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

August 31, 1993

MEMORANDUM TO: Samuel J. Chilk, Secretary of the Commission

R. Savio for

FROM: John T. Larkins, Executive Director, ACRS

SUBJECT: ACRS MEETING WITH THE NRC COMMISSIONERS ON SEPTEMBER 9, 1993 - BACKGROUND INFORMATION

The ACRS is scheduled to meet with the NRC Commissioners on Thursday, September 9, 1993, between 2:00 and 3:30 P.M. to discuss items of mutual interest, including the following. Background material related to these matters is attached:

1. Status of ACRS Review of Evolutionary and Advanced Light Water Reactor Designs - C. Michelson, J. Carroll, W. Lindblad, and C. Wylie (PP. 2-60)

2. Selected ALWR Policy Issues - C. Wylie (PP. 61-78)

3. Regulatory Review Group Report - H. Lewis (PP. 79-82)

Attachments: As Stated

cc: ACRS Members
ACRS Technical Staff

ITEM 1: STATUS OF ACRS REVIEW OF EVOLUTIONARY AND ADVANCED LIGHT WATER REACTOR DESIGNS

The Committee previously discussed the status of the advanced reactor reviews with the Commission on March 5, September 11 and December 11, 1992 and May 14, 1993. Since then, a number of Committee and Subcommittee meetings have been or are planned to be held to discuss various aspects of the advanced reactor reviews, mainly, the GE ABWR, EPRI Requirements Document, Test programs in support of the certification of the W AP600 and GE SBWR passive plant designs, Design Acceptance Criteria (DAC), and Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). The following is a brief summary of the status of ACRS reviews of these matters.

EVOLUTIONARY PLANTS

- General Electric Advanced Boiling Water Reactor (GE ABWR)

The ACRS Subcommittee on Advanced Boiling Water Reactors (ABWR) and other subcommittees have held 26 meetings beginning in October 1989, to discuss the NRC staff's Draft Safety Evaluation Report (DSER), the GE Standard Safety Analysis Report (SSAR) for the ABWR, and related matters. The Committee has provided four letters to the EDO and five reports to the Commission on matters related to this review.

The ABWR Subcommittee visited the GE facility in San Jose, California on June 15 and 16, 1993. The purpose of this visit was to gather information associated with the review of the ABWR SSAR. In addition, the Subcommittee held a meeting on June 17, 1993 in San Jose, CA to continue its review of the SSAR. Since then, the ABWR Subcommittee held meetings on July 28, 1993 to discuss fire PRA, fire hazards analysis, and fire barrier design and on September 8, 1993 to discuss portions of the SSAR not covered or completed previously and to obtain a current estimate from the staff regarding their schedule for issuing the FSER. The Committee has not yet received the updated SSAR that incorporates the latest ITAAC submittals. Other subcommittee(s) meetings are scheduled as follows:

- o Severe Accidents - September 22-24, 1993 to discuss severe accident issues and PRA considerations.

- o ABWR - October 26-27, 1993 to begin the review of the staff's FSER.

- o Ad Hoc on Design Acceptance Criteria/Computers in Nuclear Plant Operations - November 2, 1993 to discuss ITAAC/DAC and Computer Verification and Validation

- o ABWR - November 16-17, 1993 to continue the review of the staff's FSER.

- o ABWR - January 25-26, 1993 to complete the review of the staff's FSER.

The Committee anticipates issuing its final report on the final design approval (FDA) for the ABWR by March 1994. This schedule is based on the assumption that GE and the NRC staff will provide the ACRS with reasonably complete documentation. This will enable the Subcommittees on Advanced Boiling Water Reactors and Severe Accidents, and the Ad Hoc Subcommittee on Design Acceptance Criteria to hold meetings as appropriate and gather information for the Committee to use in the discussions of the FSER to complete a final ACRS report on the FDA for the ABWR.

The following documents are attached:

- ACRS report to the Commission dated March 18, 1993. Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule (PP. 7-8)

- ACRS report to the Commission dated October 16, 1992. Subject: Second Interim Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design (PP. 9-12)

- ACRS report to the Commission dated August 12, 1992. Subject: Inspections, Tests, Analyses, and Acceptance Criteria Program for the GE ABWR Design (PP. 13-16)

- ACRS letter to James M. Taylor (EDO) dated August 12, 1992. Subject: ACRS Plan For Reviewing The Application For Certification of the GE Advanced Boiling Water Reactor Design (PP. 17-19)

- ACRS letter to James M. Taylor (EDO) dated April 13, 1992. Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design (PP. 20-28)

- ACRS report to the Commission dated August 13, 1991. Subject: Additional Comment on Schedules for Advanced Reactor Reviews (PP. 29)

- ACRS report to the Commission dated July 18, 1991. Subject: Schedules for Advanced Reactor Reviews (PP. 30)

- ACRS letter to James M. Taylor (EDO) dated July 18, 1991. Subject: Concerns Related to the General Electric Advanced Boiling Water Reactor Design (PP. 31-35)

- ACRS letter to James M. Taylor (EDO) dated November 24, 1989. Subject: Module 1 of the Draft Safety Evaluation Report for the Advanced Boiling Water Reactor Design (PP. 36-39)

• Westinghouse RESAR SP/90 Design

The ACRS review of the Westinghouse's application for the Preliminary Design Approval (PDA) for the RESAR SP/90 design has been completed. The Committee provided a report to the Commission dated December 12, 1990 on this matter.

The following document is attached:

- ACRS Report to the Commission dated December 12, 1990. Subject: Westinghouse's Application for Preliminary Design Approval for the RESAR SP/90 Design (PP. 40-45)

• ABB-CE System 80+ Design

The ACRS Subcommittee on Advanced Pressurized Water Reactors has held six meetings beginning in April 1990 to discuss the ABB-CE Systems 80+ design features and related issues such as the Licensing Review Basis (LRB) document. The Committee provided a report to the Commission dated November 14, 1990 in regard to the LRB. The staff's Draft SER on the Systems 80+ design was provided to the ACRS on October 1, 1992, and a Subcommittee meeting was held on February 10, 1993 to discuss this document. During the Subcommittee meeting, two major issues were discussed. These were human factors engineering and diversity of instrumentation and control. As these issues were not yet resolved, the Committee did not comment on them.

On April 13, 1993, some members of the ACRS Subcommittee on Advanced Pressurized Water Reactors visited the dynamic mockup of the control room at the ABB-CE facility in Windsor, Conn. The next Subcommittee meeting is scheduled for December 8, 1993 to begin review of the Standard Safety Analysis Report for the ABB-CE Systems 80+ design and additional meetings will be scheduled as appropriate.

The following document is attached:

- ACRS report to the Commission dated November 14, 1990. Subject: SECY-90-353, Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor (PP. 46-47)

- EPRI Utility Requirements Document for Evolutionary Plants

The ACRS Subcommittee on Improved Light Water Reactors has held seven meetings to discuss the staff's draft SERs related to various chapters of the EPRI Requirements Document for Evolutionary Light Water Reactor Designs. The Committee provided reports to the Commission dated April 23, 1991 and August 18, 1992 on this matter. The staff plans to issue a supplement to the FSER after all evolutionary policy issues have reached final resolution. The ACRS expects to review the supplement to the FSER.

The following document is attached:

- ACRS Report to the Commission dated August 18, 1992. Subject: Electric Power Research Institute Advanced Light Water Reactor Utility Requirements Document -- Volume II, Evolutionary Plants (PP. 48-51)

PASSIVE PLANTS

- Westinghouse AP600

The Committee and the Subcommittees on Advanced Pressurized Water Reactors/Thermal Hydraulic Phenomena have heard presentations regarding design details for the Westinghouse AP600 passive plant and the test programs proposed by both Westinghouse and the staff in support of the AP600 passive plant design certification. The Committee provided reports dated November 14, 1991 and March 10, April 6, and July 17, 1992 to the Commission in regard to the test programs.

At this time, the Thermal Hydraulic Phenomena Subcommittee and the Committee plan to review both the status of the Westinghouse analytical and experimental programs noted above and the associated NRC staff review of the same during October and November, 1993 meetings. Following this review, the Committee will provide the Commission with formal comments on this matter.

The Standard Safety Analysis Report for the AP600 was issued on June 26, 1992. The Committee will continue its discussion of this matter on a schedule consistent with the development of the staff's SER.

The following document is attached:

- ACRS report to the Commission dated July 17, 1992. Subject: Integral System and Separate Effects Testing in Support of the Westinghouse AP600 Plant Design Certification (PP. 52-56)

• General Electric SBWR

The Committee and the Subcommittees on Advanced Boiling Water Reactors/Thermal Hydraulic Phenomena have heard presentations regarding design details and test programs for the General Electric SBWR passive plant. The Committee provided a report to the Commission dated June 10, 1992 regarding the proposed test programs in support of the SBWR design certification. The Committee will continue its review of the ongoing experimental and analytical programs related to the certification of the SBWR design.

The Standard Safety Analysis Report for the SBWR was received on August 26, 1992. The Committee will continue its discussion of this matter on a schedule consistent with the development of the staff's SER.

The following document is attached:

- ACRS report to the Commission dated June 10, 1992. Subject: Testing and Analysis Programs in Support of the Simplified Boiling Water Reactor Design Certification (PP. 57-60)

• EPRI Requirements Document for Passive Plants

The Committee and the Subcommittee on Improved Light Water Reactors have been briefed on the EPRI Requirements Document for Passive Plant Designs. The ACRS plans to continue its review of this matter after receiving the staff's final SER. The Improved Light Water Reactors Subcommittee is scheduled to meet on October 6, 1993 to begin its review of the staff's SER for the EPRI document.

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

March 18, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: ADVANCED BOILING WATER REACTOR (ABWR) REVIEW SCHEDULE

During the 395th meeting of the Advisory Committee on Reactor Safeguards, March 11-12, 1993, we discussed the staff's revised estimate of the schedule (proposed in SECY-93-041) for completing its review of the ABWR design. We also had the benefit of the documents referenced.

We note that in SECY-93-041, the time proposed for our review of the Final Safety Evaluation Report (FSER) is one month. In our July 18, 1991, report to you on "Schedules for Advanced Reactor Reviews," we agreed with the staff's estimate of three months for completing our review of the FSER. It is still our view that three months will be needed to perform a meaningful review, given the proposed schedule for transmitting the information to us.

Regarding our present ABWR review status, our work on the ABWR design certification application stalled in November 1992, pending the development of additional technical information by General Electric Nuclear Energy (GE) and decisions by the NRC staff on a number of important areas such as:

- design acceptance criteria/inspections, tests, analyses and acceptance criteria, digital control systems, control room and human factor provisions, and severe accident/probabilistic risk assessment considerations

- interface requirements and representative conceptual designs for uncertified portions of the design

- technical resolution of Unresolved Safety Issues and Generic Safety Issues as required by 10 CFR 52.47

- closure of open and confirmatory items in the October 1992 draft of the FSER

- closure of open items and concerns from the ACRS Advanced Boiling Water Reactors Subcommittee meetings of August 19, October 21, and November 18-19, 1992

Our subcommittee meetings with the NRC staff and GE were, in general, limited to consideration of the October 1992 draft of the FSER and the initial submittal and first twenty amendments (through March 13, 1992) of the ABWR Standard Safety Analysis Report (SSAR). We have not met with the staff or GE on these matters since November 1992, although we have planned a subcommittee meeting on severe accidents on March 18, 1993.

We will meet again to complete our review when the staff and GE provide us with reasonably complete final documentation for our consideration. There are now several additional voluminous amendments to the SSAR to consider, and extensive revision of the FSER is likely. From the nature of past ACRS open items and concerns on the ABWR and the uncertainty concerning their resolution, we believe that significant problems may still persist.

If it would expedite the schedule, we would be willing to meet with the staff and GE to review portions of the final FSER and associated SSAR beyond Amendment 20 as they are completed and made available. This would ensure a more timely resolution of any remaining concerns and could shorten the three months otherwise needed for our review of the advance copy of the complete FSER package (referred to in SECY-93-041) and preparation of our final report required by 10 CFR 52.53.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Letter dated February 9, 1993, from Dennis M. Crutchfield, NRR, to Paul Shewmon, Chairman, ACRS, Subject: Review Schedule for the Advanced Boiling Water Reactor (ABWR)
2. SECY-93-041, dated February 18, 1993, for the Commissioners from James M. Taylor, Executive Director for Operations, Subject: Advanced Boiling Water Reactor (ABWR) Review Schedule

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

October 16, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: SECOND INTERIM REPORT ON THE USE OF THE DESIGN ACCEPTANCE CRITERIA PROCESS IN THE CERTIFICATION OF THE GENERAL ELECTRIC NUCLEAR ENERGY ADVANCED BOILING WATER REACTOR DESIGN

During the 390th meeting of the Advisory Committee on Reactor Safeguards, October 8-10, 1992, we continued our deliberations regarding the use of the design acceptance criteria (DAC) process and associated inspections, tests, analyses, and acceptance criteria (ITAAC) in the certification of the General Electric Nuclear Energy (GE) Advanced Boiling Water Reactor (ABWR) design. Our Ad Hoc Subcommittee on Design Acceptance Criteria considered this matter during its October 7, 1992 meeting. This Subcommittee was established to review the DAC process as requested by the Commission in its April 1, 1992, Staff Requirements Memorandum.

During these meetings we considered SECY-92-299, dated August 27, 1992, which is a staff status report on the subject of the development of DACs for the ABWR certification in the areas of instrumentation and controls (I&C) and control room design. It was evident from our meetings that the staff's review of these DACs and preparation of the supporting draft Final Safety Evaluation Report (FSER) chapters will require extensive further work. During these meetings, we had the benefit of discussions with representatives of the NRC staff and GE. We also had the benefit of the documents referenced.

Our first interim report on the DAC process, dated June 16, 1992, focused mainly on the other two DACs proposed by GE for use in certification of the ABWR design, namely, ITAAC 3.7 "Radiation Protection" and ITAAC 3.3 "Piping Design." We concluded that these DACs (with certain clarifications to the language of the drafts we reviewed) can provide an acceptable basis for the staff's final safety determination needed for design certification. We understand that these DACs will be available in final form for completing our review as part of the FSER. The staff is unable at this time to provide a schedule for completion of the FSER.

This interim report deals with the remaining two DACs - control room design, and instrumentation and controls. In our June 16, 1992 interim report, we indicated that these DACs had not been developed to a point where we could offer an opinion as to their acceptability. We did express concerns to the staff on several aspects of these DACs as they existed at that time. The staff has subsequently responded to these concerns.

Control Room Design DAC

Enclosure 3 of SECY-92-299 contains the DAC (i.e., ITAAC 3.6 "Human Factors Engineering") proposed by GE for the ABWR control room design (human factors aspects), a draft of the staff's FSER for Chapter 18 of the Standard Safety Analysis Report (SSAR), "Human Factors," and a Human Factors Review Model developed by the staff. The staff certification of control room design will be based on the design process described in this ITAAC. The implementation of the control room design process will be the responsibility of the combined operating license (COL) applicant or holder.

The draft FSER contains three open items in this DAC area, all involving documentation issues, that are being completed by GE and will then require the review and approval of the staff. These open items appear to be easily resolvable.

We learned at our meetings that GE had submitted a new revision of ITAAC 3.6 since the issuance of SECY-92-299. It was this new material, which had not been completely reviewed by the staff, that we reviewed. Although we had a number of suggested language clarifications, we conclude that this ITAAC (with appropriate modification) will be able to provide an acceptable basis for the staff's final safety determination needed for design certification. We will complete our review of FSER Chapter 18 and this ITAAC when these documents become available in final form.

