calculations than the containment overpressure. Dr. Banerjee expressed some concern about the effects of the model itself, rather than the initial assumptions, and he wondered what would be the effect if the energy partition function varied.

Mr. Wolcott also showed the results of additional calculations using "realistic parameters" and how they do not significantly affect the minimum pressure available, but do reduce the pressure required so that essentially no COP is required. Dr. Abdel-Khalik asked how the realistic analysis could be lower than the minimum pressure results. Mr. Wolcott explained that this arises out of changes to the pool temperature and its effect on the relative humidity. Several parameters offset one another, and it is purely coincidence that the two curves overlay one another. This provoked a lively discussion about the incongruity of having the "realistic value" lower than the "minimum value." Dr. Banerjee thought that this resulted from the complexities of the analysis methodology, and Mr. Lobel tried to explain why it was physically reasonable, but the members continued to express some concern about this description.

Mr. Wolcott then described the scenario for the Appendix R case, and he showed again that both the available and required COP pressures drop for the analyses with realistic parameters. The overall margin available for the realistic case increases. He noted that they do not claim that the results are entirely accurate, but the difference between the two curves, which shows the margin available, is demonstrated.

Several members expressed a desire to understand the physical phenomena that change with the use of realistic analyses, and how they affect the results.

Mr. Wolcott completed his presentation with a description of the risk analysis that they performed, which followed some guidance that arose from the Vermont Yankee EPU. He noted that there is a very small risk increase for LOCA, ATWS, and SBO CDF, and large early release fraction (LERF) related to dependence on COP. The risk increases are also well within the acceptance guidelines for CDF and LERF.

Containment Systems Review

Mr. Lobel commented that the main issues that arose from this review related to (1) the need for pump cavitation credit, (2) behavior of the drywell fan coolers, and (3) the pump flows used.

TVA performed tests of the pumps to verify that they could operate satisfactorily in cavitation mode, and the pump vendor confirmed this assessment.

Dr. Banerjee asked about vortexing into the strainers. Mr. Ebberly, TVA, explained that they had evaluated the Froude number at the strainers and determined that vortexing would not occur. The flow rate into the strainers will not support vortices.

The staff asked TVA a number of questions about the drywell fan coolers and the pump flows. The staff determined that the operating procedures already contained appropriate guidance and the proposed design basis was acceptable. Dr. Abdel-Khalik asked whether the NPSH calculation took into account the change in the elevation of the free surface due to vortexing. Mr. Ebberly replied that they do not anticipate vortexing, and therefore do not consider it.

Mr. Dyer, NRR, closed the presentation by thanking the ACRS for accelerating its schedule to accommodate the staff and TVA. He thought that it will be good to allow some time before the plant start up, and he understands that there are still a number of issues to be addressed for 120%. Many of the issues that the Committee has identified are common to other plants, and he noted that the staff is struggling to deal with them.

III. Final Review of the License Renewal Application for the Oyster Creek Generating Station (Open)

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

The Committee met with representatives of the NRC staff and its contractor Sandia National Laboratories (SNL), members of the public, and AmerGen Energy Company, LLC (AmerGen) and its contractors to review the license renewal application (LRA) for the Oyster Creek Generating Station (OCGS) and the updated Safety Evaluation Report (SER) prepared by the NRC staff. The applicant, AmerGen, has requested approval for continued operation for a period of 20 years beyond the current license expiration date of April 9, 2009.

The presentations focused on: questions which were raised during the previous license renewal Subcommittee meetings; the safety evaluation report that accepted a structural analysis performed by the applicant to demonstrate acceptability of the containment in the degraded condition; the sources of water leakage that caused the degraded condition of the drywell; and a summary of the license renewal application.

AmerGen representatives presented a summary of the corrosion of the drywell shell. They described water leakage from the reactor cavity liner that accumulated in the sand bed region and corroded the exterior surface of the drywell shell. The corrective actions taken include preventing the water from entering the sand bed region, removing the sand and coating the exterior of the drywell shell with an epoxy coating, and performing various inspections of the drywell shell. During the 2006 refueling outage, the applicant inspected the drywell shell to determine if the corrective actions had been effective. They found low leakage from the reactor cavity liner, no water in the sand bed, the epoxy coating in all the bays were in good condition, and that no further corrosion was occurring in the lower or upper regions of the drywell.

AmerGen's overall conclusions were that the corrective actions to mitigate the drywell shell corrosion has been effective, the corrosion in the embedded portion of the drywell shell is not significant, the drywell shell meets code safety margins, and there is an effective aging management program in place to ensure continued safe operation.

AmerGen responded to issues that were raised during the January 18, 2007 Oyster Creek License Renewal Subcommittee Meeting. The issues covered were 1) the acceptability of using a capacity reduction factor in the structural analysis of the drywell when no internal load is present, 2) the use of a modern 3-D (dimensional) finite-element model of the drywell shell, 3) eliminating leakage in the reactor cavity liner, 4) more aggressive monitoring of drywell shell thickness, and 5) corrective actions to eliminate the water on the drywell shell.

AmerGen presented the license renewal summary. The LRA was submitted on July 22, 2005 using the NEI 95-20 Revision 6 standard format. It was prepared using the January 2005 draft versions of NUREG-1800 (Standard Review Plan) and NUREG-1801 (Generic Aging Lessons Learned Report). The Aging Management Programs include 50 programs consistent with the GALL Report and 7 plant-specific programs. There were 65 license renewal commitments which were placed in the applicants commitment tracking system.

