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HISTORICAL PERSPECTIVES AND INSIGHTS ON ACRS REVIEW OF GE ABWR DESIGN CERTIFICATION

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) requires that each application for a standard design certification be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report on those portions of the application which concern safety. This paper begins with perspectives on the role of the ACRS in the design certification review process. It then summarizes the ACRS observations and recommendations made in the Committee's reports during the General Electric Nuclear Energy (GENE) U. S. Advanced Boiling-Water Reactor (ABWR) design certification reviews.

1. INTRODUCTION

For over 50 years the Advisory Committee on Reactor Safeguards (ACRS) has had a continuing statutory responsibility for providing independent reviews of, and advising on, the safety of proposed or existing reactor facilities and the adequacy of proposed reactor safety standards. The application for certification of the ABWR design was filed by GENE in September 1987. The ACRS held meetings and discussions as early as in 1989 on the review of ABWR design certification. The ACRS has played an important role in the GE ABWR design certification process by providing an independent review of the NRC staff's determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations. The ACRS identified many technical issues during its review process which were resolved before the Committee provided its final recommendations on the design certification in April 1994.

As a part of its commitment to the U.S. NRC effort in knowledge management (KM), the Office of ACRS has begun an initiative to capture the institutional knowledge and memory of the Committee. An important motivation for this initiative is to increase the effectiveness and efficiency of the Committee's review process by providing easy access to the background information, insights, and understanding of the technical and regulatory issues. This paper begins with discussions on the role of ACRS in the design certification review process. It then provides historical perspectives and insights on ACRS review of the GE ABWR design certification.

2. THE ROLE OF ACRS IN DESIGN CERTIFICATION REVIEW PROCESS

The ACRS provides the NRC with independent reviews of, and advice on, the safety of proposed and existing reactor facilities and the adequacy of proposed safety standards. The ACRS is independent of the NRC staff and reports directly to the Commission, which appoints its members. The ACRS is structured to provide a forum where experts representing many technical perspectives can provide independent advice that is considered in the Commission's decision-making process.

The NRC may approve and certify a standard nuclear plant design through a rulemaking, independent of a specific site and an application to construct or operate a plant. A design certification is valid for 15 years from the date of issuance, but can be renewed for an additional 10 to 15 years.

Figure 1 illustrates the design certification formal review process and the interrelationships among various review activities. According to NRC regulation (10 CFR 52.53), the design certification application is referred to the ACRS for a review and report. An application for a standard design certification must demonstrate how the applicant complies with the NRC's relevant regulations. The application presents the design basis, the limits on operation, and a safety analysis of structures, systems, and components of the facility as a whole. In addition, the application must contain proposed inspections, tests, analyses, and acceptance criteria (ITAAC) for the standard design. The ITAAC program is intended to ensure that the plant, when built, will conform to the design parameters and assumptions that existed at the time of design certification.

The views expressed in this paper are solely those of the authors and do not necessarily represent those of either the ACRS or NRC