Instrumentation and Controls (I&C) ITAAC

Enclosure 2 of SECY-92-299 contains the ITAACs proposed by GE for ABWR I&C and a draft of the staff's FSER for Chapter 7 of the SSAR, "Instrumentation and Control Systems." The staff notes that GE will not have submitted complete design information in the I&C area prior to design certification because this is an area of rapidly changing technology. GE proposes the DAC material be included in the Tier 1 design as one system ITAAC (2.75 "Multiplexing") and three generic ITAACs (3.2 "Instrument Setpoint Methodology," 3.4 "Safety System Logic and Control," and 3.5 "Software Development"). The implementation of the design process described in the Software Development ITAAC would be the responsibility of the COL applicant or holder. Our review focused on the Software Development ITAAC which describes a design process as contrasted to a design.

The draft FSER includes five open items and 19 confirmatory items in the I&C area that are being completed by GE and will require the review and approval of the staff.

We learned at our meetings that GE had submitted a new revision of ITAAC 3.5 since the issuance of SECY-92-299. It was this new material, that had not been reviewed by the staff, that we reviewed. We had a number of suggested clarifications to the language of this ITAAC. In addition, there are certain characteristics of software which, when specified at the beginning of the development process, make later assessment far easier. We believe that the staff and GE should include this concept in the Software Development ITAAC. We conclude that this ITAAC has the potential of providing an acceptable basis for the staff's final safety determination needed for design certification. We will continue our review as more information becomes available.

Finally, we are concerned about the significant number of post-design certification activities associated with these two DACs - control room design, and instrumentation and controls. The COL applicant or holder will be responsible for carrying out these activities. This will involve extensive future negotiations with the staff. It will also have the effect of diminishing the value of certified designs and seems to us to be contrary to the spirit of 10 CFR Part 52. We believe that the argument that these DACs represent areas of rapidly changing technology is being overplayed by both the staff and GE in justifying the extent to which the DAC process is being used.

We will keep you informed as our review of the DAC process in the certification of the GE ABWR design continues.

Additional comments by ACRS member Harold W. Lewis are presented below.

Sincerely,

David A. Ward
Chairman

<u>Additional Comments by ACRS Member Harold W. Lewis</u>

I have a reservation about the Committee letter, for the specific issue of software certification. I have already taken (Reference 4) a more relaxed position than the Committee in the general area of DACs. That position reflects my view that we are dealing with a mature industry, not at all inexperienced in the design of modern

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reactors, and therefore requiring a different style of regulation than may have been the case in an earlier period. The most effective role of NRC is through oversight of the safety of the industry product, rather than on certification of each detail. The DAC process lends itself to this kind of regulation, but only in areas in which the staff itself has the experience and expertise necessary to assume this more global role. I hope that the staff will not inhibit the application of modern technology through excessive specificity, as exemplified by the analog backup controversy, on which the Committee has previously commented (Reference 6).

I have a separate nagging problem with the DAC process, as it is now being implemented, one which is exacerbated in this case. The staff is negotiating with the industry not only the potential applicants' programs for compliance with the (still unclear) acceptance criteria, but also the nature of the very requirements that the applicants will later have to meet. It is important to be very circumspect about the NRC's role in this process, lest NRC independence be compromised.

<u>References</u>:

1. SECY-92-299, dated August 27, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design
2. Staff Requirements Memorandum M920305A dated April 1, 1992, from Samuel J. Chilk, Secretary of the Commission, for David A. Ward, Chairman, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992
3. GE Nuclear Energy, "Tier 1 Design Certification Material for the GE ABWR," dated June 1992
4. Report dated February 14, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews
5. Report dated June 16, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Interim Report on the Use of Design Acceptance Criteria in the Certification of the GE Nuclear Energy Advanced Boiling Water Reactor Design
6. Report dated September 16, 1992, from David A. Ward, Chairman, ACRS, to the Hon. Ivan Selin, Chairman, NRC, Subject: Digital Instrumentation and Control System Reliability

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

August 12, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: INSPECTIONS, TESTS, ANALYSES, AND ACCEPTANCE CRITERIA PROGRAM FOR THE GE ABWR DESIGN

During the 388th meeting of the Advisory Committee on Reactor Safeguards, August 6-8, 1992, we reviewed a sample of the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) which are being prepared by GE Nuclear Energy (GE) as a part of its application for certification of the ABWR design. This topic was also reviewed at a joint meeting of our Subcommittees on Decay Heat Removal Systems and Advanced Boiling Water Reactors on August 5, 1992. During these meetings, we had the benefit of presentations by members of the NRC staff and by representatives of GE. Our review has been in response to a request by the Commission made at our meeting with them on March 5, 1992, and confirmed in a Staff Requirements Memorandum dated April 1, 1992. We also had the benefit of the documents referenced.

ITAAC are an important part of Tier 1 submittals which the NRC requires of applicants for design certification under Part 52. They are intended to abstract from the more voluminous source, the Standard Safety Analysis Report (SSAR), the information needed by the NRC staff to make its final safety determination and to ensure that this information is agreed to at the time of design certification and verified in the completed plant. The form and content of individual ITAAC are still being developed by an iterative process between GE and the NRC staff.

There are several types of ITAAC, as described by the staff:

Systems
Generic
Interface
Design Acceptance Criteria (DAC)
Combined Operating License (COL)

Our present review has been confined to the general program and to the first type, which includes the largest number of individual ITAAC. We were told that the entire plant design can be described in terms of about 140 systems. Of these, GE has proposed that about 85 have sufficient safety significance to be covered by individual ITAAC. These comprise the "Systems ITAAC." We have reviewed 5 of these 85 in some detail, as a means for evaluating the ITAAC process.

We intend to continue our review by investigating examples of the Generic and Interface ITAAC. We were told there are nine Generic ITAAC for the ABWR, covering subjects which apply to many or all systems, such as welding and equipment qualification requirements. We have commented on DAC in an interim report of June 16, 1992. The COL ITAAC, which will be concerned with such matters as operator training, will be developed by a COL applicant after the design certification. We would expect to review these in the future when appropriate.

We conclude from our review that the ITAAC process appears to be generally well founded and can be made to work as the staff and GE visualize. The general form and scope of the individual ITAAC we studied were satisfactory. There is, however, a problem with content of the ITAAC. Although the examples we examined were a part of what was described as the final Stage 3 GE submittal, there was a significant lack of consistency, accuracy, and completeness. We were informed by both the staff and GE that this is a problem beyond the five examples we selected for our review. Both are individually committed to major efforts to improve the quality of the content of all ITAAC.

We were told by the Director of NRR that he plans an extensive and in-depth review of the submitted ITAAC and will not recommend approval of a Final Design Approval (FDA) until the results of the review are fully satisfactory. This could mean a delay in the presently projected date for the FDA issuance. For its part, GE expressed its commitment to respond to problems indicated by the staff review and to conduct its own quality review in parallel. GE intends to ensure consistency among ITAAC and other Tier 1 and Tier 2 documents. In addition, we were told that NUMARC intends to carry out an independent review of the ABWR ITAAC. GE already has comments from utilities on the Stage 3 ITAAC. These will be incorporated into the continuing iterations between the staff and GE.

We are concerned with the structural adequacy of walls and associated penetrations within buildings housing critical systems outside of primary containment during possible fires, floods, or pipe breaks. It was not clear from the material presented to us how structural requirements for these will be verified through the

ITAAC process. We expect to pursue this matter at a future meeting.

A PRA has been performed for the ABWR design and certain conclusions about the safety of the design can be drawn from this. In performing the PRA, many assumptions were necessary about the performance reliability of components and systems. There appears to be no means by which Tier 1 requirements (e.g., ITAAC) will ensure that components and systems in the plant can be expected to have reliabilities which are consistent with those assumed in the PRA. The SSAR provides some information on this, but does not close the loop. We were told that appropriate reliability values for components and systems will be ensured through a reliability assurance program developed by a COL applicant. We believe this matter deserves more study.

In our report to you of September 10, 1991 on ITAAC, we expressed a preference for Option 3 in SECY-91-210 which would allow for completing the ITAAC after issuance of the FDA for ABWR. The staff position is that completion of the ITAAC before the FDA is essential. Given our evaluation of the current status of ABWR documentation, we agree.

We trust the above discussion and comments have been helpful. We expect to complete our review in the near future.

Sincerely,

David A. Ward

David A. Ward
Chairman

<u>References</u>:

1. SECY-91-210, dated July 16, 1991, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval (FDA).
2. Staff Requirements Memorandum dated April 1, 1992, from Samuel J. Chilk, Secretary, for David A. Ward, ACRS, Subject: Periodic Meeting with the Advisory Committee on Reactor Safeguards on March 5, 1992.
3. Excerpts of Inspections, Tests, Analyses, and Acceptance Criteria from GE Nuclear Energy Report: "Tier 1 Design Certification Material for the GE ABWR," dated June 1992, as follows:

- Standby Liquid Control System (2.2.4)
- Residual Heat Removal System (2.4.1)
- Reactor Building Cooling Water System (2.11.3)
- Emergency Diesel Generator System (Standby ac Power Supply - 2.12.13)
- Control Building (2.15.12)

4. Report dated September 10, 1991, from David A. Ward, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications.

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

August 12, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: ACRS PLAN FOR REVIEWING THE APPLICATION FOR CERTIFICATION OF THE GE ADVANCED BOILING WATER REACTOR DESIGN

During the 388th meeting of the Advisory Committee on Reactor Safeguards, August 6-8, 1992, we discussed our plan for reviewing the GE application for certification of the Advanced Boiling Water Reactor (ABWR) design. Our goal is to complete this review prior to the issuance of the Final Design Approval (FDA) that is scheduled for December 1992. Subject to receiving relevant information from GE and the NRC staff in a timely manner, we plan to meet this goal. Any significant delay on the part of GE and/or the NRC staff in providing necessary information to support our review will delay the completion of our review.

Our plan for review of the matters associated with the ABWR design is as follows:

I. Final Safety Evaluation Report (FSER), Certain Other Staff and GE Licensing Documents, and Remainder of the ABWR Standard Safety Analysis Report (SSAR) Submittals

NRC Staff's Schedule for Submittal of the FSER - In our April 13, 1992 letter to you regarding the ABWR Draft Safety Evaluation Report (DSER), we stated, "If we are to provide our final report on this subject in December 1992, it will be necessary that we receive a complete and final SER no later than early September 1992." Although the staff plans to issue the FSER by early September 1992, we understand that it will not be complete, and will contain a large number of open items. The staff plans to issue Supplement 1 to the FSER by late October 1992, documenting the resolution of the open items. Resolution of the remaining open items, if any, is expected to be addressed in subsequent supplements. The staff is not sure at this time whether there will be multiple supplements, or on what schedule they will be issued.

Schedule for ACRS Review - In order to support the NRC's current schedule for issuing the FDA for the ABWR design, we plan to complete our final report to the Commission during our December 10-12, 1992 meeting. Our Subcommittee on Advanced Boiling Water Reactors has scheduled the following meetings to review the ABWR design:

August 19, 1992 - To discuss GE's and NRC staff's responses to the issues included in our April 13, 1992 letter.

September 23-24, 1992 - To start the review of the ABWR FSER, certain other GE and NRC staff licensing documents, and the remainder of the SSAR submittals.

October 21-22, 1992 - To continue the review of the FSER, other licensing documents, and the remainder of the SSAR submittals.

November 18, 1992 - To review Supplement 1 to the FSER and any residual issues.

If we are to complete our final report in December 1992, we will not be able to perform a meaningful review of the supplements issued after October 1992.

II. Design Acceptance Criteria (DAC)

NRC Staff's Schedule for Submittal of the FSER - At the end of May 1992, the staff provided its draft SER (SECY-92-196) on the DACs related to Radiation Protection and Piping Systems. The staff expects to provide its draft SER on the remaining DACs, in the areas of Man/Machine Interface and Control and Protection Systems, by early September 1992.

Schedule for ACRS Review - On June 16, 1992, we provided an interim report to the Commission that included specific comments on the Radiation Protection and Piping Systems DACs; owing to lack of detailed information, we provided only general comments on the Man/Machine Interface and Control and Protection Systems DACs. The staff plans to provide detailed information on the DACs related to Man/Machine Interface and Control and Protection Systems and updated information on the other two DACs by early September 1992. Based on this schedule, our Ad Hoc Subcommittee on Design Acceptance Criteria plans to schedule a meeting during September or early October 1992 to discuss this matter. We plan to complete a final report on these four DACs during our October 8-10, 1992 meeting.

III. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

NRC Staff's Schedule for Submittal of ITAACs - The staff has already provided ITAACs for a number of ABWR systems to the ACRS. The staff is reviewing these ITAACs and identifying areas where additional information is needed from GE.

Schedule for ACRS Review - In the April 1, 1992 Staff Requirements Memorandum (SRM), the Commission requested that we review in some detail representative ITAACs submitted by GE, and provide recommendations to the Commission by August 21, 1992. Accordingly, a Subcommittee meeting was held on August 5, 1992, to review the following ITAACs:

- Standby Liquid Control System (suggested by Commissioner Rogers during the March 5, 1992, meeting between the ACRS and the Commissioners)
- Residual Heat Removal System
- Reactor Building Cooling Water System
- Emergency Diesel Generator System (Standby ac Power Supply)
- Control Building

The full Committee discussed these ITAACs with representatives of the NRC staff and GE during its August 6-8, 1992 meeting and provided a report to the Commission dated August 12, 1992.

IV. Summary

Completion of our review of the above-mentioned items in accordance with the schedule noted above depends upon timely receipt of relevant information and appropriate support by the staff and GE. If the staff has any problem in supporting any of the meetings noted above, we would like to hear from you as soon as possible.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

April 13, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: REVIEW OF THE DRAFT SAFETY EVALUATION REPORTS ON THE GE ADVANCED BOILING WATER REACTOR DESIGN

During the 383rd and 384th meetings of the Advisory Committee on Reactor Safeguards, March 5-7 and April 2-4, 1992, we discussed the Draft Safety Evaluation Reports (DSERs) on the Advanced Boiling Water Reactor (ABWR) design which is described by GE Nuclear Energy (GE) in its Standard Safety Analysis Report (SSAR), as amended, and for which GE has applied for design certification in accordance with 10 CFR Part 50, Appendix O. The DSERs which are the basis for this report were sent to the Commissioners for information as six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These generally cover the SSAR and its first eighteen amendments. Our Subcommittee on Advanced Boiling Water Reactors discussed these papers with representatives of GE and the NRC staff during its meetings on September 18 and October 23, 1991 and January 23-24 and February 20-21, 1992. We also had the benefit of the documents referenced.

Our first report to you concerning the DSER for this project was dated November 24, 1989. That report conveyed our comments on Module 1 of the design (former GE designation). We also sent a report to you on July 18, 1991, outlining several ABWR design concerns that developed during subsequent review.

We note a marked improvement in the quality of the staff's DSER evaluations since our November 24, 1989 report. The staff reviewers appear to be following the guidance outlined in the applicable Standard Review Plans (SRPs) to the extent possible, and they are asking good in-depth questions in most areas.

