COMMISSION MEETING SLIDES/EXHIBITS

MEETING WITH ACRS

FRIDAY, APRIL 11, 2003

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

April 3, 2003

MEMORANDUM TO: Annette L. Vietti-Cook
Secretary of the Commission

FROM: John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards

SUBJECT: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS MEETING WITH THE U. S. NUCLEAR REGULATORY COMMISSION, APRIL 11, 2003 -- SCHEDULE AND BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners between 9:00 and 11:00 a.m. on Friday, April 11, 2003, to discuss the items listed below. Background materials related to these items are enclosed.

	ESTIMATED TIME
INTRODUCTION - NRC Chairman, Nils J. Diaz	5 minutes
ACRS PRESENTATIONS	
1. Overview - Mario V. Bonaca, ACRS Chairman • 500th Meeting Celebration • Quadripartite Meeting • License Renewal Activities • Core Power Uprates • Future Committee Activities • Sunset Activities	10 minutes
2. Advanced Reactor Designs - Thomas S. Kress • Early Site Permit Process • Options for Resolving Policy Issues • AP1000 Review Activities	15 minutes
3. Pressurized Thermal Shock (PTS) Reevaluation Project - William J. Shack	5 minutes
4. ACRS 2003 Report on the NRC Safety Research Program - F. Peter Ford	20 minutes
CLOSING REMARKS - NRC Chairman, Nils J. Diaz	5 minutes

*NOTE: Estimated times are for presentation only and do not include the time set aside for Commission questions and answers.

ACRS MEETING WITH THE U.S. NUCLEAR REGULATORY COMMISSION

April 11, 2003

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Overview

M. V. Bonaca
ACRS Chairman

Overview

- 500th Meeting Celebration
- Quadripartite Meeting
- License Renewal Activities
- Core Power Uprates
- Future ACRS Activities
- Sunset Activities

Quadripartite Meeting

Participants:

Germany, France, Japan, and U.S.

Observers:

Sweden and Switzerland

Topics:

- Safety Culture
- Probabilistic Safety Assessments
- Thermal-Hydraulic (T/H) Codes
- Stress Corrosion Cracking

ACNW Members participated in the discussion of waste management issues

License Renewal

- Reviewed three applications since July 2002
- Plan to review five applications in 2003
- Improvements to generic license renewal guidance - July 2003
- Future inspection of commitments
- Streamlined review of license renewal applications – from 2 subcommittee and 2 full committee meetings to 1 subcommittee and 1 full committee meetings

Core Power Uprates

- **Extended Power Uprate Review Standard**
  - ***Plan to review the draft final Standard after reconciliation of public comments***
- **Expect to review seven extended power uprate applications in 2004**
- **Plan to revisit the need for ACRS to review all power uprate applications after review criteria are established by the staff and the process is stabilized**

Future ACRS Activities

- Advanced Reactor Reviews
  - Early site permit process/ applications
  - Pre-application documents
- Thermal-Hydraulic Codes
- Risk-informed Regulation
- Reactor Oversight Process
- PRA quality

Future Activities (Cont'd)

- Vessel head penetration cracking and degradation
- Mixed oxide fuel fabrication facility
- Safeguards and Security matters
- American Nuclear Society Standard on low-power and shutdown risk

Sunset Activities

- Process in place to ensure that the Commission and EDO priorities are adequately considered in prioritizing the ACRS work.

Sunset Activities (Cont'd)

- **ACRS Planning and Procedures Subcommittee Reviews NRC Staff Requests and Assesses:**
  - –Value-Added from ACRS Review
  - –Previous ACRS Related Reviews
  - –Significance to NRC's Regulatory Process
  - –Timing of Committee's Review
  - –Committee's Current Workload

ACRS LETTERS

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

March 14, 2003

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

Dear Chairman Meserve:

During the 500th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 2003, we completed our review of the license renewal application for the Peach Bottom Atomic Power Station Units 2 and 3 and the final Safety Evaluation Report (SER) prepared by the NRC staff. Our Plant License Renewal Subcommittee also reviewed this matter during a meeting on October 30, 2002. During our review, we had the benefit of discussions with representatives of the NRC staff and Exelon Generation Company, LLC (Exelon). We also had the benefit of the documents referenced.

RECOMMENDATIONS AND CONCLUSIONS

1. The Exelon application for renewal of the operating licenses for Peach Bottom Atomic Power Station Units 2 and 3 should be approved.

2. The programs instituted by the applicant to manage age-related degradation are appropriate and provide reasonable assurance that Peach Bottom Atomic Power Station Units 2 and 3 can be operated in accordance with their current licensing bases for the period of extended life without undue risk to the health and safety of the public.

3. The scram at Peach Bottom Unit 2 that occurred on December 21, 2002, highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. We expect that ongoing corrective actions committed by the licensee will resolve these weaknesses.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25 which states that the ACRS review and report on license renewal applications. Peach Bottom Units 2 and 3 are General Electric boiling water reactors (BWRs) Type 4, with Mark I containments. Exelon requested renewal of their operating licenses for 20 years beyond the current license terms, which expire on August 8, 2013 for Unit 2 and July 2, 2014 for Unit 3. Peach Bottom Unit 1 is on the same site as Units 2 and 3. It is permanently shutdown and in SAFSTOR condition. There are no systems shared between Unit 1 and Units 2 and 3.

The final SER documents the staff's review of the information submitted by Exelon, including commitments that were necessary to resolve open items identified by the staff in the initial SER. Peach Bottom is the second BWR plant to seek license renewal and the first to use a system-based approach to identify structures, systems, and components (SSCs) that should be included in the scope of license renewal. The staff reviewed the completeness of the applicant's identification of SSCs that are subject to aging management; the integrated plant assessment process; the identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the aging management programs. The staff also conducted several inspections at Exelon's engineering offices and the Peach Bottom site to verify the adequacy of the methodology described in the application and its implementation.

During our Plant License Renewal Subcommittee meeting on October 30, 2002, the staff presented a well-structured and effective overview of its inspections. As in other applications, the review of the Peach Bottom license renewal application required a substantial number of requests for additional information (RAIs) and depended heavily on review of plant drawings at the site.

On the basis of our review of the final SER, we agree with the staff's conclusion that all open items and confirmatory items have been appropriately closed, and there are no issues that would preclude renewal of the operating licenses for Peach Bottom Units 2 and 3. We also concur with all four license conditions requiring the applicant to take certain actions before beginning the period of extended operation.

The process implemented by the applicant to identify SCCs that are within the scope of license renewal has been effective. The applicant included portions of nonsafety-related systems in the scope of license renewal if their failure could impact in-scope safety-related systems. When a system met this criterion, the entire system, passing through seismic Class I structures, was considered in scope. Portions of these systems that run through non-seismic structures were evaluated by walkdowns and were added to the scope as appropriate. An example of such a system is the service water system that could spray liquid on the safety systems.

Certain nonsafety systems have portions that perform a safety function, and the applicant realigned these portions to be included as part of the in-scope safety system. For example, a nonsafety-related system such as chilled water or instrument air that penetrates the containment has been realigned to be considered in scope as a part of the containment pressure retaining function. The in-scope portions of the realigned system typically include the first valve outside and inside containment and all of the piping in between.

Peach Bottom is located on the Susquehanna River on a large pond created by the Conowingo Dam (also owned by Exelon). Peach Bottom relies on the pond for operation of the units, but does not depend on the pond for emergency service water. It does depend, however, on power from Conowingo for station blackout (SBO) via a submerged electrical cable. Consequently, Conowingo is in scope for SBO considerations. The license for the Conowingo Dam will expire before the extended license period for the Peach Bottom Plant and is expected to be renewed. Should this not occur, other provisions for SBO will be required.

Open items have been closed by bringing all identified SSCs into scope. During our review, we questioned why certain other SSCs were not included in scope and, in all cases, the applicant provided appropriate justification for their exclusion. We conclude that the applicant and the staff have appropriately identified all SSCs that are within the scope of license renewal.

The applicant also performed a comprehensive aging management review of all SSCs that are within the scope of license renewal. The application describes 34 aging management programs for license renewal, which include existing, augmented, and new programs.

The applicant has proposed to inspect only the refueling water storage tank and infer from that inspection the condition of the condensate storage tank. Since these storage tanks are similar in construction, are exposed to similar water chemistry, and are located in similar environments, we agree with the staff that this is an acceptable approach.

Peach Bottom Units 2 and 3 have toroidal suppression pools and there was discussion regarding the material condition of the coating and steel. The applicant satisfactorily described inspections conducted to date to ensure the quality of material condition of the coating and steel and also described plans for future inspections.