The staff presented information regarding the drywell shell and discussed the License Renewal Activities that have occurred. The staff clarified that the 1992 GE analysis is the current analysis of record and that the analysis performed by Sandia National Laboratories (SNL) in 2006 was confirmatory. The key difference between these analyses is the inclusion of hoop tensile stresses. The staff concluded that if the SNL analysis included these hoop tensile stresses, the minimum thickness results would be similar to the GE analysis.

The staff provided an overview of the License Renewal Process. The draft SER was issued on August 18, 2006 with five open items, no Confirmatory items and three license conditions. An updated SER was issued December 29, 2006 which closed the five open items and included additional commitments from the applicant. The Final SER will be issued after the ACRS letter is received and will include additional applicant commitments, two license conditions and will discuss the confirmatory analysis from SNL.

The OCGS application either demonstrates consistency with the Generic Aging Lessons Learned (GALL) Report or documents deviations from the approaches specified in the GALL Report. The staff reviewed this application in accordance with NUREG-1800, the "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants."

The applicant identified those structure systems components (SSCs) that fall within the scope of license renewal. For these SSCs, the applicant performed a comprehensive aging management review. Based on the results of this review, the applicant will implement 57 AMPs for license renewal including existing, enhanced, and new programs. In the SER, the staff concludes that the applicant has appropriately identified SSCs within the scope of license renewal and that the AMPs described by the applicant are appropriate and sufficient to manage aging of long-lived passive components that are within the scope of license renewal.

The staff conducted inspections and an audit of the license renewal application. The purpose of the inspections was to verify that the scoping and screening methodologies are consistent with the regulations and are adequately reflected in the application. The audit confirmed the appropriateness of the AMPs and the aging management reviews. Based on the inspections and audit, the staff concluded that these programs are consistent with the descriptions contained in the OCGS license renewal application. The staff also concluded that the existing programs, to be credited as AMPs for license renewal, are generally functioning well and that the applicant has established an implementation plan in its commitment tracking system to ensure timely completion of the license renewal commitments.

The applicant identified those systems and components requiring Time Limited Aging Analyses (TLAAs) and reevaluated them for 20 more years of operation. Affected TLAAs include those associated with neutron embrittlement, metal fatigue, irradiation-assisted stress corrosion cracking, environmental qualification of electrical equipment, and stress relaxation of hold-down bolts. The staff concluded that the applicant has provided an adequate list of TLAAs. Further, the staff concluded that in all cases the applicant has met the requirements of the license renewal rule by demonstrating that the TLAAs will remain valid for the period of extended operation, or that the TLAAs have been projected to the end of the period of extended operation, or that the aging effects will be adequately managed for the period of extended operation.

Members of the public provided their concerns regarding the drywell liner, the analysis methods used to evaluate the drywell liner, and the adequacy of the inspection data used in the analyses.

The ACRS members received a letter from Jon S. Corzine, Governor of the State of New Jersey (NJ), inviting the Committee to tour OCGS and hold its public meeting in NJ to facilitate public attendance. The ACRS members also received a letter from Senator Frank Lautenberg (NJ), Senator Robert Menendez (NJ), Congressman Christopher Smith (NJ), and Congressman Jim Saxton (NJ) asking the Committee to ensure that the safety issues regarding the drywell are fully resolved before it makes any decisions regarding the OCGS license renewal application. The NRC is in the process of responding to these letters.

Committee Action

The Committee issued a report to the NRC Chairman on this matter dated February 8, 2007, recommending that the application for license renewal for OCGS be approved with the incorporation of certain license conditions. These license conditions are (1) to increase the frequency of the drywell inspections and to monitor the two drywell trenches to ensure that the sources of water are identified and eliminated; (2) to ensure that the applicant fulfills its commitment to perform an engineering study prior to the period of extended operation to identify options to eliminate or reduce the leakage in the OCGS refueling cavity liner; and (3) to ensure that the applicant fulfills its commitment to perform a three dimensional finite-element analysis of the drywell shell prior to entering the period of extended operation.

IV. Development of the TRACE Thermal-Hydraulic Code

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

The Committee met with representatives of the NRC staff concerning the development of the TRACE thermal-hydraulic system analysis code. Dr. Banerjee recalled that the Thermal-Hydraulic Phenomena Subcommittee met in December 2006, to discuss TRACE development. Dr. Bajorek, RES, opened the meeting with a recapitulation of the code history, which started in 1998 when NRC began to consolidate the capabilities of four separate codes into one platform.

At the end of December 2006, the Office of Nuclear Regulatory Research (RES) issued version 5.0 of the code to staff for its use in licensing. He noted that it is important to actually start using TRACE.

Dr. Abdel-Khalik asked whether there was an adequate user manual to make good use of the code, and Dr. Bajorek replied that they do have such a manual. The manual is up-to-date and ready to use, and a user should be able to use it to build models. Dr. Shack asked whether there are staff assigned to provide code support, and Dr. Bajorek replied that people have been assigned to this function. Now they can complete the rest of the documentation, and move forward with its use. Dr. Corradini asked about the theory manual, and Dr. Bajorek explained that this manual is still being revised, because of the large number of structural changes being made to the code.