The SECY-91-161 schedule indicates that the Final Design Approval (FDA) is to be issued before the end of Calendar Year 1992. If we are to provide our final report on this subject in December 1992, it will be necessary that we receive a complete and final SER no later than early September 1992. There are now more than three hundred open items in the DSERs, many of which are major. In

addition, there is a number of important policy issues which are unresolved. With the staff programs in place, it is probable that these issues can be resolved. However, this is a large undertaking, and we have concerns about whether it can be accomplished on the schedule now indicated.

In the course of our review, we have identified technical issues for which resolutions should be achieved before we write our final report. These are listed and discussed as follows:

1. Control Building Flooding

   The proposed ABWR plant design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building, with the essential 250 V dc battery rooms and the main control room at a higher elevation, but still below ground.

   Our concern with this arrangement is the potential for control building flooding due to an unisolated break in the Reactor Service Water (RSW) System which provides cooling water from the Ultimate Heat Sink (UHS) to the RBCW System. The proposed UHS is a ground-level spray pond which we assume to be at building grade and likely to contain sufficient water to flood the control building.

   The staff should obtain sufficient information on the interface and conceptual design of the RSW System and UHS to support an adequate evaluation of the flooding potential. The staff's evaluation should include consideration of isolation valve arrangements, the feasibility of and time available for response, and the assumption of a single active component failure during the response. The design information and flooding analysis should be included in the SSAR.

2. Adequacy of Physical Separation

   Pipe breaks, internal plant flooding, and external events such as fire are of major concern if their effects cannot be confined in order to protect required safe-shutdown equipment. We believe that the key to confinement is the provision of appropriate separation barriers. However, a classical barrier such as the 3-hour-rated fire barrier wall and its penetrations (e.g., doors and dampers) may not, of itself, be sufficient to ensure separation under (a) the combined effects of pressure, heat, and smoke from a fire, and the flooding which results from fire mitigation, (b) the effects of pipe whip, jet impingement, or compartment pressurization due to pipe breaks, or (c) the influx of water and hydrostatic pressure buildup due to internal floods.

We believe that the SSAR should describe and the staff should evaluate the adequacy of proposed separation barriers for the full range of events and conditions for which separation must be ensured. We continue to recommend that systems required for safe shutdown not share a common Heating, Ventilating and Air Conditioning (HVAC) System during normal plant operation. The secondary containment HVAC System for the ABWR is such a shared system.

3. Protection of Environmentally Sensitive Equipment

The ABWR makes extensive use of environmentally sensitive equipment (including solid-state electronic components) for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe breaks, fire, internal flooding, or loss of room cooling may create an adverse environment. Such environments need to be identified in the SSAR to ensure appropriate environmental qualification of the equipment.

4. Review of Chilled-Water Systems

The ABWR uses large chilled-water systems to provide essential environmental cooling, which in turn includes cooling of the solid-state electronic components. Because there was no SRP for chilled-water systems, the staff used other guidance such as SRP Section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when the safety evaluation was performed. However, this guidance is not appropriate for the evaluation of refrigeration systems.

The NRC staff needs to evaluate the performance of chilled-water systems under varying accident heat loads and during loss-of-offsite-power events, and to consider their ability to restart and function after a prolonged station blackout. The DSER sections which should evaluate the performance of large chiller packages do not address these issues. We believe they should.

5. Use of Leak-Before-Break Methodology

It is our understanding that GE will not propose the use of leak-before-break methodology for the ABWR standard plant. Thus, the DSER should be revised to ensure that consideration is given to pipe break effects for all systems and locations. This may introduce additional structural protection and environmental qualification requirements in the SSAR.

6. Use of Integral Low-Pressure Turbine Rotors

In our July 18, 1991 report to you, we recommended that the staff review the issues involved with the use of integral low-pressure (LP) turbine rotors. It is our understanding that this new design for LP rotors will be used for the ABWR. (Rotors of this type are being used in rotor replacement programs at currently operating plants.) The practice of turbine manufacturers has been to bore the centerline of this type of rotor to remove impurity inclusions. We were concerned that the use of unbored rotors was being contemplated. The Electric Power Research Institute (EPRI) has recently added a requirement in its Advanced Light Water Reactor Utility Requirements Document (URD) that LP rotors be center-bored.

7. Cavity Floor Area Beneath Reactor Vessel

The cavity area beneath the reactor vessel is sized to meet the EPRI URD specification of 0.02 m^2/Mwt. The ABWR design includes flooding of the cavity. Little consideration has been given to how this should be accomplished. There is little evidence that the planned cavity area will lead to quenching following flooding or that the ABWR flooding plans will not lead to ex-vessel steam explosions. Further attention needs to be given in the SSAR as to when and how fast the cavity should be flooded in order to avoid exacerbating a core-melt accident if it should occur.

8. Adequacy of the ABWR PRA

It is impossible to determine whether the PRA submitted by the applicant will be adequate for a safety determination absent information on how it is to be used by the staff. In our February 14, 1992 report to the Commission on the Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews, we commented on the need for guidance on the use of PRA in the review of new plant designs. At this point the applicant has submitted a PRA, a contractor has performed an extensive review, and the staff has prepared a DSER. However, the use of the PRA in the design certification process is still undefined.

Presumably, the results of the PRA will be used in the course of the staff's determination that the design is expected to produce a nuclear power plant that has an appropriate response to severe accidents. In the Severe Accident Policy Statement, the Commission indicated that a PRA would be required for each new design, and that the results of this PRA would be part of the information which would guide the staff in its determination that a design is adequate to deal with severe

accidents. The policy statement published in the Federal Register of August 8, 1985, also states that "Accordingly, within 18 months of the publication of this Severe Accident Policy Statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs...." The Statement says further, "The PRA guidance will describe the appropriate combination of deterministic and probabilistic considerations as a basis for severe accident decisions."

The staff has yet to produce the promised guidance. We urge that the staff formulate a set of criteria that it plans to use in making severe accident decisions. This should include the way in which the results of a PRA are to be used in the process (not just whether the PRA has been done properly).

9. Containment Hydrodynamic Loads

Air-clearing loads on containment structures are the result of a complex process resulting from the drywell air being forced into the wetwell by the primary system blowdown. The water in the vent system is pushed down and out until the horizontal vents are cleared. The water-clearing process produces a jet of water into the suppression pool which causes a load on the outer part of the wetwell wall. This water clearing is followed by an air-steam mixture which creates a large bubble as it exits into the pool. The steam condenses but the air expands forcing the water above it up into the wetwell air space. The wetwell air space is compressed due to the momentum of the water in the layer above the bubble.

The wetwell air space will be subjected to an energetic two-phase eruption as a result of the air-clearing process. The vacuum breakers which are in the vicinity will be exposed to this environment unless protected. The SSAR should describe what the environment will be and what protective measures, if any, are needed to ensure survival of the vacuum breakers. If a vacuum breaker does not close, the suppression pool is bypassed and the wetwell/drywell pressures will rise at a rate dictated by the capability of some means other than the suppression process (e.g., containment sprays) to remove heat and condense steam. The SSAR should contain an analysis of such a situation.

The early work to address problems arising from analyses of the Mark I, II, and III containments is not sufficient to address similar processes that will occur following a LOCA in an ABWR containment. The ABWR is different for two reasons: (a) the volume of the wetwell air space in the ABWR is approximately that of a Mark II, and (b) the impact of the

air-clearing loads will be alleviated somewhat because the expected blowdown flows are much smaller than those expected in a Mark I or Mark II. Nevertheless, the combination of a much smaller wetwell and the lower mass flow from the break have not received sufficient attention to be written off by the staff or GE without further analysis or experimental investigation. We are not aware of any testing of the ABWR type geometry. We believe there are sufficient differences in both geometry and LOCA characteristics to require further evaluation of the air- clearing phase of the LOCA by more extensive analysis and/or experimental investigation.

10. <u>Adequacy of SSAR Treatment of the Reactor Water Cleanup System</u>

We performed a review of the Reactor Water Cleanup (RWCU) System using our own staff. This system was chosen because it is a non-safety system located outside of primary containment, but inside the building which houses engineered safety features. It uses pipes up to 8-in. nominal diameter whose rupture would result in a LOCA and a source of serious environmental disruption in the building. This system is not seismically qualified or built to quality assurance standards.

Our review identified a number of deficiencies in the SSAR, some of which are listed below:

- There is little useful information presented in the SSAR that describes how the Japanese codes and standards used for the RWCU System design can be converted to domestic design standards. The Quality Group classifications for certain portions of the RWCU System are inconsistent with the Japanese code-related classifications shown on the Piping and Instrumentation Diagrams. The Safety Class/Quality Group transition between the piping inside primary containment and that outside primary containment is not in accordance with ANSI/ANS safety class standards for BWR fluid systems.

- The questionable ability of system isolation valves to close under large-break-LOCA conditions has been the subject of extensive NRC testing and a Generic Letter (GL 89-10). However, the SSAR specifies no special performance requirements for these valves.

- The safety-grade leak detection and isolation system which actuates the system isolation valves was not described in detail sufficient to support an assessment of its adequacy.

- The ABWR PRA did not evaluate as initiating events RWCU System line breaks (or other LOCAs) outside the primary

containment. The exclusion of these breaks was based erroneously on an analysis of the effects of suppression pool bypass events on overall risk. However, the analysis failed to take into account that the bypass path (e.g., RWCU System pipe break) could be the initiator for the core-damage event.

- The PRA analysts took credit for the RWCU System as a heat removal system in all sequences where reactor pressure is assumed to remain high. The analysts assumed that the capacity of the non-regenerative heat exchanger (NRHX) is adequate to remove the decay heat. The capacity appears to be adequate; however, our calculations indicate that the outlet temperatures on the RWCU System side and cooling water side of the NRHX would exceed the design limits for the piping. Furthermore, a temperature sensor between the NRHX and the RWCU System pumps in the present design would automatically isolate the NRHX on high temperature, making it unavailable.

The items mentioned above are among a number of issues that were identified. It is important for the staff to ensure that the shortcomings of the RWCU System and PRA related portions of the SSAR are not indicative of problems in the remainder of that report.

11. Plant Design Life and Aging Management

We recommend that the SSAR clearly define the scope of the 60-year design life for the ABWR and describe a program plan for achieving it. This program should include those aging management measures which are necessary to maintain the plant within its design basis throughout its design life. This program should specify the original design and application criteria and, where required, the projected refurbishment or replacement requirements with appropriate rationale. To the extent applicable, the lessons learned from the NRC's Nuclear Plant Aging Research Program as well as other aging research projects should be incorporated into this program.

We note that the EPRI URD (Volume II, Chapter 1, Paragraph 3.3) includes a requirement for a plant design life of "60 years without necessity for an extended refurbishment outage," and discusses the requirements for its achievement in Paragraph 11.3.

12. Station Grounding and Surge Protection

Chapter 8 of the ABWR SSAR defines the scope of and specifies the requirements for the electrical power systems. The scope is limited to the onsite electrical power systems and to the interface requirements with the offsite electrical power systems.

Notably absent are lightning protection, station grounding systems, and surge protection measures which are necessary to protect plant personnel and equipment during normal and abnormal conditions. These measures are required to eliminate or reduce electrical shock hazards to personnel, and to protect systems and equipment against damage or misoperation as the result of lightning strikes, switching operations, electrical arcs, short circuits, static electricity, etc. These protective measures and their interface requirements should be included in the SSAR.

The ABWR makes extensive use of sensitive solid-state electronic components for essential protection, control, and data transmission functions. These components should be protected from extraneous electrical impulses that will damage them or cause improper performance. To the extent practical, these components should be isolated from potential adverse signals that may be transmitted over control or data links from remote locations, meteorological stations, switchyards, etc.

We note that the EPRI URD (Volume II, Chapter 11, Item 9, "Electrical Protective Systems") addresses requirements for these systems. We recommend that these grounding, surge protection, and isolation features be included in the SSAR.

13. Corrosion Control for Structures

The SSAR should include an interface requirement for a corrosion control program to identify the potential for the corrosion of structures and components and to determine the corrective measures to be taken. The program should commence prior to the completion of the detailed design of building substructures and underground installations. The program should consider the potential for corrosion from galvanic direct currents which may flow as the result of copper ground mats on site, including the electrical switching stations' ground mats. The potential for corrosion of containment building substructures and liners should be considered. The mitigation measures may include coatings, wrappings, cathodic protection, electrical bonding, elimination of galvanic currents, or other mitigation means.

We do not expect to receive a separate reply to the above items if they are covered appropriately in the final SER. We will keep you informed of any additional concerns as our review proceeds.

Sincerely,

David A. Ward

David A. Ward
Chairman

<u>References</u>:

1. GE Nuclear Energy, Standard Safety Analysis Report, "Advanced Boiling Water Reactor," Chapters 1 through 20 (Amendments 1 through 18)
2. SECY-91-153, dated May 24, 1991, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Safety Evaluation Report (DSER) on the General Electric Company Advanced Boiling Water Reactor Design Covering Chapters 1, 2, 3, 4, 5, 6, and 17 of the Standard Safety Analysis Report (SSAR)
3. SECY-91-235, dated August 2, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapters 1, 3, 9, 10, 11, and 13 of the SSAR
4. SECY-91-294, dated September 18, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapter 7 of the SSAR
5. SECY-91-309, dated October 1, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapter 19 of the SSAR, "Response to Severe Accident Policy Statement"
6. SECY-91-320, dated October 15, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Advanced Boiling Water Reactor Design Covering Chapter 18 of the SSAR
7. SECY-91-355, dated October 31, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapters 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15 of the SSAR
8. Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document" (Volume II)/ALWR Evolutionary Plant, Revision 3, Issued November 1991

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

August 13, 1991

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: ADDITIONAL COMMENT ON SCHEDULES FOR ADVANCED REACTOR REVIEWS

In our report to you of July 18, 1991, on "Schedules for Advanced Reactor Reviews," we noted that the time required for Committee review of the final Safety Evaluation Reports (SERs) and Final Design Approvals will be three months, as stated in the text of SECY-91-161, rather than two months as shown on the bar charts. We failed to note that the three months review time (starting at time of receipt) also applies to the draft SERs. Except for ABWR, the bar charts show only one month for ACRS review. The text is silent on this point.

Sincerely,

David A. Ward

David A. Ward
Chairman

Reference:
U.S. Nuclear Regulatory Commission, SECY-91-161, dated May 31, 1991, from J. Taylor, Executive Director for Operations, for the Commissioners, Subject: Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions

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

July 18, 1991

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: SCHEDULES FOR ADVANCED REACTOR REVIEWS

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we discussed the staff's proposed "realistic" schedules identified in SECY-91-161 for completing the reviews of the evolutionary and passive advanced light water reactor (ALWR) design certification applications and the review of the Electric Power Research Institute's (EPRI) ALWR Utility Requirements Document. We had the benefit of presentations by and discussions with members of the NRC staff and NUMARC, as well as the documents referenced. Consideration of this matter by the Committee was based on the request of the Commission, as reflected in Staff Requirements Memorandum M910607A dated June 18, 1991.

We believe that, barring unforeseen circumstances, the ACRS will be able to meet these schedules. Note, however, that the time required for Committee review of the final SERs and FDAs will be three months, as stated in the text of SECY-91-161, rather than two months as shown on the bar charts.