There was a concern that the applicant did not appear to have an aging management program for the buried portions of the standby gas treatment system (SGTS) ductwork. The applicant stated that the ductwork was either hot and/or insulated and no aging management program was required. During the third license renewal inspection at Peach Bottom, the inspectors visually examined accessible exterior and interior surfaces of the SGTS and found no age-related degradation. Based on the results of this inspection, the staff agreed with the applicant.

Peach Bottom has had a history of cable failure due to moisture intrusion in 4Kv and 13Kv service. Many cables have been replaced with moisture-resistant cables. In recent NRC inspections, water intrusion was evident in certain manholes and seems to be an ongoing problem. Consequently, the applicant committed to a program to manage the aging of inaccessible medium-voltage cables. This aging management program provides reasonable assurance that the intended functions of the systems and components will be maintained consistent with the current licensing basis during the period of extended operation.

With regard to the inspection of reactor vessel internals, the applicant has committed to the programs prescribed in 15 BWR Vessel and Internals Project (BWRVIP) reports. These programs have all been approved by the NRC staff for 60 year plant life except those described in BWRVIP-78, BWR Integrated Surveillance Program, and BWRVIP-86, BWR Integrated Surveillance Program Implementation Plan, which are approved only for 40 year plant life. The staff is currently reviewing these BWRVIP reports for 60 years. The applicant has agreed to a license condition to notify the NRC, before entering the period of extended operation, of its decision to implement either the staff-approved integrated surveillance program (ISP) or a staff-approved plant-specific ISP. Also, the staff has not yet approved BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines." Because the staff's review is not complete, the applicant has agreed to another license condition to notify the NRC of its decision to implement either the staff-approved core shroud inspection and evaluation guidelines program, or a staff-approved plant-specific program.

Exelon has also identified those components at Peach Bottom that are supported by time-limited aging analyses (TLAAs). These TLAAs show that the components analyzed have sufficient margin to operate for the period of extended life.

Peach Bottom Unit 2 experienced a scram on December 21, 2002. This event highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. We expect that ongoing corrective actions committed by the licensee will resolve these weaknesses. During inspections, the staff should assess the effectiveness as well as the adequacy of implementation of these programs.

The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. Adequate programs have been established to manage the effects of aging so that Peach Bottom Units 2 and 3 can be operated in accordance with their current licensing bases for the period of extended life without undue risk to the health and safety of the public.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

<u>References</u>:

1. Letter dated July 2, 2001, from J. A. Benjamin, Exelon Generation Company, LLC, to U. S. Nuclear Regulatory Commission, transmitting Application to Renew the Operating Licenses of Peach Bottom Atomic Power Station Units 2 and 3
2. U.S. Nuclear Regulatory Commission, NUREG-XXX, "Safety Evaluation Report Related to the License Renewal of Peach Bottom Atomic Power Station, Units 2 and 3" February, 2003.

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

February 14, 2003

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE MCGUIRE NUCLEAR STATION UNITS 1 AND 2 AND THE CATAWBA NUCLEAR STATION UNITS 1 AND 2

Dear Chairman Meserve:

During the 499th meeting of the Advisory Committee on Reactor Safeguards on February 6–8, 2003, we completed our review of the License Renewal Application (LRA) for the McGuire Nuclear Station Units 1 and 2 (McGuire) and the Catawba Nuclear Station Units 1 and 2 (Catawba), and the related final safety evaluation report (SER) prepared by the NRC staff. Our review included a meeting of our Plant License Renewal Subcommittee on October 8, 2002. During our review, we had the benefit of discussions with representatives of the NRC staff and Duke Energy Corporation (Duke). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Duke application for renewal of the operating licenses for McGuire Units 1 and 2 and Catawba Units 1 and 2 should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that McGuire Units 1 and 2 and Catawba Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25, which states that the ACRS should review and report on all license renewal applications. McGuire Units 1 and 2 and Catawba Units 1 and 2 are 3,411- MWt, four-loop Westinghouse pressurized-water reactors (PWRs) in ice condenser containments. In its application, Duke requested that the NRC renew the operating licenses for all four units beyond their current license terms, which expire on June 12, 2021 (McGuire Unit 1); March 3, 2023 (McGuire Unit 2); December 6, 2024 (Catawba Unit 1); and February 24, 2026 (Catawba Unit 2). At the time of the application, only McGuire Unit 1 met the requirements of 10 CFR 54.17(c), which prohibits an applicant from submitting an application for license renewal earlier than 20 years before the expiration of its current operating license. Duke requested an exemption from this requirement, which the NRC staff granted based on the similarities of the four units and the efficiency of a single application.

The final SER documents the results of the staff's review of information submitted by Duke, including commitments that were necessary to resolve open items identified by the staff in the initial SER. In particular, the staff reviewed the completeness of the applicant's identification of structures, systems, and components (SSCs) that are subject to aging management; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the applicant's aging management programs. The staff also conducted several inspections at Duke's engineering offices and at the McGuire and Catawba sites to verify the adequacy of the methodology described in the application and its implementation.

During our Plant License Renewal Subcommittee meeting on October 8, 2002, the lead NRC license renewal inspector for Region II provided an overview of the NRC's inspection process. This process, which is well-structured and effective, is becoming increasingly important as license renewal applications become less detailed. As a result, as in other recent applications, the review of the McGuire and Catawba LRA required a substantial number of requests for additional information and depended heavily on review of plant drawings at the sites.

On the basis of our review of the final SER, we agree with the staff's conclusion that all open and confirmatory items have been closed appropriately, and there are no issues that preclude renewal of the operating licenses for McGuire Units 1 and 2 and Catawba Units 1 and 2.

The process implemented by the applicant to identify SSCs that are within the scope of license renewal was effective. However, in the initial SER the staff identified a number of SSCs that should have been in the scope of license renewal but were excluded by Duke's interpretation of license renewal requirements. Among those SSCs were fan and damper housings, building sealants, electrical equipment connecting the units to the offsite power source for recovery from station blackout (SBO), and jockey pumps and manual fire suppression equipment in potential fire exposure areas. The inclusion of fan and damper housings, building sealants, and SBO equipment has been disputed in previous license renewal applications.

For fan and damper housings, Duke initially took the position that loss of pressure retention or structural integrity function would be evidenced by functional failure, as is a failure of the active components of dampers and fans. By contrast, the staff views the passive components of these assemblies as being within the scope of license renewal, just like pump casings, which are explicitly called for in 10 CFR 54.21. We agree that the explicit example provided in the rule supports the staff's interpretation. With regard to jockey pumps, the staff determined that these components are relied upon to meet the requirements of 10 CFR 50.48, "Fire Protection." We concur with the staff's determination. Duke agreed to close these open items by bringing all of the identified SSCs into the scope of license renewal.

During our review, we questioned why certain other SSCs were not included within the scope and, in all cases, the applicant provided appropriate justification for exclusion. We conclude that the applicant and the staff have appropriately identified all SSCs that are within the scope of license renewal.

The applicant performed a comprehensive aging management review of SSCs that are within the scope of license renewal. Appendix B to the LRA describes 51 aging management programs for license renewal, which include existing, enhanced, and new programs. In addition, the resolution of staff questions and SER open items has resulted in further commitments, including the implementation of a one-time inspection of the condenser circulating water system expansion joints at Catawba to characterize potential degradation, one-time VT-1 inspection of the pressurizer spray head, one-time inspection of the internal surfaces of the auxiliary feedwater system carbon steel piping components, and an inspection program for non-environmentally qualified neutron flux instrumentation circuits. The SER lists 21 such committed actions to be implemented by the applicant.

The McGuire and Catawba LRA includes a new aging management program, the Alloy 600 Aging Management Review. This program is intended to identify Alloy 600/690, 82/182, and 52/152 locations; to rank susceptibility to primary water stress corrosion cracking (PWSCC); and to verify that nickel-based alloy locations are adequately inspected by the Inservice Inspection Program, the Control Rod Drive Mechanism and other Vessel Head Penetration (VHP) programs, the Reactor Vessel Internals Program, and the Steam Generator Integrity Program. This review will provide general oversight and management of cracking due to PWSCC. We applaud this initiative to provide comprehensive oversight of activities to manage PWSCC. Given the current challenge created by PWSCC, we encourage Duke to implement this program soon, in the current license term, rather than waiting for the end of the initial license terms of the four units.

With regard to reactor vessel penetration nozzle cracking and head wastage issues, Duke has committed to incorporate the future industry resolution of these issues into the VHP Nozzle Program and the Alloy 600 Management Review Program. This provides reasonable assurance that the effects of aging associated with the VHP Nozzle Program and the Alloy 600 Review Program will be adequately managed so that the intended function(s) will be maintained in a manner that is consistent with the current licensing basis throughout the period of extended operation.