Dr. Banerjee asked why RES had invested in the development of new models for TRACE, rather than just using proven models from other codes. Dr. Bajorek replied that many of these decisions were made based on the amount of understanding that was available, and the decision was made back in 1998 that the code would be founded on TRAC-PF1/MOD2. Neither TRAC nor RELAP was considered to be state-of-the-art at the time, and one code was picked to be the starting point. Since then, as good models have been identified, they have incorporated them into the code. Much effort was spent in the intervening years developing model improvements for reactors such as the ACR-700, which never materialized. The code developers also spent considerable amount of time dealing with emerging non-code issues.

In addition, as the assessment base expanded, it became clear that the models in the code needed to be revised because the predictions were sometimes conservative and sometimes non-conservative. In general, the small-break loss-of-coolant accident (SBLOCA) results have been better than the large-break loss-of-coolant accident (LBLOCA) cases. He noted that TRACE is supposed to be able to model a wider range of conditions than any of its predecessors, with a wider range of phenomena, so this should not be a surprise.

Dr. Banerjee asked about how well TRACE models containment, and Dr. Bajorek explained that TRACE is expected to model the reactor coolant system alone, but it has been coupled to containment codes such as CONTAIN. For the ESBWR, they are trying to develop an integrated model using only TRACE. TRACE should also be ready to model the EPR, with minor modifications. They are looking at assessment against reflux condensation tests right now to verify this ability.

Dr. Bajorek presented the results of some assessment cases, which show good agreement with data. Dr. Shack asked about run times, and Dr. Bajorek replied that for certain integral tests, especially for advanced reactors such as the AP1000, the code is having trouble, and run times are quite long.

The TRACE assessment matrix is based on phenomena identification and ranking tables (PIRT) for LBLOCA and SBLOCA in conventional light water reactors (LWR). Separate PIRTs are used to augment these for new plants. There are more than 500 individual simulations in the assessment base. RES believes that the assessment matrix is consistent with recommendations from the Committee on the Safety of Nuclear installations (CSNI) , and is sufficient for a code scaling applicability and uncertainty (CSAU) application to most plant types. Dr. Powers asked how TRACE compared to CATHARE, and Dr. Bajorek replied that some of the TRACE models are close to the ones in CATHARE. RES is aware of that code, and how it works, and they have pushed some of their models to be more like CATHARE. Dr. Powers asked whether it was useful to have a multiplicity of codes, and Dr. Bajorek replied that he thought that it is better to have more codes. Comparison exercises with different codes have been quite useful to code developers. Dr. Banerjee commented that CATHARE seems to be able to perform 3-D calculations when they are needed, and 1-D when they are not, while TRACE seems to try to use a 1-D model for all situations.

Dr. Bajorek noted that the TRACE Theory Manual documentation has been slow, but it is now the focus of attention with the internal release of TRACE V5.0. The V5.0 executable and the user Manual were released in December 2006, and the assessment report should be complete by April 2007. The theory manual should be updated by June 2007, and a supplement with information that is relevant to code developers should be ready in August 2007. The ESBWR applicability report, which will provide guidance to users for ESBWR analyses, will be issued in November 2007.

Once the documentation is complete, a peer review will be initiated to provide critical reviews of the code, including the conservation equations, the numerical solutions, the assessment matrix, and the special features. They will not perform a line-by-line review of the code. It is not cost effective, and does not identify problems as readily as identification by users. Dr. Banerjee expressed some concern about how these sorts of fixes get implemented, in a piecemeal fashion, which can lead to obscurity in the code structure.

Dr. Powers asked about how the peer review would be conducted, and Dr. Bajorek explained that they would expect to make significant use of the results. The review will consider how the code will be applied, and will focus on issues that relate to the application, and not just on academic correctness.

User support will be an important component in integrating TRACE into the NRC regulatory process. The support includes a graphical user interface, plant input deck generation, training workshops, and expert support. They need to convince the people who are currently comfortable using RELAP to use TRACE.

Dr. Banerjee asked about converting old RELAP decks to TRACE, and Dr. Bajorek explained that SNAP can only convert 90% of a RELAP deck. The rest needs to be converted by hand. This can be quite frustrating for a user. The parts that do not convert include trips and signals, and logic. The hardware translates reasonably well. Regarding TRAC decks, TRACE can run all TRAC-P and TRAC-B input decks with little or no modification. Input decks are available for most plant types. RES plans to complete the initial set of input deck updates by August 2007. This will include Browns Ferry, RESAR 412, HB Robinson, and Calvert Cliffs. Within 6 to 24 months, input models for a wide variety of plant types should be available. Dr. Banerjee asked how much effort is going into this, and Dr. Barjorek replied that it takes about 2-3 staff-months per plant.

Dr. Bajorek briefly described plans to address issues raised in an anonymous letter. Upon consideration of the comments received by the Committee, RES plans to have Dr. Mahaffy, Pennsylvania State University, perform a rigorous evaluation of the comments and formally document them. The ACRS will be informed of the results of this evaluation. Dr. Banerjee commented that although the members may have agreed with this conclusion, the case was not properly made, and he was concerned that in two months, nothing more has been done. Dr. Bajorek replied that they will have Dr. Mahaffy resolve the comment, and revise the description of these models in the theory manual. This should close the issue.

Dr. Bajorek also discussed an issue regarding ranging of Pi-groups for scaling experimental facilities. It arose because of a decision to use a particular limit for the Pi-groups, which did not have any defined basis. As a result, the staff and its contractors developed a figure-of-merit approach to determining the appropriate Pi-group range. Dr. Bajorek described the method, and concluded that it is important to not used fixed values for the Pi-group range, but instead should be tailored to the parameter of interest. The members commented that this is important, because it points to the need for the facility to produce the type of data that is important to the analysis, so that the codes can be assessed against the proper figure-of-merit, and the results can be trusted. Dr. Banerjee recalled that the members had suggested that this approach be documented so that it could see a wider application, and Dr. Bajorek explained that this was being done by Dr. diMarzo.