Sincerely,

David A. Ward
Chairman

References:

1. U.S. Nuclear Regulatory Commission, SECY-91-161, dated May 31, 1991, from J. Taylor, Executive Director for Operations, for the Commissioners, Subject: Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions
2. Electric Power Research Institute, Utility Requirements Document, June 1986
3. Memorandum dated June 18, 1991 from Samuel J. Chilk, Secretary of the Commission, for David A. Ward, ACRS, and James M. Taylor, EDO, Subject: Staff Requirements - Periodic Meeting with the ACRS, June 7, 1991

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

July 18, 1991

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: CONCERNS RELATED TO THE GENERAL ELECTRIC ADVANCED BOILING WATER REACTOR DESIGN

During the 375th meeting of the Advisory Committee on Reactor Safeguards, July 11-13, 1991, we discussed the status of the Advanced Boiling Water Reactor (ABWR) design, described in the Standard Safety Analysis Report (SSAR), for which the General Electric Company (GE) has applied for design certification in accordance with 10 CFR Part 50, Appendix O. Our Subcommittee on Advanced Boiling Water Reactors also discussed this matter during its meetings on October 31, 1990, and May 30, 1991, with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

Our previous letter to you concerning the ABWR design was dated November 24, 1989, and conveyed our comments on Module 1 of the Draft Safety Evaluation Report (DSER). Since this letter, we have been kept apprised of the design and the status of the review while awaiting receipt of additional DSERs. The staff now says that DSER preparation by modules will be discontinued in favor of preparation by SSAR chapters and Standard Review Plan (SRP) sections.

To ensure the completeness of our review, it will be necessary to account for any additions or revisions to each DSER as forwarded by a SECY subsequent to issuance of our respective comment letter. An arrangement acceptable to us is needed to ensure the identification of any additions or revisions, and we should agree on an appropriate time for their review. Our comments will not be complete, however, until we have submitted a report to the Commission concerning the final SER on which we expect to comment by mid-November 1992.

Our activities subsequent to the completion of our November 1989 letter have focused on several design concerns that were discussed with GE and the NRC staff in an effort to ensure an early awareness and understanding. We believe that it is appropriate to document them here for timely consideration and resolution in appropriate DSER sections. We expect to have additional items later. We do

not expect separate replies to our concerns provided the staff responds in the appropriate DSER.

1. Control Building Flooding

The proposed ABWR design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building with the essential 250-V. DC battery rooms immediately above, and the main control room at the next higher elevation. This arrangement places the main control room below ground grade. Our concern with this arrangement is the potential for control building flooding due to an unisolated break in the open-cycle cooling water piping or components inside the building. The ultimate heat sink (cooling pond) is likely to provide sufficient water to flood the building to near ground grade.

2. Physical Separation Barriers

Internal plant flooding and external events such as fire are of major concern if their effects cannot be confined to a single division of required safe-shutdown equipment. We believe that the key to confinement is the provision of an appropriate separation barrier. However, a classical barrier such as the 3-hour-rated fire barrier may not of itself, be sufficient to ensure divisional separation under the combined effects of pressure, heat, smoke, and flooding which accompany a fire and its mitigation. Also, it would appear from the SRP that the effects of delayed suppression on room temperature, pressure, and barrier leakage need to be considered when determining that safe shutdown can be achieved. We remain unconvinced that divisional separation barriers for the ABWR have been adequately prescribed for the range of events and conditions during which they must provide separation.

Of particular concern is a diesel fuel fire which may be subject to delayed suppression in the ABWR diesel generator rooms which are located inside the reactor building. It is not clear how these rooms will be qualified by design or testing to withstand burning fuel if spread across the floor by a fuel line rupture. Furthermore, it is not apparent how the compartment doors will be qualified for this condition or whether they can confine the fuel to the room. If manual mitigation is required, a fire barrier door must be opened. It is not certain that this can be achieved safely or that the external environmental effects of a prolonged opening of the door have been considered.

3. Environmental Protection for Solid-State Electronics

The ABWR makes extensive use of solid-state electronic components for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe rupture, fire, internal flooding, or loss of room cooling may create an adverse environment. The response of such components to the environmental change may be unpredictable and lead to unacceptable system interactions or responses. The behavior of solid state electronic components in environments created by off-normal or accident situations needs to be considered before the adequacy of any physical separation and environmental control measures can be evaluated.

4. Review of Chilled-Water Systems

The ABWR makes extensive use of large chilled-water systems to provide essential environmental cooling functions including those for the solid-state electronics. Since there is no SRP for chilled-water systems, the staff uses other guidance such as SRP Section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when performing its safety evaluation. This guidance does not include evaluation of the large refrigeration equipment that is required for chilling the closed-cycle cooling water.

The NRC staff and GE need to evaluate the safety implications of chilled-water systems, including performance under varying accident heat loads, loss-of-offsite-power loading characteristics, and ability to restart and function after a prolonged station blackout. The NRC staff should develop appropriate guidance for such reviews by preparing a suitable SRP now.

5. Use of Leak-Before-Break Methodology Outside of Primary Containment

In our report of March 14, 1989 to then NRC Chairman Zech on "Additional Applications of Leak-Before-Break Technology," we expressed our belief that an avenue for consideration of further extension of the leak-before-break (LBB) concept should exist. This is still our position. We are concerned that the NRC staff is not giving serious consideration to GE proposals to extend the concept to systems outside of the primary containment because the staff feels constrained by General Design Criterion 4 which does not propose review of methodology.

We would like to see a renewed effort by GE and the NRC staff to determine if a real potential for substantial safety and/or economic benefits can be realized in applying properly the LBB concept outside of the primary containment.

6. Use of Integral Low-Pressure Turbine Rotors

The catastrophic failure of a low-pressure (LP) turbine rotor can lead to high-energy missiles that are capable of damaging safety-related equipment. The domestic turbine manufacturers (General Electric and Westinghouse) have been using an LP turbine design for large turbine generators consisting of a relatively small-diameter bored shaft with shrunk-on and keyway locked blade ring disks. The manufacturers are now offering an integral LP turbine rotor machined from a single large-diameter forging. A rotor of this design would operate at much higher stresses than the shaft of a shrunk-on disk rotor.

We were told by the Electric Power Research Institute (EPRI) representatives that a decision has not as yet been made with respect to a requirement in the ALWR Utility Requirements Document for boring the LP turbine rotors. Boring has historically been performed to remove impurity inclusions near the forging centerline. Such inclusions are stress risers and have led in the past to a number of catastrophic turbine and generator rotor failures in fossil-fueled power plants. Modern forging practices minimize such inclusions and present-day nondestructive examination and evaluation techniques provide much greater assurance of the soundness of turbine-generator rotors.

The NRC staff should follow this issue closely since the use of integral LP turbine rotors, particularly if they are not bored, will require the development of an entirely new set of preoperational and periodic operational inspection, evaluation, and acceptance requirements to protect against turbine missiles. (The staff should also consider this issue for LP turbine rotor replacement programs for currently operating plants.)

7. Cavity-Floor Area Beneath Reactor Vessel

The layout of the containment for the proposed ABWR design makes use of a cavity floor area beneath the reactor vessel to deal with core/concrete interaction. This area is based on an EPRI requirement of $0.02m^2$ per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design

but also to the design of all Advanced Light Water Reactor designs.

Sincerely,

David A. Ward
Chairman

<u>References</u>:

1. Letter dated August 17, 1989 from Charles L. Miller, Office of Nuclear Reactor Regulation, NRC, to Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, dated August 1989.
2. Letter dated August 7, 1987 from Thomas E. Murley, Office of Nuclear Reactor Regulation, NRC, to Ricardo Artigas, General Electric Company, enclosing GE Advanced Boiling Water Reactor, Licensing Review Bases, dated August 1987.
3. GE Nuclear Energy, Standard Safety Analysis Report, Advanced Boiling Water Reactor, Chapters 1 through 20.

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

November 24, 1989

Mr. James M. Taylor
Acting Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: MODULE 1 OF THE DRAFT SAFETY EVALUATION REPORT FOR THE ADVANCED BOILING WATER REACTOR DESIGN

During the 355th meeting of the Advisory Committee on Reactor Safeguards, November 16-18, 1989, we met with representatives of the Office of Nuclear Reactor Regulation (NRR) and the General Electric Company (GE) to discuss Module 1 of the staff's Draft Safety Evaluation Report (DSER) for the Advanced Boiling Water Reactor (ABWR) design. This matter was also considered by our ABWR subcommittee during several meetings, the latest on October 31, 1989. We also had the benefit of the documents referenced.

The staff's DSER relates to the GE application for final design approval (FDA) and design certification of the ABWR design. The DSER is scheduled for completion in four modules. Module 1 is the subject of this letter and addresses Chapters 4, 5, 6, and 17 of the ABWR Standard Safety Analysis Report (SSAR) and corresponding chapters of the Standard Review Plan (SRP), NUREG-0800. Our review of these chapters of the SSAR has been completed through Amendment 7.

A number of the SSAR and DSER sections included in the Module 1 chapters are presently missing and will be issued as SSAR revisions and supplements to the DSER. Even within the included sections, there are a number of open, unresolved, and confirmatory issues and incomplete interface requirements or other information that will delay completion of our review until the revisions and supplements are issued. Comments on such missing or incomplete information will be included with our review of future modules.

Our comments should not be considered complete until we have prepared a report to the Commission concerning the final integrated DSER, which is presently scheduled for late 1990. For now, we are providing the following comments and recommendations concerning Module 1.

GENERAL

1. The staff's ABWR licensing review bases letter to GE (Reference 2) states, "The degree of design detail necessary for providing an essentially complete design is to be that detail that is suitable for obtaining specific equipment or construction bids and to demonstrate

conformance to the design safety limits and criteria." We believe that the level of design detail in Module 1 falls short of this requirement. For example, we find that while GE has committed to follow applicable codes, standards, and regulatory guides, they have developed internal specifications for materials used in the fabrication of pressure boundary components that have not been submitted for NRC review. We also find that a number of design details (such as those relating to design temperature and pressure and pipe size) are indicated on drawings in the SSAR as "to be established by others" or similar statements. Unless such information is included in the SSAR or other documents that are reviewed by the staff, it is clear that the level of design detail is inadequate. We recommend that the staff revisit the issue of what constitutes an "essentially complete" design. The staff should also consider the question of form and depth of reporting differences between the ABWR being designed for construction in Japan and the ABWR design being proposed for certification.

2. The SSAR chapters contain a number of sections for which there are no corresponding sections in the DSER or SRP, or the subjects of the DSER or SRP sections are different. Also, there are cases wherein the SRP contains sections that do not appear in the SSAR or DSER. We recommend that the DSER sections be referenced by number and title to the corresponding SSAR sections they evaluate. Differences, including the absence of any corresponding SRP sections, should be identified in the DSER.

CHAPTER 4 - REACTOR

3. The fine motion control rod drive system (FMCRDS) materials list discussed in SSAR Section 4.5.1.1 shows Stellite guide rollers and roller pins. Section 5.2.3.2.2.2 states that cobalt base alloys used for pins and rollers in the FMCRDS have been replaced with noncobalt alloys. The list of materials should be corrected.

4. We were told by GE that the design of the integral rod ejection support system for the FMCRDS has been changed from that described in SSAR Section 4.6.1. The staff should determine that their evaluation in the DSER is based on the revised design and the SSAR should be corrected.

CHAPTER 5 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5. The SSAR states that the automatic depressurization system (ADS) utilizes safety relief valves (SRVs) each of which is equipped with an air accumulator and check valve arrangement designed to ensure two actuations following failure of the air supply. Although not stated in the SSAR, GE indicated that the accumulators are backed up by the nitrogen supply system. This backup arrangement needs to be described in the SSAR together with how check valve operability will be ensured.

6. The specifications given in the SSAR for the materials of the primary pressure boundary do not meet current "good practice," or the practice GE says they would require in the construction of an ABWR--they should. To clarify this issue, the SSAR should contain answers to the following questions: (1) will the steel in the core beltline be forged rings or welded plate?; (2) will upper limits on sulfur content of the rolled plate in the pressure vessel be those given in the ASME Code SA-533, Specifications for Pressure Vessel Materials (0.04%) or lower values consistent with good modern practice (under 0.015% with shape control)?--an adequate level is specified for forged segments (ASME Code SA-508, Class 3, Specification for Quenched and Tempered Vacuum-Treated Forgings) and is available as an option in SA-533 but not called out by GE; and (3) what will be the upper limit on delta ferrite for cast stainless steel components? The Code's allowed value of 25% should be halved to substantially remove concern about long-term aging.

7. SSAR Section 5.3.3 states that design for vessel annealing is not required because the predicted value of adjusted RT_{NDT} does not exceed 200° F. The DSER states that the integrity of the reactor vessel is ensured because the vessel may be annealed, if necessary. GE stated during our meeting that the vessel is not designed to be annealed. The DSER statement should be resolved with GE.

8. We believe that potential safety hazards (e.g., excessive internal pressure) associated with an uncleared electrical fault inside a reactor internal pump (RIP) should be analyzed and documented in the SSAR.

9. We were told by GE that motor restraint rods are provided to prevent ejection of an RIP. We believe that this important feature should be described in the SSAR and evaluated by the staff.

10. SSAR Section 5.4.6 states that the design basis for the Reactor Core Isolation Cooling (RCIC) system is only 30-minutes of operation during a loss-of-ac power event. We believe that a more complete discussion of the station blackout capability should be included in the SSAR. The DSER should include an evaluation of the 30-minute capability as an acceptable design basis.

11. The DSER contains no specific references to SSAR Sections 5.4.4-5, 5.4.9, and 5.4.12-14. These sections discuss feedwater piping, main steam line flow restrictors, isolation systems and piping, component supports, and valves. There are no comparably numbered sections in the SRP. It is not clear where the staff intends to report its evaluation of these important topics.

<u>CHAPTER 6 - ENGINEERED SAFETY FEATURES</u>

12. The design basis for the ECCS and the conclusions given about its performance do not include the ejection of an RIP (450 cm^2 break).

The rationale for excluding such an event as a design basis break should be discussed in the SSAR.

13. DSER Section 6.2.6 indicates that inflatable seals will be used for primary containment equipment and personnel air lock penetrations. We believe that an appropriate description of the seals and the air supply arrangement and reliability should appear in the SSAR. The discussion should include the capability of the seals to function under elevated pressure and temperature conditions for prolonged periods of time following a design basis accident.

14. There is a new section 6.5.5 (Pressure Suppression Pools as Fission Product Clean-Up Systems) in the SRP which does not appear in the SSAR or DSER. Why is this SRP section not being used for the ABWR?

CHAPTER 17 - QUALITY ASSURANCE

15. Chapter 17 of the SSAR is intended to describe how GE and its major technical associates (not mentioned by name in the SSAR but we assume to be Toshiba Corporation and Hitachi Limited) engage in the joint development and engineering of the ABWR design. The quality assurance programs used by the technical associates are not described or referenced in the SSAR. We believe they should be.

In conclusion, we believe that significant progress has been made by the staff in its review of the SSAR for the Advanced Boiling Water Reactor. A considerable amount of work remains to be completed before the FDA is issued as expected by the end of 1990. We will continue to review this work as the documentation becomes available.