Duke is the first utility to seek license renewal for plants that use ice condensers in the containment to absorb thermal energy in the event of a loss-of-coolant-accident or a steamline break. Duke has developed a new program to manage aging degradation of ice baskets and ice condenser components at McGuire and Catawba. We agree with the staff's conclusion that the proposed program is adequate to identify and manage aging effects during the period of extended operation.

Duke identified those components of the McGuire and Catawba plants that are supported by time-limited aging analyses and provided sufficient data to demonstrate that the components have sufficient margin to operate properly for the period of extended operation.

As noted in previous applications, LRAs include a substantial number of activities and commitments that will not be accomplished until near the end of the current license period. Consequently, the NRC staff will need to conduct a substantial amount of inspection activity just before the plants enter the extended period of operation. The staff is aware of this future workload and has issued Inspection Procedure 71003, "Post-Approval Site Inspection for License Renewal," to manage this significant effort. Given the large number of power plants that will approach the license renewal term at approximately the same time, this nationwide inspection effort is likely to impose a major demand for staff resources.

The staff has performed an outstanding review of the Duke application. The applicant and the staff have identified plausible aging effects associated with passive, long-lived components. The applicant has also established adequate programs to manage the effects of aging so that McGuire Units 1 and 2 and Catawba Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:
1. Letter dated June 13, 2001, from M. S. Tuckman, Duke Energy Corporation, to U. S. Nuclear Regulatory Commission, transmitting Application to Renew the Operating Licenses of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2.
2. U.S. Nuclear Regulatory Commission, NUREG-XXX, "Safety Evaluation Report Related to the License Renewal of McGuire Nuclear Station, Units 1 and 2 and Catawba Nuclear Station, Units 1 and 2," January 2003.
3. U.S. Nuclear Regulatory Commission, NRC Inspection Procedure 71003, "Post-Approval Site Inspection for License Renewal," December 9, 2002.

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

December 18, 2002

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATIONS FOR THE NORTH ANNA POWER STATION UNITS 1 AND 2 AND THE SURRY POWER STATION UNITS 1 AND 2

Dear Chairman Meserve:

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we completed our review of the License Renewal Application for North Anna Power Station (NAS) Units 1 and 2, the Surry Power Station (SPS) Units 1 and 2, and the final Safety Evaluation Report (SER) prepared by the staff of the U. S. Nuclear Regulatory Commission (NRC). Our review included a meeting of our Plant License Renewal Subcommittee on July 9, 2002. During our review, we had the benefit of discussions with representatives of the NRC staff and Virginia Electric and Power Company (Dominion). We also had the benefit of the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The Dominion application for renewal of the operating licenses for NAS Units 1 and 2 and SPS Units 1 and 2 should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that NAS Units 1 and 2 and SPS Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

BACKGROUND AND DISCUSSION

This report fulfills the requirement of 10 CFR 54.25 which states that the ACRS should review and report on license renewal applications. Dominion requested renewal of the operating licenses for NAS Units 1 and 2 and SPS Units 1 and 2 for a period of 20 years beyond the current license terms, which expire on April 1, 2018 (NAS Unit 1); August 21, 2020 (NAS Unit 2); May 25, 2012 (SPS Unit 1); and January 29, 2013 (SPS Unit 2). The final SER, issued on November 5, 2002, documents the results of the staff's review of information submitted by Dominion, including commitments that were necessary to resolve the open items identified by the staff in the initial SER. This review of the application was conducted concurrently for two stations with a total of four units. Given the similarity of the units and the formatting of the application, which clearly highlighted the few differences, the concurrent review did not present any unusual difficulties.

The staff reviewed the completeness of the identification of structures, systems, and components (SSCs) subject to aging management; the integrated plant assessment process; the applicant's identification of the possible aging mechanisms associated with passive, long-lived components; and the adequacy of the aging management programs. The staff also conducted three inspections. First, a 1-week inspection was performed to assess the applicant's scoping and screening methodology. Next a 1-week inspection was conducted at each facility to assess plant material condition and aging management programs. Lastly, an inspection was performed to close open items resulting from the earlier inspections.

The staff provided the Committee with details of the scope and results of its inspections of material condition at both plants. We agree with the staff's assessment that there are no issues that would preclude renewal of the operating license for NAS Units 1 and 2 and SPS Units 1 and 2.

On the basis of our review of the final SER, we agree that all open items and confirmatory items have been appropriately closed. We also discussed several items that were raised at the Subcommittee meeting on July 9, 2002, and found that the staff and the applicant have satisfactorily addressed each item.

The processes implemented by the applicant to identify SSCs that are within the scope of license renewal were effective. As with several previous applicants, the staff engaged in considerable discussion with the applicant regarding the portion of the offsite power system to be included within the scope of license renewal. After reviewing the information provided by the applicant, we agree that appropriate portions of the offsite power system are included in scope. During our review, we questioned why certain other SSCs were not included within the scope and, in all cases, the applicant provided appropriate justification for exclusion.

The applicant has performed a comprehensive aging management review of SSCs that are within the scope of license renewal. There are 19 existing aging management programs and four new programs.

The applicant has satisfactorily responded to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," dated March 18, 2002. Further, the applicant has committed to replace all four reactor vessel heads. The replacement of the NAS Unit 2 head is currently in progress.

The applicant used the guidance specified in Westinghouse Owners Group reports for reactor coolant system piping, pressurizer, and reactor internals. The staff reviewed and approved the use of these reports with certain stipulations. Each stipulation was sufficiently addressed in the staff's review.

We questioned the method by which reactor coolant piping is to be inspected in light of the failure of the initial volumetric inservice inspection to detect vessel nozzle cracking at V.C. Summer. Although continued improvement in the inspection methodology is warranted, the staff considers current methods adequate to detect primary water stress corrosion cracking. This is a generic issue and we remain concerned with the effectiveness of inspection techniques. Dominion has committed to employ best industry practices as they are developed.

Dominion has also committed to conduct a one-time inspection of a representative sample of buried piping. Opportunistic inspections of in-scope buried piping will be performed when the piping is uncovered during other maintenance activities. If significant degradation is identified, the results will be entered into the licensee's corrective action program and the inspection will be expanded. If no opportunity presents itself by the end of the current license period, excavations will be made to inspect the piping.

The applicant's erosion/corrosion program is of particular interest in light of the previous carbon steel piping failures at SPS. Dominion uses the CHECWORKS program to identify locations to be monitored and trend erosion/corrosion rates. The program appears to be effective in managing erosion/corrosion.

Certain medium-voltage cables exposed to moisture for long periods of time fail due to a phenomenon called "water treeing." To preclude this failure, the applicant has committed to a program that will control water in manholes and underground ducts associated with energized power cables. The Cable Monitoring Activities Program for non-environmentally qualified cable has been enhanced to ensure that if degraded cable is identified, the cable environment, including the potential for moisture shall be evaluated and appropriate corrective actions initiated through the corrective action program.

During the discussion of time-limited aging analyses, we expressed a concern that the applicant had not submitted its evaluations of the reactor vessel margins for pressurized thermal shock and upper shelf energy. The staff had accepted the applicant's position that these values were acceptable without performing an independent evaluation. Subsequently, the staff obtained this information from the applicant and the staff performed an independent evaluation. Although in some cases the margins are small, we agree with the staff's position that margin does exist. We believe that in the future such critical parameters should be reviewed by the staff. The staff agreed to require that these data be provided with future license renewal applications.

In several situations, Dominion and other applicants have committed to actions based on future technology development. In Dominion's case, two examples are (1) the method for inspecting reactor coolant piping, and (2) the method for testing of medium-voltage cables exposed to moisture. The NRC staff needs to continue to keep abreast of these developing technologies and review and approve methodologies at the appropriate time.

License renewal applications include a number of activities and commitments, for example one-time inspections, that will not be accomplished until near the end of the current license period. There is a large amount of inspection activity that needs to be conducted at that time period. The staff is aware of this future work load and is working on a plan to properly manage this significant effort.

The applicant and the staff have identified plausible aging effects associated with passive, long lived components. Adequate programs have been established to manage the effects of aging so that NAS Units 1 and 2 and SPS Units 1 and 2 can be operated in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public.

Sincerely,

George E. Apostolakis
Chairman

References:
1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the North Anna Power Station, Units 1 and 2, and the Surry Nuclear Station, Units 1 and 2," issued November 2002.
2. Dominion Application for Renewed Operating License for North Anna Power Station, Units, 1 and 2, and Surry Power Station, Units 1 and 2, submitted May 29, 2001.