V. <u>Proposed Revision to 10 CFR 50.46 LOCA Criteria for Fuel Cladding Materials</u>

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

The Committee met with representatives of the NRC staff and its contractor, Argonne National Laboratory (ANL), and the Electric Power Research Institute (EPRI) regarding the development of the technical basis for a revision to the fuel cladding embrittlement criteria in 10 CFR 50.46. Dr. Armijo opened this session with a brief report on the meeting of the Materials, Metallurgy, and Fuels Subcommittee on January 19, 2007. At that meeting, the staff presented the results of its high burnup fuel research program that relate to revising the embrittlement criteria. He noted that this topic was covered in some depth, and they heard presentations from the industry as well as the staff. He commented that this work is quite admirable, but the industry is reluctant to use this information because they believe that it is not yet complete, and the technical issues are not settled.

Ms. Uhle, RES, opened the staff presentation and reviewed the history of the 10.46 acceptance criteria. They believe that this work is ready to move forward to rulemaking. This would enable the development and use of new materials and would reduce regulatory burden. Dr. Armijo asked how RES intends to proceed. Ms. Uhle replied that a regulatory guide is being prepared, and a proposed rule should be issued in January 2009. Based on the current understanding of this issue, the staff believes that this should be a high-priority rulemaking. Dr. Landry, NRR, commented that NRR would like to proceed to rulemaking in an orderly fashion, with criteria that will stand for a long period of time, and not need to be revised. They would like to move in the direction of a more "performance-based" rule and leave the details in regulatory guidance, but they are not yet sure how this will eventually end.

Mr. Meyer, RES, described the high burnup fuel program and noted the support from the Kurchatov Institute and the Russian fuel vendor, Tavel, that has proven to be important to understanding the phenomena.

The focus of the work has been the ductility of the fuel cladding. He described the metallurgical changes that occur in the fuel during a LOCA scenario, which includes phase changes, increased corrosion/oxidation, and an overall loss of ductility. The current embrittlement criteria in 50.46 include a temperature limit and an oxidation limit that involves calculation of the oxidation on the outside of the rods, as well as that on the inside in any balloon regions of fuel. Information Notice 98-29 clarified the staff position that the oxidation limit includes both the oxidation during a transient and any pre-existing corrosion.

Dr. Meyer described the various alloys that were tested in the high-burnup fuel program, which include Zircaloy-4 (Zry-4), ZIRLO, M5, and E110. These materials were tested in a furnace that used external heaters and allowed steam to pass over the samples. Dr. Powers asked whether there was any significance to the fact that during the transient the heat would come from outside the rod, while in the experiment, the heat comes from the outside. Dr. Meyer replied that this is why they performed both single-sided tests and double-sided tests.

Dr. Meyer described the test method, which involves ring compression tests of small samples, to determine when the cladding loses ductility during an accident. Cladding ductility is the key to ensuring that the core will remain in a coolable geometry following a loss-of-coolant accident. They tested a number of actual high burnup fuel rod samples from power plants and have plans to test additional rods that use the newer types of cladding. However, they were not able to perform those tests in time to support this NUREG report. Dr. Meyer noted that it has always been the plan to examine unirradiated ZIRLO and M5 rods, and both irradiated and unirradiated Zry rods, and use those results to infer the behavior of the irradiated ZIRLO and M5.

Dr. Meyer described in detail the change in morphology of the cladding during a LOCA transient, and he pointed out that the most important phenomenon that occurs is the diffusion of oxygen that creates a brittle oxygen-stabilized alpha phase. He also noted that for this program, they have shifted from using the Baker-Just oxidation correlation to the Cathcart-Pawel (C-P) correlation. Dr. Armijo asked whether C-P has been shown to be valid for all zirconium alloys, and Dr. Meyer replied that they have found that at lower temperatures, some alloys have lower oxidation. Ms. Uhle commented that when the rule is eventually written, the staff plans to include a requirement that an appropriate correlation be used for the material proposed. Dr. Meyer believes that this method will work for all Zr/Sn/Nb alloys with Sn-Nb values in the 1% range.

With regard to burnup effects, they observed that the major effect of burnup is the consequence of hydrogen absorption during normal operation. High-burnup Zry embrittles at a much lower equivalent cladding reacted (ECR) than fresh material. This is due to the hydrogen effect on the oxygen solubility and diffusion rates in the metal. They have taken fresh Zry materials and pre-hydrided them to demonstrate this phenomenon.

Dr. Armijo asked whether the hydrogen contributed directly to embrittlement, and Dr. Billone, ANL, commented that it has some effect, and the cooling rate does have some effect, because it freezes in the hydrogen. This is accounted for in the F-factor.

Dr. Meyer also noted another phenomenon related to the impurity levels that were identified from testing of E110, which is very similar to M5, but which behaves much differently. It is believed that this arises from the Zr ingot fabrication process. E110 is produced from a very pure electro-refined ingot, while M5 is produced from a Kroll-process ingot.

He pointed out that there are two sources of oxygen during a transient - the expected source on the outside of the cladding and the UO_2 fuel that is bonded to the cladding on the inside of the fuel rod.