Sincerely,

Forrest J. Remick
Chairman

References:

1. Letter dated August 17, 1989 from Charles L. Miller, Office of Nuclear Reactor Regulation, NRC, to Mr. Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, dated August 1989
2. Letter dated August 7, 1987, from Thomas E. Murley, Office of Nuclear Reactor Regulation, NRC, to Ricardo Artigas, General Electric Company, enclosing GE Advanced Boiling Water Reactor, Licensing Review Bases, dated August 1987
3. GE Nuclear Energy, Standard Safety Analysis Report, Advanced Boiling Water Reactor, Chapters 4, 5, 6, and 17

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

December 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: WESTINGHOUSE'S APPLICATION FOR PRELIMINARY DESIGN APPROVAL FOR THE RESAR SP/90 DESIGN

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we completed our review of Westinghouse's application for Preliminary Design Approval (PDA) for the Westinghouse Reference Safety Analysis Report (RESAR SP/90) nuclear power block (NPB). We heard presentations from the NRC staff and the applicant concerning the staff's draft Safety Evaluation Report (SER) (NUREG-1413) for this PDA during our meeting. Representatives of the staff and of the Office of the General Counsel (OGC) discussed the related draft PDA document. Our Subcommittee on the Advanced Pressurized Water Reactors has held a series of meetings with the staff and representatives of the applicant regarding this matter over the past two and a half years. We also had the benefit of the documents referenced.

1.0 Scope and History of RESAR SP/90 Application

The RESAR SP/90 is an evolutionary (as contrasted with passive) Advanced Light-Water Reactor (ALWR) design for a single-unit NPB, rated at a reactor power of 3800 MWt. Although many basic design decisions were made by Westinghouse prior to completion of the EPRI ALWR Utility Requirements Document, the design of this four-loop pressurized water reactor generally conforms to the EPRI requirements for such designs.

RESAR SP/90 NPB contains preliminary design information for the portion of the design that encompasses NPB buildings, structures, systems, and components. Specifically excluded from the scope are the turbine building, the waste disposal building, the service building, the administration building, the service water/cooling water structure, and the ultimate heat sink. These features will be the design responsibility of an applicant proposing to build a facility referencing the RESAR SP/90 design. Interface information addressing the pertinent safety-related design requirements necessary to ensure the compatibility of the referenced system with

the plant-specific portion of the facility has been included in the RESAR SP/90 application.

On October 24, 1983, Westinghouse submitted an application for a PDA for RESAR SP/90 NPB design in accordance with 10 CFR Part 50, Appendix O, "Standardization of Design: Staff Review of Standard Designs," which was the then existing regulatory basis for this type of application. The application was docketed on November 30, 1983 (Docket No. 50-601). The RESAR SP/90 application describing the design of the NPB was submitted in modular form during the period from October 23, 1983 to March 9, 1987. In addition, the information in RESAR SP/90 has been supplemented by 47 amendments to these modules.

2.0 Regulatory Background

Before the promulgation of 10 CFR Part 52 in May of 1989, the review of RESAR SP/90 had been performed by the staff pursuant to Appendix O to 10 CFR Part 50, using a procedure similar to that used for custom plant reviews for which guidance to staff reviewers is provided in the Standard Review Plan. This evaluation was analogous to a construction permit (CP) licensing review for a specific facility and conducted with the intent that, following satisfactory completion of the reviews performed by the staff and the ACRS, a PDA could be issued by the staff. The promulgation of 10 CFR Part 52 resulted in the transfer of Appendix O to 10 CFR Part 52; hence a PDA can now be issued for this application pursuant to 10 CFR Part 52. A PDA is optional for a Final Design Approval (FDA) and/or Design Certification under the provisions of 10 CFR Part 52.

3.0 The Staff's SER and the PDA

The SER and PDA represent the first stage of the staff's review of the design, construction, and operation of the RESAR SP/90 design. During our meetings, we learned that there is no prospective CP applicant nor does Westinghouse intend to apply for an FDA and/or Design Certification of the RESAR SP/90 design until there is a proven interest on the part of a domestic or foreign utility. The staff's SER summarizes the results of the staff's radiological safety review of the RESAR SP/90 NPB design and delineates the scope of the technical details considered in evaluating the proposed design. This review took place over the period of October 1983 to October 1989 (the date on which the staff decided to close its review). Environmental aspects were not considered in the staff review of RESAR SP/90, but would be addressed in a utility's plant-specific application.

3.1 Comments on the Staff's SER

There are 170 open items that will require resolution during the review of a plant-specific application for an Operating License (OL). Most of these appear to be the kind of open issues expected at this stage of the design. Of the 170 open items, 17 are site specific, 110 involve information in the scope of an OL or FDA and/or Design Certification application, and 43 had not been resolved by the staff when it closed its review in October 1989. (Westinghouse submittals on many of these 43 open items, including its proposed resolution of Generic Safety Issues, Unresolved Safety Issues, post-TMI regulatory requirements, and outstanding PRA issues are yet to be reviewed by the staff.) In view of these open items and our concerns regarding the SER and the many unresolved severe accident issues, we indicated to the staff that its conclusions on page 25-1 of the draft SER were stated too strongly. The staff agreed to revise this language.

The Committee is not of one mind regarding the issuance of a PDA for the RESAR SP/90. On the one hand, there is merit to the argument that Westinghouse's application for the RESAR SP/90 PDA was made in good faith in 1983 under a different set of regulations and that it is now appropriate to document the reviews that have taken place to date and issue the PDA for potential future use as a reference design for an individual plant CP application or as the starting point for an FDA and/or Design Certification application. Both Westinghouse and the staff advocate this approach; neither believes that it can devote further resources to this effort.

On the other hand, we view the RESAR SP/90 SER as a mixed bag of staff evaluations that were performed over the seven-year period since the application was filed. Some are current and well done; others are poorly done and/or were performed years ago and do not meet the standards that we believe should be applied to a current SER. A major contributor to this problem appears to be the staff's reliance on the July 1981 Standard Review Plan (SRP) (NUREG-0800) in performing this review. This SRP needs updating to reflect the current situation for the licensing of ALWRs.

Some examples of our concerns with the staff's SER are:

3.1.1 SER Chapter 7, Instrumentation and Controls, references a staff review that was performed in 1979 for the Westinghouse RESAR 414 design. The staff concluded that the computer based integrated reactor protection system design for RESAR SP/90 is acceptable for a PDA on the basis of the "similarity" of the RESAR 414 design to that proposed for RESAR SP/90. It is our view that the staff should have developed improved standards for the review of such systems during this 11-year period. We are

particularly concerned about the verification and validation of the software employed with computer based reactor protection systems. It appears that there is a need to augment existing staff resources with expertise in the computer science area so that appropriate standards can be developed for the review of computer based reactor protection systems. All of the proposed evolutionary and passive ALWRs employ such systems.

3.1.2 For materials used in the fabrication of pressure boundary components, Westinghouse has committed to follow applicable codes, standards, and regulatory guides. Many of these are not representative of current industry practice for such materials. We learned that Westinghouse has developed internal specifications for pressure boundary materials that presumably do reflect current industry practice. These were not submitted for the staff's review.

3.1.3 The proposed design employs water displacer control rods and associated control rod drive mechanisms, which is a new feature for Westinghouse plants. The SER describes the function of and strategy for use of these control rods. The SER, however, does not discuss the pressure boundary integrity of these new control rod drive mechanisms or the potential for reactivity insertion accidents that could result from misoperation of these control rods. Although Westinghouse submitted information on these subjects, the staff has not completed its review of this information. In general, we believe that new features of this kind should be thoroughly reviewed at an early stage of review.

3.1.4 Our review, which represents only a sampling effort, revealed a number of factual errors and inconsistencies in the SER; the staff has agreed to correct these errors. We believe that a review of the draft SER by Westinghouse, which has not yet had access to this predecisional document, would reveal additional errors that should be corrected. We recommend that this be done.

3.2 Comments on the PDA Document

The PDA states that the preliminary design information contained in RESAR SP/90 "complies with the requirements of 10 CFR Part 52, Appendix O . . . and is acceptable for incorporation by reference in applications for individual construction permits . . . ." The PDA does not describe how this preliminary design information would be used in a future FDA and/or Design Certification application.

We were told by OGC that this results from the fact that Westinghouse has not made an application under 10 CFR Part 52.

Given the quality of the SER for this PDA, we are concerned with the language of the PDA that requires the staff and ACRS to utilize and rely on the "approved preliminary design" in their reviews of any individual facility construction permit application " . . . unless significant information which substantially affects the determination set forth in this PDA, or other good cause, is present." OGC advised us that this requirement would apply only to the staff and ACRS reviews of a CP application and that both entities would be able to revisit any issue in their review of any type of application that would lead to an OL. This is satisfactory to us but could present problems for the staff in dealing with a contested CP application.

4.0 Comments on the SP/90 Design

We have two concerns regarding SP/90 design features:

4.1 Our review of the NPB layout indicates that Westinghouse has provided many desirable features from the standpoint of separation of equipment trains for protection against fires and industrial sabotage. However, we are concerned about the location of the emergency diesel generators (EDGs) on the same floor and corridor from the control room. We believe that another location for the EDG room should be specified in view of the potential for fire and/or explosions associated with the operation of large diesel generators.

4.2 The proposed RESAR SP/90 design employs a spherical containment. To deal with core/concrete interaction, the layout of the containment employs a cavity floor area beneath the reactor vessel that is based on the EPRI requirement of 0.02 m^2 per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all of the ALWRs.

5.0 ACRS Recommendations on the Issuance of a PDA

We believe, subject to the above comments, that the proposed design of the RESAR SP/90 NPB can be successfully completed and used in an application for an individual plant CP. Accordingly, we recommend that a PDA be issued for the proposed Westinghouse RESAR SP/90 NPB.

6.0 Concluding Remarks

Finally, we wish to commend the Westinghouse Electric Corporation, the Japanese APWR program participants, the EPRI ALWR Utility Steering Committee, and the EPRI staff for the effort they have expended in the development of this evolutionary design. The RESAR SP/90 design represents an important step forward in providing improved LWR designs that incorporate many of the lessons related to safety, performance, and reliability that have been learned by the nuclear power industry over the past 30 years.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft NUREG-1413, "Safety Evaluation Report Related to the Preliminary Design of the Standard Nuclear Steam Supply Reference System, RESAR SP/90" (Predecisional)
2. Draft Westinghouse Electric Corporation, Docket No. 50-601, Reference Safety Analysis Report (RESAR SP/90 Nuclear Power Block Standard Design), Preliminary Design Approval (PDA) (Predecisional) (Discussed during the November 8-10, 1990 ACRS full Committee meeting)
3. Letter NS-EPR-2675 dated November 1, 1982 from E. P. Rahe Jr., Westinghouse Electric Corporation, to F. Miraglia, U.S. Nuclear Regulatory Commission, Subject: Westinghouse Advanced Pressurized Water Reactor Licensing Control Document

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

November 14, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: SECY-90-353, LICENSING REVIEW BASIS DOCUMENT FOR THE COMBUSTION ENGINEERING, INC. SYSTEM 80+ EVOLUTIONARY LIGHT WATER REACTOR

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we reviewed the staff's SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990. Our Subcommittee on Advanced Pressurized Water Reactors also considered this matter during a subcommittee meeting on November 1, 1990. During this review, we had the benefit of discussions with representatives of the NRC staff and of Asea Brown Boveri Combustion Engineering. We also had the benefit of the documents referenced.

The staff has recommended that the Licensing Review Basis (LRB) effort for the Combustion Engineering (CE) System 80+ design, which is well advanced, be continued to completion. There does not appear to be any substantive disagreement between the staff and CE on issues addressed in the LRB document.

The only approved LRB document was proposed by the General Electric Company (GE) as a way of obtaining early agreement with the staff on major process and technical issues for the review of its advanced boiling water reactor design certification application. It was approved by the Director of NRR in a letter to Mr. R. Artigas, GE, on August 7, 1987. This letter contains the qualification that the LRB represented the approach in "certain key areas" that GE was committed to follow ". . . until final Commission positions and staff requirements are defined and implemented." At that time, neither 10 CFR Part 52 nor Commission-approved staff positions relating to the certification of advanced light water reactors such as SECY-90-016 (referenced) were available. We note that 10 CFR Part 52 does not discuss the use of LRB documents as a part of the final design approval or certification process. These regulatory requirements and others under development have preempted the need for and diminished the usefulness of an LRB document for the CE System 80+ design. We recommend that no further effort be devoted to the proposed LRB document for the CE System 80+ design.

Additional comments by ACRS members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr., are presented below.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

Additional Comments by ACRS Members Ivan Catton, Paul G. Shewmon, and J. Ernest Wilkins, Jr.

We understand that this LRB document can be completed and issued with relatively little additional effort. If so, we would prefer to see an orderly disposition of this LRB document in accordance with the staff recommendation in SECY-90-362 (referenced). We would agree with our colleagues that the CE System 80+ LRB effort be terminated now if the Commission, the staff, and the ACRS need to invest any significant additional effort.

References:

1. SECY-90-353, "Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor," dated October 12, 1990.
2. SECY-90-362, "Staff Comments on the Continuing Need for a License Review Basis Document for Each Passive Design," dated October 24, 1990.
3. SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements," dated January 12, 1990.
4. Letter LD-90-005 dated January 22, 1990 from A. E. Scherer, Combustion Engineering, to R. Singh, Subject: System 80+ Licensing Review Basis Document.
5. Letter LD-90-060 dated August 28, 1990, from E. H. Kennedy, Combustion Engineering, to Thomas V. Wambach, NRC, Subject: Licensing Review Basis for the System 80+ Standard Design.

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

August 18, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: ELECTRIC POWER RESEARCH INSTITUTE ADVANCED LIGHT WATER REACTOR UTILITY REQUIREMENTS DOCUMENT -- VOLUME II, EVOLUTIONARY PLANTS

During the 387th and 388th meetings of the Advisory Committee on Reactor Safeguards, July 9-11 and August 6-8, 1992, we reviewed the NRC staff's Final Safety Evaluation Report (FSER) for Volume II of the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) for Evolutionary Plants. Our Subcommittee on Improved Light Water Reactors held meetings on June 17-18 and July 27, 1992, to review this subject. During these meetings, we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the documents referenced.

In the early 1980s, EPRI established the ALWR program to support the United States utility industry efforts to ensure a viable nuclear power generation option for the 1990s and beyond. The objective of the program was to ensure that future nuclear power plants would be safer, simpler, more robust with greater margins, more easily operated and maintained, and more certain of being constructed and licensed without delays. This was accomplished using utility experience, by establishing design philosophy, producing design criteria and guidance to achieve the objective, and addressing the policies and regulations of the NRC.

The EPRI ALWR URD is a compendium of technical requirements for design, construction, and performance of ALWR nuclear power plants for the 1990s and beyond. The URD consists of three volumes:

- Volume I, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, including the design objectives and philosophy, the overall physical configuration and features of a future nuclear plant design, and the steps necessary to take the proposed ALWR design criteria beyond the conceptual design state to a completed, functioning power plant.

- Volume II, "ALWR Evolutionary Plant," consists of 13 chapters and contains utility design requirements for an evolutionary nuclear power plant (approximately 1350 Mwe).

- Volume III, "ALWR Passive Plant," contains utility design requirements for passive nuclear power plants (approximately 600 Mwe).

We have followed the development of the EPRI ALWR program from its inception and offered suggestions regarding safety improvements on several occasions. We also held numerous subcommittee and Committee meetings to consider and discuss the development of the EPRI URD program and the NRC staff's reviews.

The staff's review of the URD was conducted as described in NUREG-1197. As noted therein, the staff used NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," for review guidance. In addition, the staff's review reflects the requirements of 10 CFR 52, the Commission's policy statements on severe accidents, and the safety goals.

Although the SRP was used by the staff as guidance, the level of detail in the EPRI submittal did not permit a review of its completeness. (The SRP was written to support the review of safety analysis reports on specific plant designs for which a significant amount of design and construction information was available.) The staff conducted its review with the understanding that EPRI design criteria would meet all current regulations, except where deviations were identified. The staff's review of the URD focused primarily on determining whether the EPRI criteria did or did not conflict with current regulatory requirements.