EDO RESPONSE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 29, 2003

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE NORTH ANNA POWER STATION (NAPS), UNITS 1 AND 2, AND THE SURRY POWER STATION (SPS), UNITS 1 AND 2

Dear Dr. Bonaca:

In your letter to Chairman Meserve dated December 18, 2002, you summarized the results of the review by the Advisory Committee on Reactor Safeguards (ACRS) of Virginia Electric and Power Company's (Dominion's) license renewal application (LRA) for the NAPS, Units 1 and 2, the SPS, Units 1 and 2, and the NRC staff's final safety evaluation report (SER) on the application. On the basis of its review, the ACRS concluded that all open and confirmatory items had been resolved and that there was reasonable assurance that NAPS, Units 1 and 2, and SPS, Units 1 and 2, could be operated safely in accordance with their current licensing bases for the period of extended operation without undue risk to the health and safety of the public. The ACRS' timely review helped the staff maintain the review schedule. The staff has prepared a recommendation to the Director of the Office of Nuclear Reactor Regulation for the issuance of the NAPS, Units 1 and 2, and SPS, Units 1 and 2, renewed licenses. We appreciate the ACRS' effort in supporting the license renewal activities.

Your December 18, 2002, letter also contains a background discussion section. In the following we address certain comments from this section.

Your Comment:

"We questioned the method by which reactor coolant piping is to be inspected in light of the failure of the initial volumetric inservice inspection to detect vessel nozzle cracking at V.C. Summer. Although continued improvement in the inspection methodology is warranted, the staff considers current methods adequate to detect primary water stress corrosion cracking. This is a generic issue and we remain concerned with the effectiveness of inspection techniques. Dominion has committed to employ best industry practices as they are developed."

Response:

In investigating the V.C. Summer incident, the NRC formed a special inspection team to review activities associated with the weld leakage and cracking and determine if any potential generic issues contributed to the cracking and the failure of the required inservice inspection (ISI) program to detect the cracking. The primary issue identified by the team with potential generic implications was the inability to reliably detect cracks of all sizes using the Code-required ultrasonic (UT) inspection method. Although the crack that resulted in the leak of boric acid at V.C. Summer was identified by current technology UT inspection, a number of smaller cracks identified by subsequent eddy current (ET) inspections were not previously identified by the code-required UT inspection method.

In an industry initiative under Project No. 689, the Materials Reliability Project (MRP) of the Electric Power Research Institute (EPRI), is currently pursuing a study of any generic implications of primary water stress-corrosion cracking (PWSCC) (NEI letter dated December 14, 2000, ADAMS ML003779569). The MRP is a utility-directed oversight organization of the Pressurized Water Reactor (PWR) Owners Group. The purpose of the MRP is to address and resolve, on a consistent industry-wide basis, PWR material-related issues. This effort is ongoing and the staff is continuing its interactions with the MRP. In addition, when licensees perform ultrasonic examinations of dissimilar metal welds, they are now required to use the more reliable performance demonstration methods stipulated in the ASME code. Dominion has committed to employ the best industry practices as they become available as a result of this study.

Your Comment:

"During the discussion of time-limited aging analyses, we expressed a concern that the applicant had not submitted its evaluations of the reactor vessel margins for pressurized thermal shock and upper shelf energy. . . . We believe that in the future such critical parameters should be reviewed by the staff. The staff agreed to require that these data be provided with future license renewal applications."

Response:

This comment relates to the level of detail for the time-limited aging analyses (TLAAs) in an LRA. The staff agrees with the ACRS' recommendation. In the near term, the staff will obtain more TLAA data from the applicants through requests for additional information (RAIs), as appropriate. For the long term, the staff is discussing with NEI a revision of NEI 95-10, the industry guidance document.

Your Comment:

"In several situations, Dominion and other applicants have committed to actions based on future technology development. In Dominion's case, two examples are (1) the method for inspecting reactor coolant piping, and (2) the method for testing of medium-voltage cables exposed to moisture. The NRC staff needs to continue to keep abreast of these developing technologies and review and approve methodologies at the appropriate time."

Response:

We addressed the inspection of PWSCC in reactor cooling piping, as mentioned above. Regarding the testing of the cables, the industry, through the Institute of Electrical and Electronic Engineers (IEEE), is developing a suitable testing method (e.g., a partial discharge test) for detecting any defects and abnormalities in medium-voltage cables exposed to moisture. The purpose of this test is to determine the adequacy of cables to perform their intended function. Also, the Office of Nuclear Regulatory Research recently completed an aging assessment of safety-related power cables (NUREG/CR-6794). The assessment identifies other condition-monitoring techniques. The NRC staff will keep abreast of these developing technologies and will review and approve methodologies, as appropriate.

Your Comment:

"License renewal applications include a number of activities and commitments, for example one-time inspections, that will not be accomplished until near the end of the current license period. There is a large amount of inspection activity that needs to be conducted at that time period. The staff is aware of this future work load and is working on a plan to properly manage this significant effort."

Response:

The staff agrees with the ACRS that verifying the implementation of renewal commitments before the end of the current license period will be a significant inspection effort. On December 9, 2002, the staff issued Inspection Procedure (IP) 71003, "Post-Approval Site Inspection for License Renewal," to specify the scope of this inspection. In accordance with the ACRS' recommendation, the staff also plans to include a list of plant-specific comments in IP 71003 to focus the activities of future inspections. Headquarters staff will be working with regional staff to plan and manage the inspection effort.

I hope this responds to the comments in your December 18, 2002 letter.

Sincerely,

William Travers

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

2

ADVANCED REACTOR DESIGNS

T. S. Kress

Recent ACRS Reviews Associated With Advanced Reactors

I. Early Site Permit process (ESP)

II. Options for resolving policy issues

III. AP1000 review activities

Early Site Permit Activities

Full Committee Meeting
November 7, 2002

- NEI's approach for ESP
- Staff's approach for a review standard
- Briefing only, no report

Early Site Permit Activities (Cont'd)

Full Committee Meeting
March 7, 2003

- Reviewed a draft of the proposed review standard
- ACRS Report March 12, 2003

ACRS March 12, 2003 Report

The Review Standard

- **Is appropriate for reviewing ESP applications**
- **Will accommodate industry's proposed use of plant parameter envelope concept**

Policy Issues

Staff identified 7 policy issues

- Expectations for enhanced safety
- Defense-in-depth
- International safety standards and requirements
- Event selection and safety classification
- Source term
- Containment vs. Confinement
- Emergency preparedness

ACRS Report
December 13, 2002

- We agreed that the Key Technical Issues (KTIs) identified by the staff needed resolution before certification reviews
- The preferred options to address the KTIs were consistent with opinions we had previously expressed

AP1000 Review Activities

- Phase 1 – Establish goals and estimate for pre-licensing review

  Completed - Letter 6/21/00
- Phase 2 – Develop positions on 4 key issues identified in Phase 1

  Completed - Report 3/14/02

AP1000 (Cont'd)

Phase 2 - Report 3/14/02

- Agreed with staff position on key issues
- Raised flag on appropriate range of PI-group values for scaling

AP1000 (Cont'd)

Phase 3 (Design Certification) - In progress

- Westinghouse/ACRS meeting 11/7/02
- ACRS PRA Subcommittee 1/23-24/03
  - Reliability of ADS-4 squib valves questioned

AP1000 (Cont'd)

T/H Subcommittee 3/19-20/03

- Entrainment of liquid at ADS-4 and top of core still an issue
- Potential for Boron precipitation
- Sump strainer design

AP1000 (Cont'd)

- **Future Plant Designs and T/H Subcommittees 7/03**

  (Containment structural design, materials, regulatory treatment of non-safety systems, shutdown maintenance, open items)
- **Full Committee Interim Report/DSER 9/03**
- **Full Committee Final Report/FSER 7/04**

ACRS LETTERS

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

March 12, 2003

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT REVIEW STANDARD, RS-002: "PROCESSING APPLICATIONS FOR EARLY SITE PERMITS"

During the 500th meeting of the Advisory Committee on Reactor Safeguards, March 6-8, 2003, we met with representatives of the NRC's Office of Nuclear Reactor Regulation (NRR) to discuss the staff's draft Review Standard for processing applications for early site permits (ESPs). We also had the benefit of the document referenced.

CONCLUSIONS

The draft ESP Review Standard is appropriate for review of early site permit applications and will accommodate the industry's proposed use of the Plant Parameter Envelope (PPE) concept.