The proposed method would involve retaining the existing temperature limit of 2200F, while revising the oxidation limits to values that would be determined for each material from specific tests. The test material would have to be prototypical material, with respect to fabrication and surface condition. It would also include an allowance for normal corrosion that would be multiplied by an "F-factor" of 1.2 to account for cooling rate effects. The F-factor value incorporates information from the experiments and expert judgment. There would also be a limit on the amount of time that the fuel would be allowed to remain above the measured breakaway oxidation time.

Dr. Meyer presented examples of how the method would be applied to several different alloys and showed how some modern alloys would fare well with the new criteria, while other alloys, which are not used, but which has some similarities, would be screened out.

Dr. Ozer, EPRI, commented that the industry fully supports the ongoing high burnup fuel program, but it does not believe that the data obtained thus far indicates the presence of a public safety issue. The revisions proposed by RES are premature and not adequately supported by data. The evidence does not support use of 2-sided oxidation away from the balloon region, and the bounding approach will have a significant negative impact on the industry with little or no safety benefit.

Dr. Powers expressed skepticism about the statement that there is no safety issue, given the evidence that high burnup fuel may shatter during a LOCA. Dr. Ozer replied that the experimental evidence supports the view that even brittle material will withstand quench and post-LOCA impact forces. He presented data from operating reactor calculations to show that high burnup fuel will not be operated under conditions that would even approach 2200F during a LOCA. Mr. Dunn, Areva, explained that during a LOCA, the high burnup fuel which operated at a much lower peaking factor, will experience much lower temperatures, because both the decay heat and the stored energy will be much lower.

Dr. Powers asked what data was available to support the claim about loads, and Anatech commented that this comes from Japanese data where the rods were restrained in tension during the event, to see whether the rods will fail. Even 17% equivalent clad reacted (ECR) fuel does not fail, and they have discovered that much higher levels of oxidation are required to cause failure. Dr. Powers was concerned that these experiments might not be inclusive of all of the stresses that might occur, and Dr. Ozer replied that this data provide that sort of indication. Dr. Armijo commented that the high burnup focus seems to be the source of the industry concerns. The industry does not believe that there is sufficient data to support the 1.2 F-factor. Dr. Uhle noted that the F-factor would be determined for each cladding material, and this is not the time to discuss this factor.

Dr. Dunn commented that Areva believes that this method should have a well-established basis, but they are not quite there. He noted that since the last time this was discussed with the ACRS, two new phenomena have been identified, and they are concerned that this effort is moving too fast.

Mr. Ozer noted that the F-factor is a complicated function of hydrogen content, cladding design, and accident time-temperature history. It is also not appropriate for BWRs, where the hydrogen content is more important than oxide thickness. It is unclear how to address these variables through a single factor, or how to apply a single factor to a wide variety of LOCA scenarios.

Regarding the testing that was proposed, they do not believe that the quench temperatures that are proposed are appropriate, either, because predicted quench temperatures in PWRs are lower than the 800C temperature used in the methodology. Dr. Billone noted that this data is from Commissariat Energy Atomique (CEA) experiments, but Argonne National Laboratory (ANL) has not observed this sensitivity.

Mr. Dunn commented that the industry really wants to wait till the results of the high burnup ZIRLO and M5 tests are complete to fill in the rest of the data to support this proposal.

Dr. Ozer pointed to some of the results of the ANL tests where there is no significant internal cladding oxygen pickup due to fuel bonding. Dr. Armijo replied that this phenomenon is known to exist in BWRs, though. Dr. Powers pointed out that the only way to verify this is to actually look at irradiated fuel, and Dr. Ozer agreed. This is what the industry wants to do, to finish the rest of the test series. They also think that there will be new information coming out of other labs, such as Halden, and this data needs to be considered.

Finally, Dr. Ozer presented a summary of the effects of this research on the industry, and he noted that there is no urgency.

VI. Draft Final Revision 1 of Regulatory Guide 1.189 (DG-1170), "Fire Protection for Nuclear Power Plants," (Open)

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

The Committee met with representatives of the NRC staff to discuss the draft final Revision 1 of Regulatory Guide (RG) 1.189, and resolution of public comments. In a letter dated November 17, 2006, the ACRS requested the opportunity to review the draft final version of this Guide after the resolution of public comments. The RG provides comprehensive guidance on the scope and depth of fire protection programs that the staff would consider acceptable for the existing and new reactor plants. It was issued for public comments in September 2006. Ninety-five comments were received from the Nuclear Energy Institute (NEI). The staff agreed with 67 of these comments and incorporated them in the final draft. In addition, the staff addressed 16 NEI comments received on previous versions of the draft guide. The staff noted that this technical guidance has also been incorporated into Standard Review Plan (SRP) Section 9.4.1, "Fire Protection Program."

The staff discussed the resolution of the following NEI comments:

NEI commented that the draft guide promulgates new staff positions that are backfit to the industry. The staff's response is that the RG promulgates one of the acceptable methods of meeting the regulatory requirements and that the licensees/applicants may propose alternative methods for showing compliance to a regulation. Also when alternative approaches are proposed, the staff reviews the application against the licensing bases of the plant and not the

RG. The Committee to Review Generic Requirements agreed with the staff's position, and no backfit analysis was required.

NEI commented that the RG should not be issued because the Commission did not authorize the issuance of a generic letter regarding analyses of multiple spurious actuations in case of a fire. In response the staff deleted the specific guidance on the spurious actuation analysis requirements from the RG.