In its review of Volume II of the URD, the staff identified a number of issues that will require additional information before the staff can reach a final conclusion. Initially, the staff divided the outstanding issues into three categories: (1) open policy issues on which the staff has proposed a position, but for which the Commission has not yet provided guidance, (2) open issues that must be satisfactorily resolved before the staff can complete its review of the URD, or (3) confirmatory issues for which the staff will ensure follow up of commitments in the URD.

At this date the staff has identified 21 open policy issues that are included in a draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements" that was issued on February 27, 1992. We provided our recommendations on the open policy issues pertaining to evolutionary plants in our letters which addressed SECY-90-016, SECY-91-078, and the draft SECY paper of February 7, 1992.

The staff has handled the remaining 410 open issues which were identified in the FSER for Volume II by classifying them as "Vendor or Utility Specific Items" which must be satisfactorily addressed during the staff's review of a vendor- or utility-specific application. The staff plans to issue a supplement to the FSER after all evolutionary policy issues have reached final resolution. The staff indicated that they plan to interact with EPRI in an attempt to resolve significant open issues which may be resolved generically, and to include in a supplement any which are resolved.

We recommend generic resolution of as many of these issues as possible.

We commend EPRI for developing a comprehensive set of requirements. These will aid in the design of nuclear plants which will be safer, simpler, more robust, and more easily operated and maintained.

We commend the NRC staff for a very thorough review of the EPRI ALWR Evolutionary URD, and its work with EPRI to identify and resolve many issues relevant to licensing future LWRs. We recognize the NRC staff's position that its review necessarily is incomplete.

Sincerely,

David A. Ward
Chairman

<u>References</u>:

1. SECY-92-172, dated May 12, 1992, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document, including the following enclosures:
   - Draft Safety Evaluation Report for Volume I, "Program Summary of the NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992
   - Safety Evaluation Report for Volume II, "NRC Review of the Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document for Evolutionary Plant Designs," prepared by the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, dated May 1992

2. Advanced Light Water Reactor Utility Requirements Document, Volume II, "ALWR Evolutionary Plant," Chapters 1-13, through Revision 4, dated April 1992, Prepared for Electric Power Research Institute

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

July 17, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: INTEGRAL SYSTEM AND SEPARATE EFFECTS TESTING IN SUPPORT OF THE WESTINGHOUSE AP600 PLANT DESIGN CERTIFICATION

During the 387th meeting of the Advisory Committee on Reactor Safeguards, July 9-11, 1992, we discussed the programs of integral system and separate effects testing being planned by both Westinghouse and NRC to support the certification effort for the Westinghouse Electric Corporation's AP600 passive plant design. We held discussions on this matter during our 381st through 384th (January-April 1992) meetings, inclusive. Our Subcommittee on Thermal Hydraulic Phenomena held meetings on December 17, 1991, March 3, 1992, and June 23-24, 1992 to review this issue. During these meetings, we had the benefit of discussions with representatives of the Westinghouse Electric Corporation and the NRC staff. We also had benefit of the referenced documents. We have previously reported to you on this matter in our letters of March 10 and April 6, 1992.

BACKGROUND

Appropriately validated thermal hydraulic computer models must be relied on to support the safety assessments required for certification of the AP600. Westinghouse has indicated that it plans to use its more mechanistic assessment code, WCOBRA/TRAC, only for large-break LOCA analyses, and will rely on its evaluation model, NOTRUMP, for analyses of all other design-basis events. The NRC plans to use RELAP5/MOD3 to support its assessments.

The NOTRUMP code is an evaluation model code that is based on 10 CFR Part 50, Appendix K, requirements. The other two codes, WCOBRA/TRAC and RELAP5/MOD3, are more mechanistic codes that have been qualified as best-estimate tools only for large-break LOCAs. All of these analysis tools will be required to simulate the AP600 behavior in regimes where the codes are known to be weak. These regimes include phenomena such as horizontal (perhaps countercurrent stratified) flows, interface movements, thermal

stratification, rapid "shock" condensation, boron mixing, and low-pressure gravity-driven flows.

To develop the necessary data for improvement and validation of these models for AP600 assessment, Westinghouse now has plans for conducting a number of separate effects tests at several different facilities, and integral system tests. The integral system test programs are to be conducted in a low-pressure facility now nearing final design at the Oregon State University (OSU) and in an existing high-pressure facility, SPES (in Italy), to be modified to better simulate AP600.

The NRC has proposed to conduct high-pressure confirmatory testing by modifying and using the existing ROSA-IV facility at JAERI in Japan. The modified facility will be referred to as ROSA-V. The NRC has no specific plans for additional separate effects testing. The staff does plan to conduct low-pressure integral system testing in the OSU facility after the Westinghouse program has been completed.

At this time, we have the following comments and recommendations regarding various aspects of these planned and proposed efforts.

WESTINGHOUSE PROGRAM

We believe that, with certain enhancements, the Westinghouse program will be adequate for the certification process. We have the following specific comments and recommendations:

- We are concerned that Westinghouse plans to rely primarily on its NOTRUMP evaluation model (EM) code. It is a step backwards to use computer codes of only EM sophistication and capabilities to evaluate the thermal hydraulic behavior of new nuclear power plants.

- The Westinghouse separate effects tests of most importance to the certification of AP600 are the Core Make-up Tank (CMT) tests and the Automatic Depressurization System (ADS) tests. The test matrices for these do not cover ranges of conditions that are broad enough to yield an adequate data base for the required model development. We recommend that pressure disturbances of the types that would be caused by either ADS valve actuation or by rapid steam condensation when cold CMT fluid is injected into the downcomer region be part of the test program.

- An additional separate effects test facility is needed to investigate the asymmetric effects associated with the downcomer and with the cold-side plenum of the steam generator.

- SPES is generally a good choice for conducting full-height, full-pressure integral system tests. However, in addition to the scaling problems associated with a high ratio of surface area to fluid volume that plague small-scale simulations of this kind (and must be dealt with), the proposed modified version, SPES-II, has two important scaling defects that should be eliminated: (a) the aspect ratio (height to diameter) of the simulated pressurizer is different from that of the AP600 and (b) the cold leg configuration is not geometrically similar to that of AP600.

  We recommend that Westinghouse be required to preserve the scaling of the pressurizer and the geometrical configuration of the cold legs, to better simulate AP600 behavior (this would include simulation of a reactor coolant pump in each leg).

- The method proposed for simulating steam generator tube ruptures in SPES-II is flawed in that it does not appear to allow the break flow from the primary system to be from both the hot and cold sides of the tube. We recommend that Westinghouse develop a better simulation method.

- The OSU low-pressure integral system testing facility is well conceived. We commend Westinghouse for its efforts with respect to this facility. Our evaluation of the scaling rationale for the facility design (discussed during the subcommittee meeting of June 23-24, 1992) is that it is soundly based. Further, the 400 psia design capability should allow considerable simulation of high-pressure effects, while providing the more important low-pressure behavior.

NRC PROGRAM

Our understanding of the justification provided by the NRC staff for its proposed confirmatory high-pressure integral system testing in the ROSA-V facility is as follows:

- Because ROSA-V is considerably larger than SPES-II, such confirmatory testing would provide an additional check on the adequacy of the scaling capabilities of the codes, and would help confirm that important effects have not been overlooked.

- The confirmatory test program would provide the opportunity to maintain the staff's thermal hydraulic expertise and up-to-date knowledge in this field.

While we agree that the above considerations have some merit, we have not been persuaded that confirmatory high-pressure testing by the staff is needed before the AP600 design certification and, even if this were the case, we have significant reservations about the

adequacy of the ROSA-V facility for this purpose. These positions are based on the following observations:

- The NRC staff has not presented convincing arguments supporting its needs for confirmatory testing, particularly at high pressures.

- The SPES-II facility appears to be sufficient to meet all the high-pressure integral system testing needs. The NRC will be able to use the SPES-II facility for its confirmatory testing needs just as it plans to use the OSU facility.

- The desired staff experience will come from pre-test and post-test evaluations of the various tests using the RELAP5/MOD3 code. This experience can just as easily be obtained by evaluating the SPES-II and OSU tests and results.

- The ROSA-V facility contains several atypicalities that will manifest themselves in difficult-to-explain behavior relative to that expected for AP600 (the sensitivity of the ROSA-V thermal hydraulic behavior is well documented in the INEL report, NUREG/CR-5853).

- The tests would be in a distant location. There would be a very limited number of tests, because of the expense involved. In addition, we are concerned that the adequacy of instrumentation (for example) might have to be compromised in order to reduce overall program costs.

For the above reasons, we believe that NRC resources would be better used by focusing on three areas: (a) possible additional separate effects testing to support the modeling needs for RELAP5/MOD3, (b) participation in the pre-test and post-test analyses efforts associated with the SPES-II and the OSU test programs, and (c) consideration of utilizing the SPES-II facility for high-pressure confirmatory testing needs in the same way the staff plans to use the OSU facility for its confirmatory low-pressure testing needs.

To accomplish the above objectives, we believe that the staff should consider the establishment of a task force of experts in related fields to assist it in the development of the analytical and experimental programs necessary for timely certification of the AP600 passive plant design.

Sincerely,

Paul Shewmon

Paul Shewmon
Acting Chairman

References:

1. U.S. Nuclear Regulatory Commission, NUREG/CR-5853, "Investigation of the Applicability and Limitations of the ROSA-IV Large Scale Test Facility for AP600 Safety Assessment (Draft)," dated May 1992
2. T. Boucher, Idaho National Engineering Laboratory, et al., "Scaling Issues for a Thermal-Hydraulic Integral Test Facility," Paper transmitted via a memorandum from L. Shotkin, NRC-RES, for P. Boehnert, ACRS, dated June 29, 1992
3. Oregon State University Report, OSU-NE-9204 (Draft), "Scaling Analysis for the OSU AP600 Integral System and Long Term Cooling Test Facility," J. Reyes, Jr., dated June 1992 (**W Proprietary Report**)
4. Letter dated January 22, 1992, from G. Saporano, ENEA, Italy, to E. S. Beckjord, NRC, transmitting documentation on SPES test facility
5. Memorandum dated June 13, 1991 from S. Modro, INEL, for L. Shotkin, NRC-RES, transmitting draft report, "Evaluation of Scaled Integral Test Facility Concepts for the AP600" by Modro, et al.
6. U.S. Nuclear Regulatory Commission, SECY-92-219, "NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design," dated June 16, 1992 (**Predecisional**)
7. U.S. Nuclear Regulatory Commission, SECY-92-219A, "Addendum to SECY-92-219 - Providing Additional Information to Justify Sole Source Procurement," dated July 9, 1992 (**Predecisional**)
8. Memorandum dated April 21, 1992, from S. Chilk, Secretary, for J. M. Taylor, EDO, and W. Parler, General Counsel, Subject: SECY-92-037 - Need for NRC-Sponsored Confirmatory Integral System Testing of the Westinghouse AP600 Design
9. Westinghouse Topical Report, WCAP-13277, "Scaling, Design and Verification of the SPES-2, the Italian Experimental Facility Simulator of the AP600 Plant," dated April 1992 (**W Proprietary Report**)

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

June 10, 1992

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: TESTING AND ANALYSIS PROGRAMS IN SUPPORT OF THE SIMPLIFIED BOILING WATER REACTOR DESIGN CERTIFICATION

During the 385th and 386th meetings of the Advisory Committee on Reactor Safeguards, May 6-9 and June 4-5, 1992, we reviewed the testing and analysis programs in progress and proposed by GE Nuclear Energy (GE) in support of the certification effort for the Simplified Boiling Water Reactor (SBWR) passive plant design. Our Subcommittee on Thermal Hydraulic Phenomena held meetings to discuss this topic on April 23 and June 2, 1992. During these meetings, we had the benefit of discussions with representatives of GE and the NRC staff. We also had the benefit of the documents referenced.

GE will use its best-estimate code, TRACG, to evaluate the SBWR thermal hydraulic behavior under accident conditions ranging from ATWS with instabilities to long-term behavior of the Passive Containment Cooling System (PCCS). GE representatives presented a very good analysis of processes and phenomena important to accident scenarios postulated for the SBWR. The results were summarized in tables which are to be used by GE to validate the TRACG computer code. However, these same tables appear not to have been used to guide the design and operation of the experimental facilities that are to support the code validation process.

The GE experimental program consists of three elements:

1) Laboratory scale experiments to obtain fundamental heat transfer data,

2) Separate effects tests to obtain data for parts of the total system and full-scale components where necessary, and

3) Integral system tests to obtain system data.

Although we were shown some comparisons of TRACG predictions with data from GE's integral system tests (GIST and GIRAFFE facilities), the question of whether or not the facilities can scale the important phenomena was not addressed in either GE's presentation or in the documents supplied to the ACRS by GE. A rigorous scaling analysis is needed if integral system test data alone are to be used to demonstrate that a TRACG calculation is meaningful.

We have some comments about the elements of the GE test plan. The initial conditions for the integral system tests are based on conditions assumed to exist some time after vessel depressurization. These conditions include an initial drywell and PCCS nitrogen mass fraction of 15 percent. The nitrogen concentration could be much higher. GE should develop a basis for its choices of initial conditions or broaden its test matrix to include some tests at much higher values of the nitrogen concentration, both in the drywell and in the PCCS.

Separate effects tests to be conducted in the PANTHERS facility will yield the data needed to characterize heat exchanger behavior under a variety of expected conditions. In particular, GE has agreed to add instrumentation to the individual heat exchanger tubes to obtain local heat transfer data. This will make the GIRAFFE integral system experiments more meaningful. We believe GE has been very responsive to issues raised by both the ACRS and the NRC staff in this regard.

The oscillatory behavior observed in the GIRAFFE integral system tests needs more detailed study to ensure that the suppression pool does not overheat due to steam bypass of the PCCS through the suppression pool top horizontal vents. The steam flow rate will be low which could lead to a stratified condition. The suppression pool is not a very effective heat sink when this process occurs. This may well require a separate effects study to obtain data for development of a low steam flow model for the horizontal vent. Further, review of the GIRAFFE facility instrumentation is needed to ensure that the resulting data will support TRACG model validation.

The SBWR has full pressure isolation condensers (IC) capable of removing 4.5 percent of full power decay heat at full system pressure. The behavior of isolation condensers is well understood and introduces no new processes. GE has indicated that it will collect relevant IC operating data for staff review. The SBWR is automatically depressurized when the vessel water level drops to some prescribed value by a staged opening of squib-type valves. Further, GE has had a great deal of experience with automatic depressurization and only the squib-type valve itself is of a new design. As a result, we do not believe that full-height, full-

pressure integral system testing is required for certification of the SBWR design.

The GE program includes conduct of integral system testing at the PANDA facility located in Switzerland. The NRC staff would like GE to obtain data from this facility in time to support its design certification review of the SBWR. To do so, GE would have to accelerate its schedule by six months. We agree with the NRC staff that further integral system testing of the PCCS is needed prior to the final design approval. It has not been demonstrated by GE that existing data obtained from GIRAFFE or GIST testing are sufficient for validation of the TRACG code, nor that the PANDA test facility will yield the needed data. A more definitive assessment by GE is needed; this assessment should include both the scaling rationale for the GIRAFFE, GIST, and PANDA facilities, and a demonstration of how the effects of test facility scaling distortion impact the important processes and phenomena outlined by GE in its evaluation of TRACG. As a part of such an effort, it may be possible to show that one can obtain the needed data by some combination of additional separate effects tests and judicious use of the GIRAFFE and GIST facilities.