DISCUSSION

The staff has modified the appropriate sections of the Standard Review Plan (SRP) to make use of existing guidance to the extent possible. The modifications generally consist of elimination of the contents of the SRP that are not applicable to ESP and revisions to bring the SRP up to date. In general, references to plant layouts or design details are deleted and replaced with a statement of the form: "... [specify these details for] a nuclear power plant or plants of specified type that might be constructed on the proposed site to the extent this information is available." Some review issues that require knowledge of items that are design-specific, such as source terms, will be accommodated by bounding values specified in the PPE portion of the application and confirmed at the Combined License (COL) stage. For already approved sites with existing plants, most of the review areas called for by the standard will have already been sufficiently addressed. The applicant will merely need to verify, compile, and docket these review areas.

Although the sections of the current SRP that deal with siting issues require a specific design, the proposed ESP standard recognizes that by specifying parameters such as distance to the exclusion area boundary, source term characteristics, and relative concentration (χ/Q) values in the PPE, it will be possible to demonstrate that a plant that fits within the PPE can be safely located on the site. The PPE can also accommodate the need to assess incremental environmental impact to ensure that it is acceptable. We believe that granting ESP based on the guidance of this proposed review standard will assure adequate protection and acceptable environmental impact when a plant is built on the approved site.

Sincerely,

Mario V. Bonaca
Chairman

Reference:

U.S. Nuclear Regulatory Commission Review Standard (Draft) RS-002, "Processing Applications for Early Site Permits," Draft for Interim Use and Public Comment, December 23, 2002.

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

December 13, 2002

The Honorable Richard A. Meserve
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: DRAFT COMMISSION PAPER ON POLICY ISSUES RELATED TO NON-LIGHT-WATER REACTOR DESIGNS

During the 498th meeting of the Advisory Committee on Reactor Safeguards, December 5-7, 2002, we met with representatives of the NRC's Office of Nuclear Regulatory Research (RES) to discuss the subject draft Commission paper. We also had the benefit of the referenced documents.

The RES staff has identified seven technical issues with policy implications that need resolution prior to certification reviews of advanced non-light-water reactor designs. The staff has also provided options for resolving those issues and has recommended the preferred options. We agree with the staff's recommended preferred options for each of the seven issues.

We commend the staff on its thoughtful study and look forward to further interactions on this subject.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Draft Predecisional SECY paper, undated, from William D. Travers, Executive Director for Operations, NRC, to the Commission, Subject: Policy Issues Related to Licensing Non-Light-Water Reactor Designs.
2. U. S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants."

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

March 14, 2002

The Honorable Richard A. Meserve
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: PHASE 2 PRE-APPLICATION REVIEW FOR AP1000 PASSIVE PLANT DESIGN

During the 490th meeting of the Advisory Committee on Reactor Safeguards (ACRS), March 7-9, 2002, we completed our evaluation of the Phase 2 pre-application review of the Westinghouse AP1000 passive plant design, conducted by the NRC staff. This matter was also reviewed during joint meetings of our Subcommittees on Thermal-Hydraulic Phenomena and Future Plant Designs on February 13-15, 2002, and a meeting of the Subcommittee on Thermal-Hydraulic Phenomena on March 15, 2001. During our review, we had discussions with representatives of the Westinghouse Electric Company and the NRC Staff. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The staff has made a competent and thorough review of the Phase 2 issues.

2. We agree that the proposal by Westinghouse to use Design Acceptance Criteria (DAC) for the piping design should be approved.

3. The staff's positions on the other pre-application review issues should also be approved.

4. The Office of Nuclear Regulatory Research (RES) should further investigate acceptable ranges of ratios of Pi-groups for use in scaling.

5. The ad hoc introduction of compensating processes to tune codes to the integral test data should be discouraged.

Discussion

The NRC staff and Westinghouse have agreed to a three-phased approach to the AP1000 standard plant design review. Phase 1, which was to identify the key review issues, was completed previously and resulted in the identification of four key issues:

1. Acceptability of the proposed use of DAC for particular parts of the design review.

2. Acceptability of certain exemptions that Westinghouse intends to request.

3. Applicability of the AP600 test program to the AP1000 design.

4. Applicability of the AP600 analyses codes to the AP1000 design.

The purpose of the Phase 2 review was for the staff to develop positions on these four key issues. These positions are discussed below.

Proposed Use of DAC

The Commission has determined that the level of detail in a design certification application must be sufficient to enable the Commission to judge the applicant's proposed means of ensuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design.

The staff has interpreted this policy to mean that the certification application must be complete, with two exceptions:

- items for which the technology is rapidly changing and may be significantly different at the combined operating license (COL) stage.

- items for which the level of detail cannot be provided at the time of certification review (or for which the as-procured and as-built characteristics are needed).

For these exceptions, DAC are required of the applicant. Some precedents for DAC satisfying these criteria were established with the certifications of the Advanced Boiling Water Reactor (ABWR) and System 80+ designs. For these, the staff accepted DAC for the instrumentation and control (I&C) and for the control room design, both of which were deemed to satisfy one or more of the above criteria.

In addition to these two areas for which precedents have been established, Westinghouse has proposed DAC for the AP1000 piping design.

The staff has concluded that the DAC approach should be approved for I&C and control room portions of the design based on the two criteria above and that the DAC on piping design should be approved based on the similarity of AP1000 to AP600 designs, for which the certification included sufficient piping design detail.

While we have some sympathy with this view by the staff and agree that the piping DAC should be approved, we believe the piping DAC could have been approved without invoking the similarity to the AP600 design. Our view is that, as long as sufficient detail is available to permit resolution of safety questions, the degree of detail that an applicant wishes to provide at the certification phase is a business decision. We believe the use of DAC for the piping design fits this characterization.

Exemptions

Westinghouse is requesting exemptions from the regulations in three areas:

(a.) Section 50.34 (f)(2)(iv) requires a "safety parameter display console that will display to operators a minimum set of parameters defining the safety status ... displaying a full range of important plant parameters ..., and capable of indicating when process limits are being approached or exceeded."

(b.) Section 50.62(c)(1) requires that equipment be available to ensure the automatic startup of the auxiliary feedwater system under ATWS conditions.

(c.) GDC 17 of 10CFR50 Appendix A requires two physically independent offsite power sources.

The staff agrees with the Westinghouse positions that: Item (a) will be part of the DAC for control room design; the underlying purpose of Item (b) is satisfied because AP1000 does not have (or need) an auxiliary feedwater system as the emergency core cooling system (ECCS) requirement is met by the passive residual heat removal (PRHR) system automatic initiation under ATWS; and that the underlying purpose of Item (c) is satisfied because, with the passive ECCS, AP1000 does not need offsite power to make its safety case. We also agree with these positions.

Applicability of AP600 Standard Plant Design Analysis Codes and Test Program

To address the applicability of the AP600 codes and test program, Westinghouse prepared a new AP1000 phenomena identification and ranking table (PIRT) and conducted new scaling assessments for both the codes and the tests. The AP1000 PIRT resulted in the same high- and medium-ranked phenomena as were found for the AP600, and it was noted that the AP1000 design did not entail any important new phenomena. In addition, the scaling analyses indicated that the Pi-groups identified as being important and which were to be substantially matched in the integral test program were still in the acceptable range when compared to their values for the full-scale AP1000 design. Thus, Westinghouse maintains that these results demonstrate that the AP600 test database used to validate the analysis codes is applicable to AP1000 and that the codes should be approved for use in evaluating the safety status of AP1000 design.

The staff conducted independent top-down and bottom-up scaling assessments and made audit calculations using RELAP5 for a postulated 2-inch diameter break in the cold leg and for a postulated double-ended direct vessel injection (DVI) line rupture. The staff found that, with some noted exceptions, the experimental data produced by the AP600 separate effects and integral effects test programs are appropriate for verification of the processes expected in an AP1000 plant, and the analysis codes validated for the AP600 standard plant design are applicable to the AP1000 design.

The most significant of the exceptions is that the tests are not considered sufficient to validate the entrainment model used in the NOTRUMP code for the upper plenum regions and for the hot-leg exit through the automatic depressurization system (ADS-4) depressurization valve.

Westinghouse claims that the scaling test data and analyses are sufficient to ensure that the core remains covered and that the entrainment is a self-limiting process that decreases as the core water level decreases. Westinghouse also claims that the period during which the entrainment is important in affecting the water level is so short that entrainment is not safety significant. We think such a case can be made during the certification review and, if so, additional tests would not be necessary.

Nonetheless, the staff's position has merit in that it will be necessary to better predict the entrainment behavior before judgments can be made regarding its safety significance. We believe phenomena that are ranked high or medium in importance should be properly treated in the models partly because unanticipated applications could invalidate the "non-safety-important" judgment. We remain concerned that the codes do not properly model entrainment because inapplicable maps are being used to characterize the flow regimes. The use of inapplicable maps could impact the results of the codes in unanticipated ways. Thus, we are convinced that the technical basis codes need better modeling with respect to entrainment and flow regime maps.