Public Law 104-113 requires the use of available consensus standards in governmental rulemaking. NEI commented that equivalent guidance for this RG exists in NFPA-804 (Standard for Fire Protection for Advanced light Water Reactor Electric Generating Plants) and NEI 00-01 (Guidance for Post-Fire Safe-Shutdown Circuit Analysis) such that the RG could be replaced with portions of these documents. In response, the staff noted that specific endorsement of an NFPA standard has already been made via the rulemaking process, guidance on acceptable use of NEI 00-01 has been issued in a generic communication, and the regulatory review of NFPA-804 is ongoing. The staff's position is that the issuance of the RG does not prevent future endorsement of such industry standards.

NEI commented that industry would like to have credit for operator manual action to achieve and maintain post-fire safe shutdown in lieu of the separation required under Section III.G.2 of Appendix R. The RG clarifies that such credit may not be allowed as operator manual action does not provide the same level of protection provided by the separation requirements of Section III.G. 2. The staff also addressed the industry's question on the need for detection and suppression capabilities with the use of operator manual action. The staff noted that fire detection and suppression are essential elements of the defense-in-depth requirements of Appendix R, and the use of operator manual action as a substitute for separation does not obviate the detection and automatic suppression requirements.

NEI commented that automatic suppression in the peripheral rooms and smoke detectors in cabinets for the control room complex should be deleted from the guidance. The staff noted that automatic suppression may be required in the rooms if separation by a three hour barrier between the redundant trains is not provided. Also, cabinet detectors provide earlier warning and an exact location of the fire, and NFPA-804 recommends having them.

NEI asked for removal of the guidance that stated minimal reliance be placed on operator manual actions and alternative/dedicated shutdown systems for new reactors. The comments also stated that similar guidance on minimal reliance of electrical raceway fire barrier system be deleted. The staff pointed out that this guidance is appropriate for new plants and is consistent with the Commission's concept of enhanced fire protection for new reactors.

Dr. Apostolakis asked if the staff is planning to codify a requirement for a detailed fire protection PRA in the new reactor licensing process. The staff indicated that the requirement for a fire PRA is optional. The new reactor applicant must submit a plant specific version of the fire PRA if it references a certified design approved by the NRC that used a fire PRA. Also, the staff noted new reactor designs are risk informed, and the risk values are usually much lower than operating reactors. Dr. Apostolakis noted the benefits associated with the PRA approach.

Committee Action:

The Committee issued a letter to the Executive Director for Operations on this matter dated February 14, 2007, recommending that Revision 1 to Regulatory Guide 1.189 be issued.

VII. Wolf Creek Pressurizer Weld Flaws

[Note: Mr. Gary Hammer 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 describe the Wolf Creek pressurizer weld flaws discovered during an October 2006 inservice inspection. The staff described the nature of the large circumferential flaws in three different locations. The pressurizer surgeline nozzle contained three flaws: (1) 4 inches long and 31% throughwall (TW); (2) 2.2" long and 25% TW; and (3) 0.8" long at the inner surface. The pressurizer relief valve nozzle contained a circumferential flaw that was 7.7" long and 26% TW. The pressurizer safety nozzle contained a circumferential flaw that was 2.5" long and 23% TW.

The staff presented its evaluation of the significance of these flaws and the safety implications to other plants with similar welds. The staff described their estimates of how long it would take for such flaws to begin leaking and how long it would take for them to rupture. For some of the analyses, the staff found that such cracks may rupture at the same time they begin to leak. The staff also described significant uncertainties in the analyses which may dominate any potential sources of conservatism. As a result of these analyses, the staff has determined that currently scheduled inspections or mitigations of these welds need to be accelerated for some plants. The staff was also concerned that a more refined first-of-a-kind analysis intended to better characterize the time between the onset of leakage and pipe rupture would not reduce uncertainties in the modeling, the input assumptions, or the results. The staff was also concerned with the time frame needed to complete this kind of analysis.

NEI's presentation provided the basis for their position that the currently scheduled inspections or mitigations of these welds do not need to be accelerated. They stated that the Wolf Creek inspection results are not consistent with other experience worldwide and that it is safe to operate plants without interruption until their next scheduled refueling outage. They understand the reasons for the staff's concerns but noted that the staff's analysis is extremely conservative. They stated that the time between the occurrence of any leakage and pipe rupture allows plant personnel to take preventive actions. They also stated that advanced non-linear finite element modeling analyses are being pursued to provide more detailed calculations of the time interval between the onset of leakage and pipe rupture.

Committee Action

This was an information briefing. No Committee action was necessary. The Committee plans to review the technical basis associated with the proposed NRC staff action for dealing with dissimilar metal butt weld issues during its March 8-10, 2007 meeting.

VIII. Proposed Revisions to Regulatory Guides and Standard Review Plan Sections in Support of New Reactor Licensing (Open)

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

The Committee discussed "high-priority" SRP Sections that are being revised or developed in support of new reactor licensing. The Committee identified eleven SRP Sections that it decided not to review. The Committee's decision is documented in a memorandum dated February 6, 2007, from Frank P. Gillespie, ACRS Executive Director to Luis A. Reyes, NRC Executive Director for Operations. The Committee noted that it is awaiting receipt of additional high priority SRP Sections from the staff.

Committee Action

The Committee plans to conduct an accelerated review of all Regulatory Guides and SRP Sections that it determines warrant ACRS review.