To summarize, we agree with the NRC staff views that full-height, full-pressure integral system testing is not needed to support the SBWR design certification. Further, we agree that early integral system testing of the PCCS is essential to meet the present design certification schedule. We have not, however, seen evidence that the PANDA facility is adequate to obtain the needed data.

Sincerely,

David A. Ward
Chairman

References:

1. Memorandum dated February 26, 1992, for the Commissioners from James M. Taylor, Executive Director for Operations, transmitting Advance Copy of proposed Commission paper, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling Water Reactor (SBWR)"

2. Letter dated February 3, 1992, from R. C. Mitchell, GE Nuclear Energy, to U.S. Nuclear Regulatory Commission, Subject: GE Response to Request for Information on SBWR Testing Program

3. Joint Study Report, "Feature Technology of Simplified BWR (Phase I) GIRAFFE (Final Report)," dated November 1990, The Japan Atomic Power Company, et al. **(GE Proprietary Information)**

4. GE Nuclear Energy, GEFR-00850, "Simplified Boiling Water Reactor (SBWR) Program Gravity-Driven Cooling System (GDCS) Integrated Systems Test - Final Report," A.F. Billig, dated October 1989 **(Applied Technology Restriction)**

5. "ALPHA – The Long Term Passive Decay Heat Removal and Aerosol Retention Program at the Paul Scherrer Institute, Switzerland," by P. Coddington, et al., Paul Scherrer Institute, undated

6. Paper from the Proceedings of The International Conference on Multiphase Flows '91 – Tsukuba, Japan, September 24-27, "Condensation in a Natural Circulation Loop with Noncondensable Gases Part 1 – Heat Transfer," K. M. Vierow, GE Nuclear Energy, and V. Schrock, University of California

7. GE Draft Report: "Test Specification for IC & PCC Tests," undated **(GE Proprietary Information)**

8. Paper submitted to the Department of Energy, "The Effect of Noncondensable Gases on Steam Condensation Under Forced Convection Conditions," M. Siddique, Ph.D. Thesis - Massachusetts Institute of Technology, dated January 1992

ITEM 2: SELECTED ALWR POLICY ISSUES

The Committee has discussed the resolution of technical and policy issues at various meetings beginning in early 1990. The Committee has provided six reports to the Commission or the EDO on this matter. The Committee and the Improved Light Water Reactors Subcommittee have also discussed the regulatory treatment of non-safety systems (RTNSS) issues for the passive light water reactor designs. Subject to the availability of all necessary information from the staff and industry groups and satisfactory completion of the reviews, the Committee expects to continue providing reports to the Commission on these issues. The Committee will continue its review of additional issues as they are identified by the staff and/or the Committee.

The following documents are attached:

- ACRS report to the Commission dated April 26, 1993. Subject: SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." (PP. 62-65)

- ACRS letter to James M. Taylor (EDO) dated September 16, 1992. Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs" (PP. 66-69)

- ACRS letter to James M. Taylor (EDO) dated August 17, 1992. Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (PP. 70-74)

- ACRS letter to James M. Taylor (EDO) dated May 13, 1992. Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (PP. 75-78)

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

April 26, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: SECY-93-087, "POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS"

During the 396th meeting of the Advisory Committee on Reactor Safeguards, April 15-17, 1993, we discussed the NRC staff positions, delineated in SECY-93-087, on policy, technical, and licensing issues pertaining to evolutionary and advanced light-water reactor designs. During this meeting, we had the benefit of discussions with representatives of the NRC staff and of the documents referenced. We have discussed these issues during several of our previous meetings and provided comments and recommendations in the reports referenced.

We are in general agreement with the staff's positions in SECY-93-087; however, we have concerns regarding some issues and offer our comments and recommendations as follows. (The section titles and letter designations correspond to those in SECY-93-087.)

I. <u>SECY-90-016 ISSUES</u>

E. <u>Fire Protection</u>

In our April 26, 1990 report, we pointed out that redundant train separation is likely to be the most significant feature leading to reduced fire risk. We recommended that the proposed fire protection enhancements include separation of environmental control systems (i.e., separate heating, ventilating, and air conditioning (HVAC) systems for each train). The staff responded by conceding that separate HVAC arrangements may be needed, although other options may be available to the designer. The Commission endorsed the staff's response.

We remain concerned that a common normal ventilation system (such as that proposed for the ABWR) will be difficult to design to prevent the effluent from a postulated accident in one train of engineered safety features from reaching essential mitigating equipment in the other trains and

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creating conditions that exceed their environmental qualifications. Of particular concern is the capability of ventilation dampers to isolate the effects of high energy pipe ruptures in confined compartments served by the common HVAC system.

G. Hydrogen Control

The staff claims that it has sufficient basis for understanding hydrogen behavior to go forward with licensing criteria. It has not been demonstrated to us that this basis is as extensive, or applicable, as the staff believes. Further, the AP600 and ABB-CE System 80+ designs have containments that are more susceptible to significant damage from hydrogen detonation than most existing and evolutionary plants. This requires that the licensing criteria for this issue be reconsidered.

H. Core Debris Coolability

The staff has weakened the position taken in SECY-90-016 by not requiring that the core debris be adequately quenched. We believe that the present criterion for coolability, namely a cavity floor area greater than $0.02m^2/MWt$, is not soundly based. We recommend that the staff validate containment response to core-on-the-floor accident sequences by independent analyses using, for example, MELCOR, or CORCON and CONTAIN.

J. Containment Performance

We agree with the requirement that containment stresses not exceed ASME Code Service Level C for metal containments, but it is not clear how electrical penetrations through the containment should be considered. Such penetrations utilize nonmetallic electrical insulation as a portion of the containment boundary and need further consideration.

L. Equipment Survivability

We agree that passive plant design features provided only for severe accident mitigation need not be subject to the environmental qualification requirements of 10 CFR 50.45. We believe, however, that such mitigation features must be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the timespan for which they are needed.

II. OTHER EVOLUTIONARY AND PASSIVE DESIGN ISSUES

Q. Defense Against Common-Mode Failure in Digital Instrumentation and Control Systems

The staff's second recommendation is that the vendor or applicant analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR). We recommend that the scope of this assessment include consideration of common-mode failures during all events postulated in the SAR (e.g., fire, flood, pipe rupture, and extensive loss of essential power sources) and not be restricted to those events discussed in Chapter 15, "Accident Analysis."

T. Control Room Annunciator (Alarm) Reliability

The staff's basic recommendation is that the Commission approve the position that the alarm system for ALWRs meet the applicable EPRI requirements for redundancy, independence, and separation. These requirements do not include the use of Class 1E equipment and circuits. The staff also seeks approval of an additional position that goes beyond the EPRI requirements. This position is that "alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits." We believe that the staff needs to provide clarification and additional justification for this position.

Collectively, our identified issues represent a significant array of incompletely addressed concerns. We urge that they be addressed on a timely basis to ensure their early consideration by the design teams.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. SECY-93-087, dated April 2, 1993, for the Commissioners, from James M. Taylor, Executive Director for Operations, NRC, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactors (ALWR) Designs

2. Report from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Computers in Nuclear Power Plant Operations, March 18, 1993
3. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," September 16, 1992
4. Report from David A. Ward, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Digital Instrumentation and Control System Reliability, September 16, 1992
5. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, August 17, 1992
6. Report from David A. Ward, ACRS Chairman, to James M. Taylor, Executive Director for Operations, NRC, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements, May 13, 1992
7. Report from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactors Certification Issues and Their Relationship to Current Regulatory Requirements, April 26, 1990

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

September 16, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: DRAFT COMMISSION PAPER, "DESIGN CERTIFICATION AND LICENSING POLICY ISSUES PERTAINING TO PASSIVE AND EVOLUTIONARY ADVANCED LIGHT WATER REACTOR DESIGNS"

During the 389th meeting of the Advisory Committee on Reactor Safeguards, September 10-12, 1992, we reviewed the NRC staff's positions and recommendations concerning the certification issues for evolutionary and passive light water reactor designs contained in the draft Commission paper, which was forwarded to the Commission on June 25, 1992. Our Subcommittee on Improved Light Water Reactors met on September 9, 1992, to review this subject. During these meetings we had the benefit of discussions with representatives of the NRC staff and EPRI. We also had the benefit of the document referenced. We previously provided comments to you on other policy issues related to design certification in our letters of May 13, 1992 and August 17, 1992.

Our comments and recommendations on the proposed policy issues contained in the draft Commission paper are given below. Issues A, B, C, D, E, and G apply to evolutionary and passive plant designs and Issues F and H apply only to passive plant designs. The issue titles and letter designations correspond to those of the draft Commission paper.

A. <u>Defense Against Common-Mode Failures in Digital Instrumentation and Control (I&C) Systems</u>

It is our view that the thrust of the staff recommendations concerning defense against common-mode failures in digital I&C systems as underlined in Issue A of the draft Commission paper is appropriate. We agree with the staff that the applicant should be required to assess the defense in depth and diversity of the proposed designs for the events postulated in the Safety Analysis Report, and demonstrate an acceptable plant response for each. The staff proposes that the instruments, controls, and equipment required to demonstrate an acceptable response be independent of any common-mode failure mechanisms associated with the event. We view this requirement to be essential, but remain open as to the

best approach. The staff proposes an independent set of safety-grade displays and controls in the main control room. We believe that other arrangements might be shown to be acceptable.

In a separate letter to Chairman Selin dated September 16, 1992, we have provided additional comments and advice regarding the general approach being taken by the staff in its review of digital instrumentation and control systems.

B. Analyses of External Events Beyond the Design Basis

To assist in the closure of severe accident issues, the staff recommends that (1) analyses submitted in accordance with the requirements of 10 CFR 52.47 (concerning the contents of applications for standard design certification) include an assessment of internal and external events and (2) during the design certification review, the staff should evaluate those external events that are not site dependent (e.g., fires, internal floods) and certain bounding analyses. We agree with this staff recommendation.

C. Elimination of the Operating Basis Earthquake from Seismic Design

The staff is still reviewing this issue and has expressed only an interim position. We believe the staff is taking an appropriate approach in its interim position.

D. Multiple Steam Generator Tube Ruptures (MSGTRs)

The staff is recommending that the applicant for design certification perform additional analyses to determine the AP600 response to multiple breaks of up to 5 steam generator tubes. We agree with the staff's recommendation, but believe the staff should have a better technical basis for estimating the frequency of occurrence of such multi-tube breaks.

The staff is also recommending that the applicant for design certification of a passive or evolutionary PWR assess design features necessary to mitigate the amount of containment bypass leakage that could result from MSGTRs. We agree with the staff's recommendation.

E. Probabilistic Risk Assessment (PRA) Beyond Design Certification

The staff is recommending that, throughout the duration of the combined or operating license, the PRA be revised to address significant plant modifications, operating experience, and other developments that may affect previous PRA insights.

We are convinced that it is worthwhile for a plant operator to have an up-to-date PRA and are, therefore, reluctant to recommend against this position. However, if this is to be required, the staff should more clearly specify how it intends to use the updated PRA and what is meant by keeping it current. We think such guidance is part of the overall issue of appropriate use of PRAs in regulation and would be helpful to licensees and to the staff.

F. Role of the Operator in a Passive Plant Control Room

We agree with the first part of the staff's position "that sufficient man-in-the-loop testing and evaluation be performed ... to demonstrate that functions and tasks are integrated properly into the man/machine interface design" of passive ALWR control rooms.

The second part of the staff's underlined position states "that a fully functional integrated control room prototype is necessary for passive plant control room designs to demonstrate that functions and tasks are integrated properly into the man/machine interface design." We pointed out to the staff that the non-underlined last sentence of this paragraph is inconsistent with this language in that it would permit an applicant to "demonstrate that a control room prototype of reduced scope is sufficient." We also pointed out that the non-underlined paragraph preceding the underlined paragraph states that such a prototype "would likely" be required (not would be required) to demonstrate that functions and tasks are integrated properly into the man/machine interface design. We believe that the staff should clarify its intent by reconciling these various statements.

The staff believes that operators of passive plants will be confronted with a new operating philosophy. The staff argues that "the operators of passive plants must understand the operation of 'investment protection' systems and their interfaces with the safety-related passive systems" and that they will be confronted with "new functions and tasks unlike those required for evolutionary plants" (or current plants) "due to the new approach in operational philosophy" and "the increase in automation, and the greater use of advanced technology in the passive plant designs." As a result of our discussions with the staff and EPRI, we believe that the staff may be overreacting to the "newness" of these issues. It appears to us that additional discussion of this issue among the staff and EPRI and the vendors is needed.

G. Control Room Annunciator (Alarm) Reliability

We agree with the staff's position that the alarm system for ALWRs should meet the requirements of the EPRI Utility Requirements Document.

H. Regulatory Treatment of Nonsafety Systems

We were told that the staff is still engaged in significant ongoing discussions and review of this issue and that the associated position and recommendations are subject to modification. We believe the issue is substantial and has broad implications with respect to such items as use of PRAs in regulation, safety goal implementation, and reduction of regulatory burdens, and we expect to have additional future interactions with the staff and the industry. Consequently, we are not prepared to express a position on this issue at this time.

Sincerely,

David A. Ward
Chairman

Reference:

1. Draft Commission Paper dated June 25, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Review of the Draft Commission Paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs"

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

August 17, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 386th, 387th, and 388th meetings of the Advisory Committee on Reactor Safeguards, June 4-5, July 9-11, and August 6-8, 1992, we discussed with representatives of the NRC staff the staff's positions, recommendations, and resolution schedules concerning the certification issues for evolutionary and passive light water reactors contained in the draft SECY paper dated February 7, 1992. This supplements our letter of May 13, 1992, and provides our comments and recommendations on some of the staff's positions for the passive light water reactors. The section titles and letter designations correspond to those in the draft SECY paper.

I. <u>SECY-90-016 Issues</u> (For Passive Plants)

E. <u>Fire Protection</u>

The NRC staff is seeking Commission approval to use the enhanced fire protection criteria previously approved for evolutionary Advanced Light Water Reactor (ALWR) plants by the Commission's Staff Requirements Memorandum (SRM) of June 26, 1990. This SRM approved the staff's position on fire protection as presented in SECY-90-016 and supplemented by the staff's April 27, 1990 response to our report on the SECY. We recommended separate Heating, Ventilating, and Air Conditioning (HVAC) systems for each division as an important step toward ensuring adequate environmental separation of safety systems. The staff agreed that consideration of smoke, heat, and fire suppressant migration may result in separate HVAC systems, but other options may be available to the designer. Our report to the Commission of April 13, 1992, on the Draft Safety Evaluation Report for the ABWR identified the adequacy of physical separation as a continuing issue for the

ABWR, due in part to the use of a shared HVAC system for multiple trains of redundant safety systems during normal plant operation.

Our concern with shared HVAC systems is related to the need for adequate isolation of such systems during certain disruptive events (e.g., fires, floods, or pipe breaks). If the isolation is not adequate, the HVAC arrangement may become a pathway whereby effluents from the event are conducted to locations where required safe shutdown equipment is located. This is not a concern if either (1) the HVAC isolation provisions are able to withstand the event consequences (e.g., pipe whip, jet impingement, static and dynamic pressure, and elevated temperature) during and after closure with consideration of single active component failures and acceptable leakage, or (2) the safe shutdown equipment is qualified for the environmental exposure resulting from a release of the adverse environment at any credible location along the HVAC pathway such as duct openings or blowout locations.