<u>Other Considerations</u>

In the scaling assessments, Westinghouse and the staff used the criterion that Pi-group ratios having values between 0.5 and 2.0 represent acceptable scaling. While this range is intuitively pleasing as an indication that the tests sufficiently match the phenomena in AP1000, we have not seen any technical justification for this criterion. Thus, we believe that RES should initiate a study with the objective of establishing a technically based approach for use in determining the significance of any general Pi-group. We think this would involve sensitivity analyses on the Pi-group in the non-dimensional scaling models. The sensitivity of the results to individual Pi-group ratios could guide the selection of acceptance ranges that might be different for different Pi-groups. Although we do not believe that this work is needed for AP1000 certification, this issue is likely to arise with certification of future reactor designs and such a study could tie down this loose end of the code, scaling, applicability, and uncertainty (CSAU) process.

There are two instances in which Westinghouse proposes to adjust its models to provide a better fit to integral data by introducing compensating processes. In one instance, the NOTRUMP code does not model the momentum flux terms in the conservation of momentum equations dealing with effects of area and density changes. This deficiency in the code impacts its ability to calculate pressurizer drainage and reactor vessel downcomer level. To compensate for this code deficiency in the AP600 certification, Westinghouse imposed a reduction in the in-containment refueling water storage tank (IRWST) level – thus reducing the driving force which would conservatively compensate for the effects that would have resulted from having the correct momentum equations. For the AP1000, instead of this same "fix," Westinghouse proposes to use an increased flow resistance penalty that would make the code calculations fit the APEX facility data for a 2-inch small-break loss-of-coolant accident (SBLOCA).

In another instance, Westinghouse concluded that the NOTRUMP PRHR model does not model the thermal plume in the IRWST. The model will over predict the outside surface heat transfer rate for the heat exchanger when the tube flow velocity exceeds 1.5 ft/sec for any significant period of time. If this situation arises in the analyses, Westinghouse proposes to account for the non-conservative calculation by an ad hoc reduction of the predicted heat exchanger performance.

These temporary fixes should provide conservative results to support the certification of AP1000 design. Nevertheless, we view both of these as instances of purposeful introduction of compensating errors in the codes rather than improving the models. We consider it bad practice to allow these errors to persist in the codes and believe that the actual physics should be properly represented in the long term.

Sincerely,

George E. Apostolakis
Chairman

References:

1. Memorandum dated February 4, 2002, transmitting draft SECY Paper, undated, Subject: Use of Design Acceptance Criteria and Exemptions for the AP1000 Standard Plant Design (Predecisional), and draft SECY Paper, undated, Subject: Applicability of AP600 Standard Plant Design Analysis Codes and Test Program to the AP1000 Standard Plant Design (Predecisional).
2. Memorandum dated June 21, 2000, from John T. Larkins, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: AP1000 Pre-Application Review.

EDO RESPONSE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 19, 2002

Dr. George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

SUBJECT: PHASE 2 PRE-APPLICATION REVIEW FOR AP1000 PASSIVE PLANT DESIGN

Dear Dr. Apostolakis:

Thank you for your letter of March 14, 2002, regarding the presentation to the Advisory Committee on Reactor Safeguards (ACRS) on March 8, 2002, of the staff's assessment of the AP1000 standard plant design pre-application review. The staff also had the opportunity to present a more detailed assessment to the ACRS subcommittees on Thermal-Hydraulic Phenomena and Future Plant Designs on February 14 and 15, 2002.

As presented on March 8, 2002, the scope of the staff's pre-application review was limited to assessing: (1) the applicability of the AP600 test program and analysis codes to the AP1000; (2) the applicability to AP1000 of three exemptions that were granted for the AP600; and (3) the acceptability of using design acceptance criteria (DAC) in lieu of detailed design information in the instrumentation and controls (I&C), control room (human factors engineering), and piping design areas.

The staff will document its assessment of the DAC issue in a Commission paper that should be made publicly available in April 2002. The staff documented its assessment of the other issues in a letter to Westinghouse Electric Company, dated March 25, 2002.

Your letter of March 14, 2002, conveyed ACRS conclusions and recommendations regarding the staff's pre-application review (Phase 2) and the planned design certification review (Phase 3) of the AP1000. In general, you indicated that ACRS agrees with the staff's conclusions regarding the pre-application review.

You also offered the recommendation that the Office of Nuclear Regulatory Research (RES) should further investigate acceptable ranges of ratios of Pi-groups for use in scaling. You stated that "[a]lthough we do not believe that this work is needed for AP1000 certification, this issue is likely to arise with certification of future reactor designs and such a study could tie down this loose end of the code, scaling, applicability, and uncertainty (CSAU) process." The Office of Nuclear Reactor Regulation conveyed this recommendation to RES which plans to present its response at a future Thermal-Hydraulic Phenomena subcommittee meeting.

The staff will continue to engage ACRS during the design certification review. If you should have any questions or comments regarding the AP1000 standard plant design, please contact Mr. Lawrence J. Burkhart, the AP1000 Project Manager, at 301-415-3053 or ljb@nrc.gov.

Thank you for your continued support of the staff's review of the AP1000 standard design.

Sincerely,

Carl J Paperiello for

William D. Travers
Executive Director
for Operations

Project No. 711

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

3

Pressurized Thermal Shock (PTS) Reevaluation Project

W. J. Shack

Current PTS Rule

- 10 CFR 50.61 provides assurance that reactor vessels will have a low likelihood of failure due to PTS
  - Only a few plants will approach current screening criteria during the initial 40 year license period
  - About 10 plants will approach the current criteria during an additional 20 year extended operation

Technical Bases for PTS Rule

Estimation of the frequency of vessel failure requires:

- **Identification of sequences that could lead to rapid cooling of the vessel**
- **Knowledge of the pressure, temperature, and heat transfer coefficient adjacent to the embrittled portion of the vessel**
- **Determination of the thermal stress, fracture toughness and flaw distributions in the vessel**
- **Probabilistic fracture mechanics analyses**

Current Reevaluation Studies

- More complete description of sequences leading in to PTS
- More realistic distributions for flaw density and geometry
- Use of improved probabilistic fracture mechanics code, FAVOR

Current Reevaluation Studies (Cont'd)

- Systematic consideration of uncertainties in:
  - Frequency of initiating events
  - Fracture toughness
  - Thermal-hydraulic conditions

Plant-Specific Studies (Three Plants)

- Current PTS screening criteria are very conservative
  - At current screening limits mean value of failure frequency is about 1×10^{-8}/year
  - Distribution of vessel failure frequencies ranges over three orders of magnitude
  - For plant lifetimes of 60-80 years, failure frequencies range from 5×10^{-10}/year to 5×10^{-8}/year

Current Reevaluation Studies

ACRS Conclusions:

- **An outstanding multidisciplinary study**
- **Demonstrates utility of systematic uncertainty analyses to reach defensible conclusions in the presence of large uncertainties**

Studies (Cont'd)

- Support staff plans for an external peer review of importance of conclusions and technical work
- Need to complete and improve documentation to address ACRS concerns and support peer review

ACRS LETTERS

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

February 21, 2003

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PRESSURIZED THERMAL SHOCK (PTS) REEVALUATION PROJECT: TECHNICAL BASES FOR POTENTIAL REVISION TO PTS SCREENING CRITERIA

Dear Dr. Travers:

During the 499th meeting of the Advisory Committee on Reactor Safeguards, February 6-8, 2003, we reviewed a draft report that the NRC's Office of Nuclear Regulatory Research (RES) staff has prepared to document its work to develop technical bases for revising the pressurized thermal shock screening criteria in the PTS rule (10 CFR 50.61). Our Subcommittee on Materials and Metallurgy also reviewed this matter on February 5, 2003. During our review, we had the benefit of discussions with representatives of the NRC staff and the documents referenced.

CONCLUSIONS AND RECOMMENDATIONS

1. The PTS Reevaluation Project has developed comprehensive technical bases for analyzing the susceptibility of reactor pressure vessels to PTS and to support rulemaking to revise the current PTS Rule 10 CFR 50.61. Plant-specific studies show that the current PTS screening criteria are very conservative for the given plants. This work may also provide a basis for reducing unnecessary conservatism in current regulation on operational limits on pressure vessel heatup and cooldown (Appendix G to 10 CFR Part 50).

2. The draft technical bases summary report needs substantial revision to describe more clearly the basic phenomena, issues, approaches, and conclusions. Topical reports on some important technical tasks have not yet been completed.