IX. Subcommittee Report on Reliability and Probabilistic Risk Assessment

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

The Chairman of the Reliability and Probabilistic Risk Assessment (PRA) Subcommittee provided a report to the Committee summarizing the results of the December 14-15, 2006 meeting with the NRC staff and representatives of GE to discuss the PRA for the Economic Simplified Boiling Water Reactor that is in the design certification process. During the meeting, the Subcommittee reviewed several topics identified at a prior meeting, including the dominant accident sequences, the common cause failure method, the effects of thermal-hydraulic uncertainties on the PRA, the regulatory treatment of non-safety systems, and staff requests for additional information. The Subcommittee raised several issues to discuss at future meetings, and decided that no interim letter was necessary at this time. The next Subcommittee meeting will focus on the effects of thermal-hydraulic uncertainties on the PRA, the Level 2 PRA, and severe accident phenomena.

IX. Executive Session (Open)

[Note: Mr. Frank P. Gillespie was the Designated Federal Official for this portion of the meeting.]

A. RECONCILIATION OF ACRS COMMENTS AND RECOMMENDATIONS/EDO COMMITMENTS

The Committee discussed the response from the NRC Executive Director of Operations (EDO) to ACRS comments and recommendations included in recent ACRS reports:

- The Committee considered the EDO's response of January 11, 2007, to comments and recommendations included in the December 12, 2006, ACRS report on draft final Regulatory Guide DG-1145, "Combined License Applications for Nuclear Power Plants (LWR Edition)." The Committee decided that it was not satisfied with the EDO's response related to the Committee's recommendation that "the proposed final rule, 10 CFR Part 52, should include the requirements that a PRA be submitted with the design certification application and that a plant-specific PRA be submitted with the combined license (COL) application." The EDO's response articulated the staff's basis for deleting these requirements from the draft final Part 52 rule and stated that the Commission will decide on this matter when it votes on the final rule. The Committee reiterates its previous position with regard to including a requirement in 10 CFR Part 52 for submitting PRAs to the staff.

  The staff committed to inform the ACRS of any significant changes to the final regulatory guide prior to publication.

- The Committee considered the EDO's response of January 19, 2007, to comments and recommendations included in the November 16, 2006 ACRS report on the proposed rulemaking to modify 10 CFR 50.46, "Risk-informed Changes to Loss-of-Coolant Accident Technical Requirements." The Committee decided that it was satisfied with the EDO's response.

  The staff committed to inform the Commission of the impact of the Committee's recommendations on its resources and schedule.

- The Committee considered the EDO's response of January 19, 2007, to comments and recommendations included in the December 15, 2006 ACRS letter on the proposed revision to SRP Section 13.3, "Emergency Planning." The Committee decided that it was satisfied with the EDO's response.

OTHER RELATED ACTIVITIES OF THE COMMITTEE

During the period from December 9, 2006, through January 31, 2007, the following Subcommittee meetings were held:

- Reliability and Probabilistic Risk Assessment - December 14-15, 2006

The Subcommittee reviewed the PRA for the Economic Simplified Boiling Water Reactor.

- Power Uprates - January 16-17, 2007

The Subcommittee discussed the proposed 5-percent power uprate for the Browns Ferry Nuclear Plant, Unit 1.

- Plant License Renewal - January 18, 2007

The Subcommittee reviewed the license renewal application for the Oyster Creek Generating Station and the associated updated Safety Evaluation Report prepared by the NRC staff.

- Materials, Metallurgy, and Reactor Fuels - January 19, 2007

The Subcommittee discussed the proposed technical basis for revising the embrittlement criteria in 10 CFR 50.46.

- Planning and Procedures - January 31, 2007

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

LIST OF MATTERS FOR THE ATTENTION OF THE EDO

- The Committee would like to be kept informed of any significant changes made to the SRP Sections, prior to issuing them in final form, listed in the February 6, 2007 memorandum from Frank P. Gillespie, Executive Director, ACRS, to Luis A. Reyes, Executive Director for Operations, NRC.

- The Committee is awaiting receipt of additional high priority SRP Sections from the staff.

- The Committee plans to review the draft final version of Generic Letter 2007-XX, “Managing Gas Intrusion in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,” during a future meeting.

- The Committee would like to be briefed by the staff on the results of the 3-dimensional finite element analysis of the Oyster Creek Generating Station drywell shell.

- The Committee plans to review the extended power uprate applications for Browns Ferry, Units 1, 2, and 3 during a future meeting.

- The Committee plans to review the technical basis associated with the proposed NRC staff actions for dealing with the dissimilar metal butt weld issue during its March 8-10, 2007 meeting.

- The Committee stated that granting of containment overpressure credit during long-term LOCA and 10 CFR Part 50 Appendix R fire scenarios at 120-percent of the original licensed thermal power for Browns Ferry Nuclear Plant Units 1, 2, and 3 will require support by more complete evaluations.

B. Report on the Meeting of the Planning and Procedures Subcommittee Held January 31, 2007 (Open)

The ACRS Subcommittee on Planning and Procedures held a meeting on January 31, 2007, in Room T-2B3, Two White Flint North Building, Rockville, Maryland. The purpose of the meeting was to discuss matters related to the conduct of ACRS business. The meeting was convened at 10:00 am and adjourned at 11:45 am. A portion of this meeting was closed to discuss organizational and personnel matters.