Except for the concern with shared HVAC, we support the staff recommendation that the passive plants should be reviewed against the enhanced fire protection criteria approved in the Commission's SRM.

F. Intersystem Loss-of-Coolant-Accident

The staff's position is that designing these low-pressure fluid systems that interface the reactor coolant system (RCS) to withstand full RCS pressure (to the extent practicable) is an acceptable means for resolving this issue. For those systems that have not been designed to withstand full RCS pressure, the staff indicates that other measures will be required. We recommend approval of the proposed staff resolution, provided consideration is given to all elements of the low_pressure piping system (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

G. Hydrogen Control

The staff recommends that the evolutionary LWR designs provide a system for hydrogen control that can safely accommodate hydrogen generated by the reaction of steam with 100 percent of the fuel cladding surrounding the active fuel. (Note: This is not 100 percent of the reactive metal in the core.) We support the staff's recommendation.

The staff also recommends that the system be capable of precluding uniform containment concentrations of hydrogen greater than 10 percent. We are aware of analytical work in

support of the resolution of Generic Issue 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," that suggests the possibility of transition to detonation at average concentrations as low as 12 percent. We recommend that the staff do a similar analysis of the impact of hydrogen combustion, and possible detonation including stratification, before establishing a limit for the average hydrogen concentration. This is of particular importance to steel-shell containments.

I. <u>High Pressure Core Melt Ejection</u>

To cope with the possible effects of direct containment heating (DCH), the staff concludes, ". . . that ALWR design should include a depressurization system and cavity design features to contain ejected core debris."

DCH is an extremely improbable event, and we see no need to require two modes of coping with the possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is the preferred approach.

J. <u>Containment Performance</u>

The staff has not yet developed an adequate technical position relating to requirements for containment performance in passive LWRs. We agree that the proposed value of 0.1 for a conditional containment-failure probability (CCFP) is reasonable but, as we stated in our letter of April 26, 1990, regarding "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," this value is defined only within the context of a family of initiating events. It should be used by the staff in the development of its requirements and not merely passed on to applicants.

The deterministic criterion proposed by the staff is not a simple alternative to the CCFP. It could be used more logically as a complement. Using ASME Code Service Level C stress limits is not unreasonable given a known loading for which the containment is to be designed. However, determination of the appropriate loading is the hard part of the problem and the suggested deterministic criterion is essentially meaningless without it. The staff states that "applicants using the deterministic approach will be required to define the challenges considered in this evaluation." The staff takes no position on what those challenges should be or how they are to be quantified. Apparently the intent is to default to a "design specific review." This approach leaves

the applicant without any real guidance from the Commission on this important topic.

We acknowledge that it is a very difficult task to establish containment performance criteria but is important. We suggested what we believe to be the best approach in our letter of May 17, 1991, "Proposed Criteria to Accommodate Severe Accidents in Containment Design."

K. Dedicated Containment Vent Penetration

The staff proposes that the decision on the need for a containment vent for passive designs should not be made at this time but should wait until specific plant designs are evaluated. We believe that the Commission should make a generic judgment about the acceptability of containment vents for LWRs. This should be a part of establishing general criteria for containment design as proposed in our letter of May 17, 1991.

L. Equipment Survivability

We agree with the staff's recommendation that features provided only for severe-accident mitigation for the passive plant designs not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirement of 10 CFR 50, Appendix B, and redundancy/diversity requirements of 10 CFR 50, Appendix A.

N. In-Service Testing of Pumps and Valves

We support the staff recommendation that the special pump and valve design, testing, and inspection provisions be imposed on all safety-related pumps and valves for the passive ALWRs.

III.E - Control Room Habitability

There were several significant differences between the staff and EPRI at the time the staff drafted this policy issue. EPRI has subsequently made a proposal to modify its Utility Requirements Document to include a requirement for a passive, safety grade, control room pressurization system that would use a bottled air supply to maintain operator doses within regulatory limits for the first 72 hours following an accident. (The regulations require that operator doses be so limited for the duration of the accident.) The pressurization system proposed by EPRI would be designed to be replenished by off-site portable supplies after 72 hours if needed. Accordingly, EPRI has recommended that the staff close this issue.

We discussed this matter with the staff and EPRI during our June 4-5, 1992 meeting. The staff told us that it is currently evaluating the EPRI proposal and is not prepared to close this issue. ACRS had several comments regarding design features of the passive control room pressurization system proposed by EPRI. We believe that the staff should take these comments into account in its evaluation. We may provide additional recommendations after the staff has completed its evaluation.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Draft SECY Paper dated February 7, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commissioners, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements
2. SECY-90-016 dated January 12, 1990, from James M. Taylor, Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
3. Memorandum dated April 27, 1990, from James M. Taylor, Executive Director for Operations, NRC, for NRC Commission, Subject: Staff Response to ACRS Conclusions Regarding Evolutionary Light Water Reactor Certification Issues

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

May 13, 1992

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT WATER REACTORS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 383rd, 384th, and 385th meetings of the Advisory Committee on Reactor Safeguards, March 5-7, April 2-4, and May 6-9, 1992, we discussed with representatives of the NRC staff the staff's positions, recommendations, and resolution schedules concerning the certification issues for evolutionary and passive light water reactors contained in the draft SECY paper dated February 7, 1992. We also had the benefit of the documents referenced. The staff requested ACRS comments on the draft SECY paper. Our comments and recommendations on some of the staff's positions are given below.

I. SECY-90-016 Issues

Item M. Elimination of Operating Basis Earthquake

Appendix A to 10 CFR Part 100 currently establishes the Operating Basis Earthquake (OBE) at a level one-half of the Safe Shutdown Earthquake (SSE). With this specification, the OBE exerts undue influence over the seismic design and requires a full spectrum analysis in addition to that of the SSE. The staff's proposal is to effectively decouple the OBE from design. We agree with the staff's recommendation.

II. Other Evolutionary and Passive Design Issues

Item A. Industry Codes and Standards

We agree with the staff's recommendation to use the newest codes and standards that have been endorsed by the NRC in its reviews of both the evolutionary and passive plant design applications, and its

recommendation that unapproved revisions to codes and standards be reviewed on a case-by-case basis.

Item D. Leak Before Break

We agree with the staff's recommendation to extend the application of the leak-before-break approach for both evolutionary and passive advanced light water reactors.

Item E. Classification of Main Steamlines of Boiling Water Reactors (BWRs)

We agree with the staff's recommendation for resolution of the main steamline classification for both evolutionary and passive BWRs.

Item F. Tornado Design Basis

Based on a study (NUREG/CR-4661) that compiled a considerable quantity of tornado data, the staff recommends that the maximum tornado wind speed of 300 mph (compared with the present 360 mph) be used for the design-basis tornado. We agree that the best available data should be used, but caution that design-basis specifications have sometimes been established conservatively to provide margins to deal with events not specifically addressed in the design basis. We recommend that the staff's position be approved with a qualification that the staff require assurance that other potential loads that may have been previously subsumed within the tornado design basis be taken into account if necessary.

Item H. Containment Leakage Rate Testing

The staff recommends that the maximum interval between Type C leakage rate tests for both evolutionary and passive designs be increased to a 30-month interval from the 24-month interval now required in 10 CFR Part 50, Appendix J. No significant safety penalty caused by this change has been identified. We agree with the proposed staff position.

Item I. Post-Accident Sampling System (PASS)

The staff is requesting approval of changes in requirements for the PASS currently found in 10 CFR 50.35(f)(2)(viii). These requirements, and the

guidance contained in Regulatory Guide 1.79 and in NUREG-0737, resulted from consideration of the TMI-2 accident.

We agree with the staff's proposal but have the following comments:

1. The requirements as contained in the above referenced regulation refer to "the reactor coolant system and containment that may contain TID-14844 source term radioactive materials" and to measurement of these and other materials. In light of source terms now considered in severe accident analysis, it is advisable to revise this obsolete description.

2. The proposal for "Elimination of the Hydrogen Analysis of Containment Atmosphere Samples" is appropriate, given that safety grade hydrogen monitoring instrumentation will be installed.

3. The Electric Power Research Institute (EPRI) proposed elimination of an existing requirement for the capability to sample the reactor coolant at operating pressure in order to measure the dissolved gas and chloride in the coolant. EPRI claims that maintaining the systems on existing plants produces significant exposure of operating personnel, and that given a severe accident, no useful information, not otherwise available, is provided by this capability. The staff proposes to retain the requirement, but to change the time after accident onset at which the capability must be available from 8 to 24 hours. During our discussion with the staff, we were unable to elicit any reason for this requirement other than that it was established following the TMI-2 accident. We cannot endorse continuation of the requirement for high pressure sampling on the basis of information available to us.

4. The staff proposes approval of a position that "would require the capability to take samples for boron and for activity measurements 8 hours and 24 hours, respectively, after the end of power operation." The intent appears appropriate, however, we suggest that it might be better to specify a time at which the information from measurements becomes avail-

able to the operator rather than the time at which samples can be taken. Further, we assume that what is required is boron concentration rather than the presence or absence of boron. Finally, we suggest that the phrase "after the end of power operation" be made more specific.

Item N. Site-Specific Probabilistic Risk Assessment

If, as concluded by the staff, enveloping analyses are practical for both seismic events and tornadoes, it is appropriate that these be part of the submittal at the time of certification. However, enveloping analyses are not as practical for other external events such as river flooding, storm surge, tsunamis, hurricanes, and volcanism. Therefore, the staff recommends that these other types of site-specific PRA information be submitted at the combined operating license (COL) stage. We agree with this recommendation but would like to hear more about how the staff proposes to deal with any unacceptable findings at the COL stage.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. Draft SECY paper dated February 7, 1992, for the Commissioners, from James M. Taylor, NRC Executive Director for Operations, Subject: Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements (Draft Predecisional)
2. SECY-90-016 dated January 12, 1990 for the Commissioners from James M. Taylor, NRC Executive Director for Operations, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements
3. U.S. Nuclear Regulatory Commission, NUREG/CR-4661, Subject: Tornado Climatology of the Contiguous United States, dated May 1986

ITEM 3: REGULATORY REVIEW GROUP REPORT

The Regulatory Review Group (RRG) was chartered to conduct a comprehensive and disciplined review of power reactor regulations and related processes, programs, and practices with special attention placed on the feasibility of substituting performance-based requirements and guidance founded on risk insights for prescriptive requirements and guidance. The RRG issued a report for public comment on May 28, 1993, and a final report on August 20, 1993. One of the major recommendations by the RRG was in regard to the processes for changes under 10 CFR 50.54, "Conditions of Licenses," and, as proposed, would allow licensees to make changes without prior NRC approval to their quality assurance, security, emergency planning and fire protection plans as long as the changes do not reduce the program or plan's content below that needed to implement prescribed NRC requirements.

The Committee discussed the report and the recommendations by the RRG during the July 8-9, 1993 meeting and issued a report dated July 15, 1993, on this matter.

The following document is attached:

- ACRS Report to the Commission dated July 15, 1993. Subject: Regulatory Review Group Report (PP. 80-82)

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

July 15, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: REGULATORY REVIEW GROUP REPORT

During the 399th meeting of the Advisory Committee on Reactor Safeguards, July 8-9, 1993, we discussed the report prepared by the Regulatory Review Group (RRG) for public comment, and the proposed rulemaking to implement its recommendations on 10 CFR 50.54. Our Subcommittee on Regulatory Policies and Practices considered these matters during a meeting on July 7, 1993. During these meetings, we had the benefit of discussions with representatives of the NRC staff. We also had the benefit of the documents referenced.

Though the report was not entirely responsive to the charter of the RRG, we find that understandable, in view of both the grand scope of the project, and the limited time allotted for the job. We think the RRG has done well.

It is important to maintain some perspective, lest the central issues get lost in a sea of minutiae. We therefore will concentrate here on the broader policy issues.

It is almost axiomatic that regulation without a defined objective tends to be uncontrollable. This presumably led the Commission to promulgate its safety goals and quantitative health objectives in 1986. Unfortunately, in part because that Policy Statement paid no attention to the inevitable uncertainties that arise in the implementation of any quantitative policy, the goals provide little guidance to the staff in discharging its everyday responsibilities. We have indeed urged that their principal use be in judging the effectiveness of the set of deterministic regulations that serve as enforceable surrogates for the goals themselves. Confusing the issue is the question of "adequate protection," words that appear in a minor clause in the Atomic Energy Act, but which play a legal role in the implementation of the Backfit Rule. Continuing along the line from the fundamental (the safety goals), through the regulations (the measures intended to achieve the objective), one

finds at the next level of regulation a potpourri of commitments, understandings, and declarations intended to supplement the rules and regulations in assuring nuclear safety. It is to this level of regulation that the most important recommendation in the report is addressed. (There is of course another level, occasional informal direction of licensees by NRC staff, which is the subject of neither this letter nor the report it reviews.)

Over the years, there has developed, rightly or wrongly, the sense that simple enforcement of the rules and regulations is inadequate to assure satisfactory protection of the health and safety of the public, and there has accumulated a long list of both plant-specific and generic commitments, to which the licensees are bound. Indeed, in the debates about license renewal, one of the stumbling blocks has been the lack of a suitable definition of the current licensing basis on which renewal decisions will be based. Further, and this is the point addressed by the RRG, a licensee seeking relief from a commitment that goes beyond the rules and regulations must prepare a case for the action, and secure NRC permission for the change. The RRG proposes to change this procedure in two related ways, one declaratory, and one procedural, but each with such substantial implications that the changes can not be expected to go down easily.

The RRG proposes that the Commission declare that adherence to the rules and regulations that have evolved constitutes the fundamental condition laid upon a licensee under 10 CFR 50.54, and that the body of further commitments should be viewed as means to that end. This would then have the consequence that a licensee would have the right, while still fulfilling its fundamental obligation, to alter or change commitments that it deems unnecessary to meet the rules and regulations, without seeking prior NRC approval. NRC would of course have to be notified, and the rationale available. NRC could then object on the basis that the action may have brought the licensee into conflict with a rule or regulation, but only on that basis. Then the burden of proof would lie with the NRC to make its case. In this way, conformance to the rules and regulations would be the governing obligation of the licensee.

This would constitute a fundamental change, and is likely to receive a rather thorough set of reviews and analyses before it takes effect, so we think it premature to comment about the more detailed implementation recommendations contained in the report. They will surely change under scrutiny.

We think that the RRG recommendation is a substantial positive step, worth serious consideration by the Commission. We do not recommend that the RRG be continued past its scheduled dissolution,

but are concerned that natural resistance to change will bury one of the few recent proposals for substantial change, without due process. We therefore recommend that you take the steps necessary to move the recommendations to the next phase, which is more detailed consideration about how such a fundamental change might be implemented. In view of our earlier discussion of the relationships among the formal and informal elements of the regulatory structure, this would be a step toward coherence.

Sincerely,

J. Ernest Wilkins, Jr.

J. Ernest Wilkins, Jr.
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Regulatory Review Group Report, Volumes 1-4, issued for public comment on May 28, 1993
2. Memorandum dated June 25, 1993, from Frank P. Gillespie, Regulatory Review Group, for John T. Larkins, Executive Director, ACRS, Subject: Proposed Rulemaking to Implement the Regulatory Review Group Recommendations on 10 CFR 50.54 (Draft Predecisional)