3. We support plans for an external peer review of the technical work.

DISCUSSION

The PTS Rule 10 CFR 50.61 was established to ensure the integrity of irradiation-embrittled reactor pressure vessels. Reactor pressure vessel steels undergo a transition from highly ductile behavior at high temperatures to brittle behavior at low temperatures. This change in behavior occurs abruptly over a narrow range of temperatures, and a temperature RT_{NDT} can be defined to characterize the transition in fracture behavior. Under irradiation, the transition temperature RT_{NDT} increases, making the vessel susceptible to brittle fracture at higher temperatures.

Estimation of the frequency of vessel failure requires (1) identification of sequences that could lead to rapid cooling of the vessel and estimation of their frequencies of occurrence; (2) determination of the pressure, temperature, and heat transfer coefficient adjacent to the embrittled portion of the vessel for each of the event sequences and use of these to determine the thermal stress on the vessel and the fracture toughness of the vessel material; and (3) probabilistic fracture mechanics analyses to determine the probability of failure under the induced thermal and pressure stresses on the embrittled vessel.

The studies conducted by the PTS Reevaluation Project to assess the frequency of vessel failure are much more comprehensive than those done in the early 1980s. These recent studies include systematic consideration of uncertainties in (1) the frequency of initiating events for PTS scenarios, (2) the thermal-hydraulic conditions that provide the driving forces for crack propagation and initiation, and (3) the assessment of the fracture toughness of the vessel materials. Substantial work has also been done to develop more realistic distributions for flaw density and geometry and improve the accuracy and rigor of the probabilistic fracture mechanics code, FAVOR, which is used in these analyses.

The results from detailed plant-specific studies of Oconee Nuclear Station, Unit 1; Palisades Plant; and Beaver Valley Power Station, Unit 1, show that the current PTS screening criteria are very conservative for these plants. Two of these plants are among the most susceptible to irradiation embrittlement in the reactor fleet. Moreover, the staff has presented good arguments as to why these results can be considered representative of the entire fleet of pressurized water reactors. The staff also currently has additional studies under way to further confirm the generic applicability of these results.

The distributions of the predicted vessel failure frequency are very broad. There are about three orders of magnitude between the 5th and 95th percentiles of the failure frequency. The distributions are also highly skewed, so that the mean and 95th percentiles are virtually identical. At embrittlement levels corresponding to the current screening criterion, the mean frequency of vessel failure is about 1×10^{-8}/year. This is a factor of about 500 lower than the current acceptance level. For plant lifetimes of 60-80 years, the predicted mean vessel failure frequencies will range from 5×10^{-10}/year to 5×10^{-8}/year.

Based on current estimates, 10 plants will be within 20°F of the current screening criteria at the end of their original 40-year licenses. Because the transition temperature increases about 1°F per year of operation, revision of the current PTS screening criteria could significantly impact the licensees decisions regarding whether to pursue license renewal for these plants.

The staff has concurred with our recommendation in our report of July 18, 2002, that a risk-informed acceptance criterion for vessel failure frequency should be based on considerations of large early-release frequency and not on core damage frequency. The scoping studies presented by the staff suggest that it is likely that the performance of containment systems after vessel failure will be adequate to ensure that a vessel failure frequency criterion of 1×10^{-6}/year will be adequate to ensure that the risk due to PTS is acceptably low. These studies also provide an approach for developing a risk-informed failure frequency criterion. Nevertheless, further consideration of the possibility of late containment failure may be needed and should be pursued if rulemaking is undertaken.

The documentation of the technical bases is currently inadequate and incomplete. Topical reports on some important technical tasks have not yet been completed. For example, no referenceable reports are available on the experiments and analyses that were performed to assess the potential for strong temperature gradients in the downcomer region near the beltline region that would invalidate the one-dimensional treatment of the thermal boundary conditions used in the probabilistic fracture mechanics analyses. Similarly, no referenceable reports are available on the studies undertaken at the University of Maryland that were used to develop a method to address thermal-hydraulic uncertainties, or to document the methods and approaches used for the probabilistic risk assessments used to determine the frequency of PTS events. A meaningful peer review cannot be performed without more complete documentation.

The draft technical bases summary document needs substantial revision to describe more clearly the basic phenomena, issues, approaches, and conclusions. Because this study synthesizes technical information from several engineering disciplines, it is important to explain how these disciplines interact and how the synthesis influences the conclusions. For example, the staff has identified a wide range of changes that reduce conservatism in the analyses. These include changes in the crack distribution model, finer binning of thermal-hydraulic sequences, removal of conservative bias in the toughness model, and crediting of operator actions. The staff also identified changes that increase the failure frequency, such as inclusion of medium and large-break loss-of-coolant accidents and errors of commission. The staff has shown that it has a good understanding of the relative importance of these various factors in producing the change in the predicted frequency of vessel failure. The staff has also made a systematic attempt to assess the impact of uncertainties. A clear explanation of which factors have the largest impact on the change in the predicted frequency of failure would focus attention on understanding those uncertainties that have the greatest impact.

The staff also needs to revise the discussion of the treatment of uncertainties. Although the studies have attempted to distinguish between aleatory and epistemic uncertainties and the FAVOR code implements methods that account for the different ways they impact the failure frequencies, the current document does not always make clear that the epistemic and aleatory uncertainties were correctly handled.

We commend the staff for an outstanding multidisciplinary study and look forward to reviewing the staff's final reports.

Sincerely,

Mario V. Bonaca

Mario V. Bonaca
Chairman

References:

1. Memorandum dated December 31, 2002, from Ashok C. Thadani, Office of Nuclear Regulatory Research, NRC, to Samuel J. Collins, Office of Nuclear Reactor Regulation, NRC, transmitting Draft NUREG-????, "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10CFR50.61)," December 2002.
2. Letter dated July 18, 2002, from George Apostolakis, ACRS Chairman, to William D. Travers, Executive Director for Operations, NRC, Subject: Risk Metrics and Criteria for Reevaluating the Technical Basis of the Pressurized Thermal Shock Rule.

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

July 18, 2002

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: RISK METRICS AND CRITERIA FOR REEVALUATING THE TECHNICAL BASIS OF THE PRESSURIZED THERMAL SHOCK RULE

During the 494th meeting of the Advisory Committee on Reactor Safeguards, July 10-12, 2002, we met with representatives of the NRC staff to discuss the status of the staff's work to identify risk metrics and criteria that can be used for reevaluating the technical basis of the pressurized thermal shock (PTS) rule. During our review, we had the benefit of the documents referenced.

We were previously briefed by the staff on the methodology and initial results of the PTS reevaluation project during our meeting on February 7-8, 2002, and we issued a letter dated February 14, 2002.

OBSERVATION

The proposed options for PTS acceptance criteria do not properly reflect the potential impact of air-oxidation source term on risk.

Discussion

The NRC staff has proposed the following three options for quantitative acceptance criteria for reactor vessel failure frequency.

> A reactor vessel failure frequency of $5x10^{-6}$/year, which is the same as the current PTS acceptance criteria provided in Regulatory Guide (RG) 1.154.
>
> A reactor vessel failure frequency of $1x10^{-5}$/year based on consideration of the core damage frequency (CDF) provided in RG 1.174 and the Option 3 framework for risk-informing 10 CFR Part 50.
>
> A reactor vessel failure frequency of $1x10^{-6}$/year based on consideration of the RG 1.174 large early release frequency (LERF) that is a surrogate for the prompt fatality safety goal and on the Option 3 framework for risk-informing 10 CFR Part 50.

Because of the potentially severe challenge to containment integrity posed by reactor vessel failure resulting from PTS sequences, we believe that a risk-informed acceptance criterion for reactor vessel failure frequency should be based on considerations of LERF and not on CDF. However, the current LERF surrogate goal in RG 1.174 is not a proper starting point for developing an acceptance criterion because the source terms used to develop the current goal do not reflect the air-oxidation phenomena that would be a likely outcome of a PTS event.

There is currently no commonly accepted source term for air-oxidation events. However, we suggest that the "SST1" source term in NUREG/CR-2239 and the resulting calculated consequences at each site be extrapolated to assess the consequences of a postulated range of air-oxidation-induced source terms that would include significant releases of ruthenium, cerium, and actinides. Given such a source term, an acceptance criterion for the frequency of vessel failure from PTS events could be developed directly from the prompt fatality safety goal with due consideration of uncertainties and defense-in-depth.

If the consideration of an air-oxidation source term is too daunting and subject to unacceptable uncertainty, it may be necessary to fall back on a frequency-based approach to identify criteria that would provide assurance that reactor vessel failure from PTS events is very unlikely. The choice of such criteria is a value judgment that should reflect consideration of the Safety Goals and uncertainties.

We believe it is likely that qualitative consideration of the likelihood of containment failure along with the potential consequences of an air-oxidation source term will lead to an acceptance criterion for reactor vessel failure frequency that would be substantially smaller than any of those currently proposed by the staff.