<u>ATTENDEES</u>
W. Shack
J. Sieber
M. Bonaca

<u>ACRS STAFF</u>
F. Gillespie
S. Duraiswamy
H. Nourbakhsh
R. Caruso
J. Flack
E. Thornsbury
M. Junge
D. Fischer
J. Gallo
T. Santos
M. Afshar-Tous
G. Hammer
Z. Abdullahi

1) <u>Review of the Member Assignments and Priorities for ACRS Reports and Letters for the February ACRS meeting</u>

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

2) <u>Anticipated Workload for ACRS Members</u>

The anticipated workload for ACRS members through March 2007 is attached. The objectives are to:

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

During this session, the Subcommittee also discussed and developed recommendations on items requiring Committee action.

RECOMMENDATION

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

3) Assignments and Due Dates to Respond to the Issues Raised by the Commission in the November 8, 2006 Staff Requirements Memorandum

In the November 8, 2006 Staff Requirements Memorandum (SRM) resulting from the ACRS meeting with the NRC Commissioners on October 20, 2006, the Commission requested ACRS to perform the following tasks. Assignments (as agreed to by the Committee at the December 2006 ACRS meeting) and due dates for completing these tasks are provided below:

- As licensing under Part 52 continues, the Committee should advise the Commission on effectiveness and efficiency of staff's implementation of lessons learned in areas it has reviewed, for example, the development of guidance documents for early site permits. [Powers/Fischer] **Due Date: 11/30/07**

- The Committee should provide its views to the Commission on staff's efforts related to digital instrumentation and controls. The Committee should consider potential means for providing reasonable backup, if appropriate. [Sieber/Junge] **Due Date: 5/31/07**

- The ACRS should provide its views to the Commission with respect to staff's work on technology neutral licensing framework with a focus on ensuring the value of such an approach versus the development of a licensing framework for specific designs, such as a high temperature gas cooled reactor or a liquid metal cooled reactor. [Kress/Fischer] **Due Date: 5/31/07**

- The ACRS should provide the Commission with its recommendations and basis for areas in which NRC should perform additional long term research. [Powers/Nourbakhsh] **Due Date: 3/15/08**

- The Committee should work with the staff and external stakeholders to evaluate the different Human Reliability models in an effort to propose either a single model for the agency to use or guidance on which model(s) should be used in specific circumstances. [Apostolakis/Thornsbury] **Due Date: 6/29/07**

4) Impact of Continuing Resolution on FY2007 ACRS/ACNW Activities

The Agency is currently operating under a Continuing Resolution (CR) which is expected to continue at least through February 15, 2007. If the budget is not appropriated by that time, the CR will continue. If the CR remains in effect through FY2007 all NRC Offices have been asked to identify cost-saving measures such as the temporary cancellation of non-essential domestic and foreign travels, and cancellation of external training not part of a formal qualification program. The ACRS/ACNW Office has provided the following cost-saving measures to the Chief Financial Officer:

- Cancellation of foreign travels.
- Cancellation of domestic travels related to non-Committee meetings.
- Cancellation of the LINK contract.
- Cancellation of external training programs for the staff.
- Cancellation of the visit to San Onofre and meeting with the Regional Administrator scheduled for June 2007.

Subcommittee and full Committee meetings will continue to be funded. However, efforts should be made to hold back-to-back Subcommittee meetings to ensure efficient use of the travel budget. When the budget is approved, all the restrictions mentioned above will be eliminated, as appropriate.

5) Interview of Candidates for ACRS Membership

The ACRS Member Candidate Screening Panel has identified four candidates with expertise in the area of digital I&C and another four candidates with expertise in plant operations. Three candidates with digital I&C experience and one candidate with experience in plant operations were interviewed by the Panel and the members during the February ACRS meeting. The other four candidates will be scheduled for interview during the March meeting.

6) Assessment of the Quality of the Selected NRC Research Projects

During its December 2006 meeting, the Committee selected the following two projects for quality assessment in FY2007:

- Associated Circuit Fire Testing (CAROLFIRE) - [Banerjee (Chair), Corradini, Sieber]

- Fatigue Crack Flaw Tolerance in Nuclear Plant Piping [Shack (Chair), Armijo, Abdel-Khalik]

The Committee requested Dr. Apostolakis and Mr. Maynard to decide whether quality assessment should be performed on the following two projects:

- Development of PRA Quality Standard and Incorporation into Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of PRA Results for Risk-Informed Activities" (Apostolakis)

- Technical Review of Online Monitoring Techniques for Performance Assessment (Maynard)

Dr. Apostolakis recommends that the Committee not assess the quality of the research project on "Development of PRA Quality Standard and Incorporation into Regulatory Guide 1.200." Mr. Maynard should provide his views during the February 2007 meeting.

7) Regulatory Information Conference

The U.S. NRC's 19^{th} Annual Regulatory Information Conference is scheduled to be held March 13-15, 2007, at the Marriott Bethesda North Hotel and Conference Center in Rockville, Maryland. A preliminary program for this Conference is attached. Drs. Shack, Apostolakis, and Kress have been invited to serve on the Panels on Acceptance Criteria – 10 CFR 50.46, PRA Models, Methods, and Tools, and on Safety Margins, respectively. Support will be provided to other members who are interested in attending this conference.

8) Reappointment of Mr. Sieber

The Commission has reappointed Mr. Sieber for a third term which will expire on July 10, 2011.

C. Future Meeting Agenda

Appendix IV summarizes the proposed items endorsed by the Committee for the **540th** ACRS Meeting, **March 8 - 10, 2007.**

The **539th** ACRS meeting was adjourned at **1:00 PM, February 3, 2007.**