Sincerely,

George E. Apostolakis
Chairman

References:

1. SECY-02-0092, Memorandum dated May 30, 2002, for the Commissioners, from William D. Travers, Executive Director for Operations, NRC, Subject: Status Report: Risk Metrics and Criteria for Pressurized Thermal Shock
2. U.S. Nuclear Regulatory Commission, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development," December 1982.
3. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.
4. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
5. Letter dated February 14, 2002, from George E. Apostolakis, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule.

EDO RESPONSE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 28, 2003

Dr. Mario V. Bonaca, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: PRESSURIZED THERMAL SHOCK (PTS) REEVALUATION PROJECT: TECHNICAL BASES FOR POTENTIAL REVISION TO PTS SCREENING CRITERIA

Dear Dr. Bonaca,

Thank you for your letter dated February 21, 2003, on the above subject. As you are aware, the staff has been working on a formal reevaluation of the technical bases for the PTS rule (10 CFR 50.61) since 1999. This project is a key element of our initiative to risk-inform 10 CFR 50. We greatly appreciate the interactions that the ACRS has had, and continues to have with the staff on this subject. These interactions have facilitated enhanced focus and rigor in this highly complex technical effort.

With regard to your primary conclusions and recommendations stemming from the ACRS meetings on February 6-8, 2003, I would like to offer the following response:

1. We agree with your conclusion that the PTS reevaluation project has established the comprehensive technical bases necessary for consideration of a revision to the PTS rule. However, we recognize that additional work remains to be completed as described in 2 and 3 below. The key stakeholders involved in the project will be working closely together to expeditiously complete the technical basis effort.

2. We also agree that the draft summary report and associated documentation is in need of revision to more clearly describe the basic phenomena, issues and approaches. We appreciate the detailed comments that ACRS has provided both as a committee and individually. The Office of Nuclear Reactor Regulation and industry participants are also currently reviewing the draft summary report. Following consideration of comments from the key stakeholders who have been actively involved in the project (NRR, RES, Industry, ACRS), the report will be finalized. After the report is completed, it will be distributed to the key stakeholders and made publicly available. In addition, the staff will be producing and finalizing a structured set of reports that support the summary report. The overall documentation, summary, and supporting reports will include a systematic assessment of uncertainties in key variables and explain which factors have the largest impact on changes in the predicted reactor vessel failure frequency. We would anticipate further discussions with the ACRS when this documentation is finalized.

3. As you have noted, a formal peer review is a key element of this project. We also agree that revision and finalization of the appropriate documentation is critical in performing a meaningful peer review. The Office of Nuclear Regulatory Research is currently pursuing initiation of a formal peer review in parallel with finalization of the documentation since certain aspects of the review (availability of reviewers, contracting, etc.) are long lead-time items. The staff is currently developing a detailed plan of actions and milestones for the peer review process.

In summary, I would like to express my thanks to the ACRS for providing a continuing and thorough critique of this important effort. As the staff moves into the concluding phase of the effort, we look forward to continuing productive interactions with ACRS.

Sincerely,

William Travers

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 3, 2002

Dr. George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-001

SUBJECT: RISK METRICS AND CRITERIA FOR REEVALUATING THE TECHNICAL BASIS OF THE PRESSURIZED THERMAL SHOCK RULE

Dear Dr. Apostolakis:

Your letter of July 18, 2002, observed that the current staff work to identify risk metrics and criteria that can be used for reevaluating the technical basis of the pressurized thermal shock (PTS) rule does not properly reflect the potential impact of an air-oxidation source term on risk (and the risk acceptance criteria). The letter also noted that the Committee believes the acceptance criteria should be based on consideration of large early release frequency and not on core damage frequency, and that, if the consideration of an air-oxidation source term is too daunting and subject to unacceptable uncertainty, it may be necessary to fall back on a frequency-based approach to identify [acceptance] criteria. Finally, the letter noted that the Committee believed consideration of containment failure and the air-oxidation source term could result in lower acceptance criteria than that now being considered by the staff.

The staff agrees that the issue of an air-oxidation source term could influence the acceptance criteria. In addition, as noted in SECY-02-0092, there are other issues related to the post-RPV failure and accident progression analysis that the staff will be studying further over the next few months. This study is expected to be completed by December 2002. The staff intends to assess, in a scoping manner, the net implications of the set of issues described in SECY-02-0092. As indicated in our December 14, 2000, letter, we had planned to complete some of the scoping studies earlier in the program; however, the focus of the staff effort was shifted to completing the plant analysis and estimating through-wall cracking frequencies by December 2002. The results of the assessment are intended to provide the staff with a better technical basis with respect to possible acceptance criteria.

The staff will contact the ACRS staff to arrange subcommittee briefings on its work on this set of issues.

Sincerely,

William D. Travers /ra

William D. Travers
Executive Director
for Operations

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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ACRS 2003 Report on NRC Safety Research

F. P. Ford

Comments on RES assessment of issues associated with Nuclear Reactor Safety for:

- AP1000
- ESBWR
- ACR-700
- GT-MHR
- PBMR
- IRIS

Overall Conclusions

The Infrastructure Assessment :

- **Is timely**
- **Identifies the technical issues comprehensively**
- **Defines RES-specific activities for FY03**

Long-Term RES Activities

We concur with Long-Term RES activities in the areas of:

Probabilistic Risk Assessment, Instrumentation & Control, Materials Analysis, Structural Analysis, Consequence Analysis, PIRT Process, and Implementation Issues

Long-Term RES Activities (Cont'd)

Specific comments on:

Generic Regulatory Framework, Human Factors, Thermal-Hydraulic Analysis, Neutronic Analysis, Fuel Analysis, Severe Accident & Source Term, and Advanced Computing Capabilities

Generic Regulatory Framework

Option 3 Framework is a reasonable starting point. However some concerns:

- **Need for additional risk metrics e.g., late containment failure**
- **Regulatory objectives vs. frequency/ consequences**
- **Balance between prevention and mitigation vs. uncertainties**

Human Factors Considerations

- **Plant staffing is an issue that NRC will need to address for advanced reactor plants**
- **Technical basis for judging adequacy of staffing levels must be firmly established**

Thermal-Hydraulic Analysis

- The timely qualification and use of TRAC-M code essential to support certification decisions
- Significant challenges in developing confirmatory data and/or subcodes
- Quantification of epistemic uncertainties in thermal-hydraulic codes

Neutronic Analysis

- Maintain ability to conduct independent analyses
- Coupling of TRAC-M code with 3-D PARCS neutronics code essential for passive reactor designs
- Modifications to analysis methods to account for the different features of ACR-700 should be initiated now to facilitate anticipated certification review

Severe Accident and Source Term

- Passive ALWR covered by modified MELCOR code:
  - -PHEBUS-FP for high burnup fuel
  - -MASCA for core retention
- Limited NRC data and analysis to cover ACR-700 configuration
- Limited NRC experience in accident analysis and fission product release for HTGRs

Fuel Analysis

- Continue research on high burnup fuels (62 GWd/t), and extend to higher values
- Little NRC experience for reviewing coated-particle fuels. Initiate long-term efforts to develop capabilities using analysis methods and data available overseas

Impact of Advanced Computer Capabilities

- Consider the impact the increase in computer capabilities that are occurring might have on NRC efficiency and effectiveness

ABBREVIATIONS

ACNW	Advisory Committee on Nuclear Waste
ACR-700	Advanced CANDU Reactor-700
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic depressurization system
ALWR	Advanced Light Water Reactor
AP1000	Advanced Passive Reactor 1000
DSER	Draft safety evaluation report
ESBWR	European Simplified Boiling Water Reactor
ESP	Early site permit
FAVOR	Probabilistic fracture mechanics code
FSER	Final safety evaluation report
FY	Fiscal Year
GT-MHR	Gas Turbine Modular Helium Reactor
GWd/t	Gigawatt day/ton
HTGR	High Temperature Gas-Cooled Reactor
IRIS	International Reactor Innovative & Safe
KTIs	Key Technical Issues

ABBREVIATIONS (Cont'd)

MASCA	Organization for Economic Cooperation and Development (OECD) experimental program for severe accident research
MELCOR	Melting of Core Program
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARCS	Purdue Advanced Reactor Core Simulator
PBMR	Pebble Bed Modular Reactor
PHEBUS-FP	International severe accident fission product research program
PIRT	Phenomena Identification & Ranking Table
PI-groups	Symbols used in scaling analysis
PRA	Probabilistic Risk Assessment
RES	Office of Nuclear Regulatory Research
T/H	Thermal Hydraulic
TRAC-M	Transient Reactor Analysis Code-Modernized
U.S.	United States of America