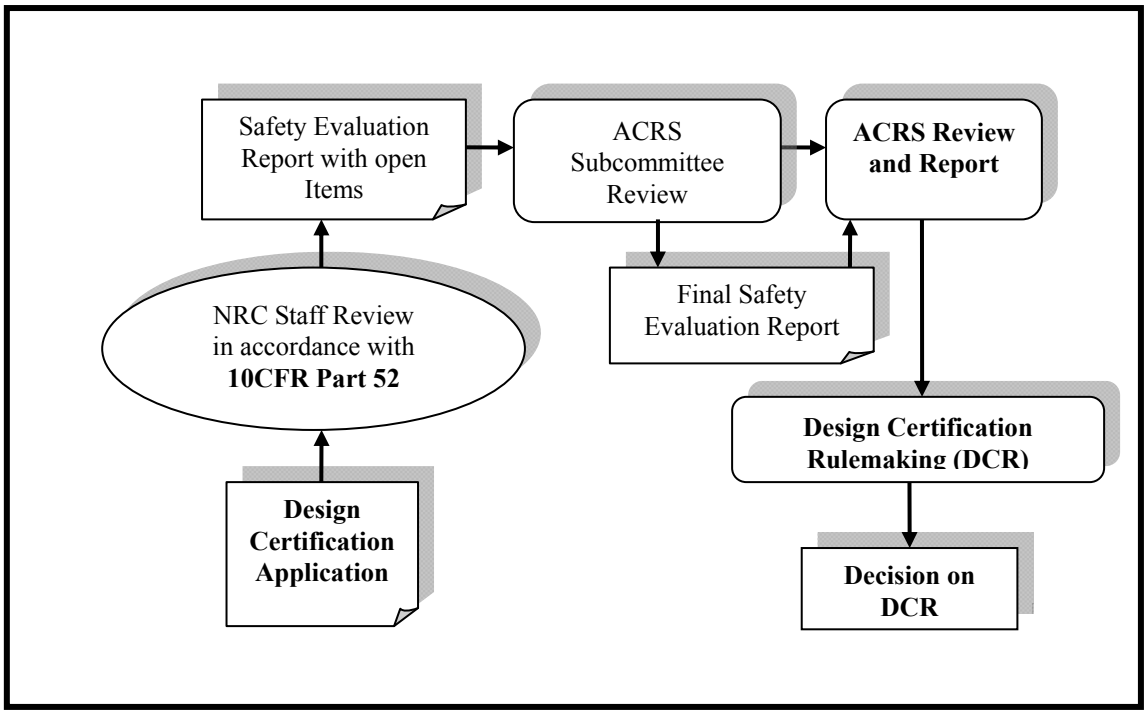

Figure 1. The Design Certification Review Process

The NRC staff reviews the design certification application and supporting documentation. The review results are documented in a safety evaluation report. The design certification application and the staff's safety evaluation report are reviewed by the ACRS. The applicant and staff appear before the ACRS, the former defending the application and the latter the safety evaluation report. In-depth reviews are done by the appropriate ACRS Subcommittees. The subcommittee has the benefit of input from its own consultants as well as its own staff support. With input from Subcommittee members, Subcommittees Chairmen develop proposed ACRS positions. Briefings by the applicant and the NRC staff are provided to both the Subcommittees and the full Committee. The ACRS meetings are open to public and thus provide an open forum for public participation in the review process. ACRS positions are developed after extensive deliberations by the full Committee. When the Committee has completed its review, its report is submitted to the Commission. At times, ACRS issues "interim" letters to identify issues of concern and items for which additional information, discussions, and clarifications are needed.

ACRS operations are governed by the Federal Advisory Committee Act (FACA), which is implemented through NRC regulations (10 CFR Part 7). ACRS operational practices encourage the public, industry, State and local governments, and other stakeholders to express their views on regulatory matters. According to FACA requirements all the ACRS records, reports, transcripts, or other documents, which are made available to or prepared for or by the Committee, are publically available, subject to the provisions of the Freedom of Information Act (5 U.S.C. 552).

3. ACRS REVIEW OF ABWR DESIGN CERTIFICATION

The ABWR is a single-cycle, forced-circulation, boiling-water reactor (BWR), with a rated power of 3926 MWt, designed by GENE. The design incorporates features of the BWR designs in Europe, Japan, and the United States, and uses improved electronics, computer, turbine, and fuel technology. The design also includes safety enhancements, such as use of reactor internal pumps, protection against overpressurizing the containment, passive core debris flooding capability, an alternate current (AC) independent water addition system, three emergency diesels, and a combustion turbine as an alternative power source. The U.S. version of the ABWR standard design utilized a significant portion of the detailed design information developed jointly by GENE, Hitachi, and Toshiba for the international version which was built at the Kashiwazaki Kariwa Nuclear Power Generation Station, Units 6 and 7 (K-6/7), by the Tokyo Electric Power Company, Inc.

The ACRS review activities for the ABWR design certification are depicted in Figure 2. Between February 1988 when the ABWR application was docketed and April 1994 when the ACRS issued its final report on the safety aspects of

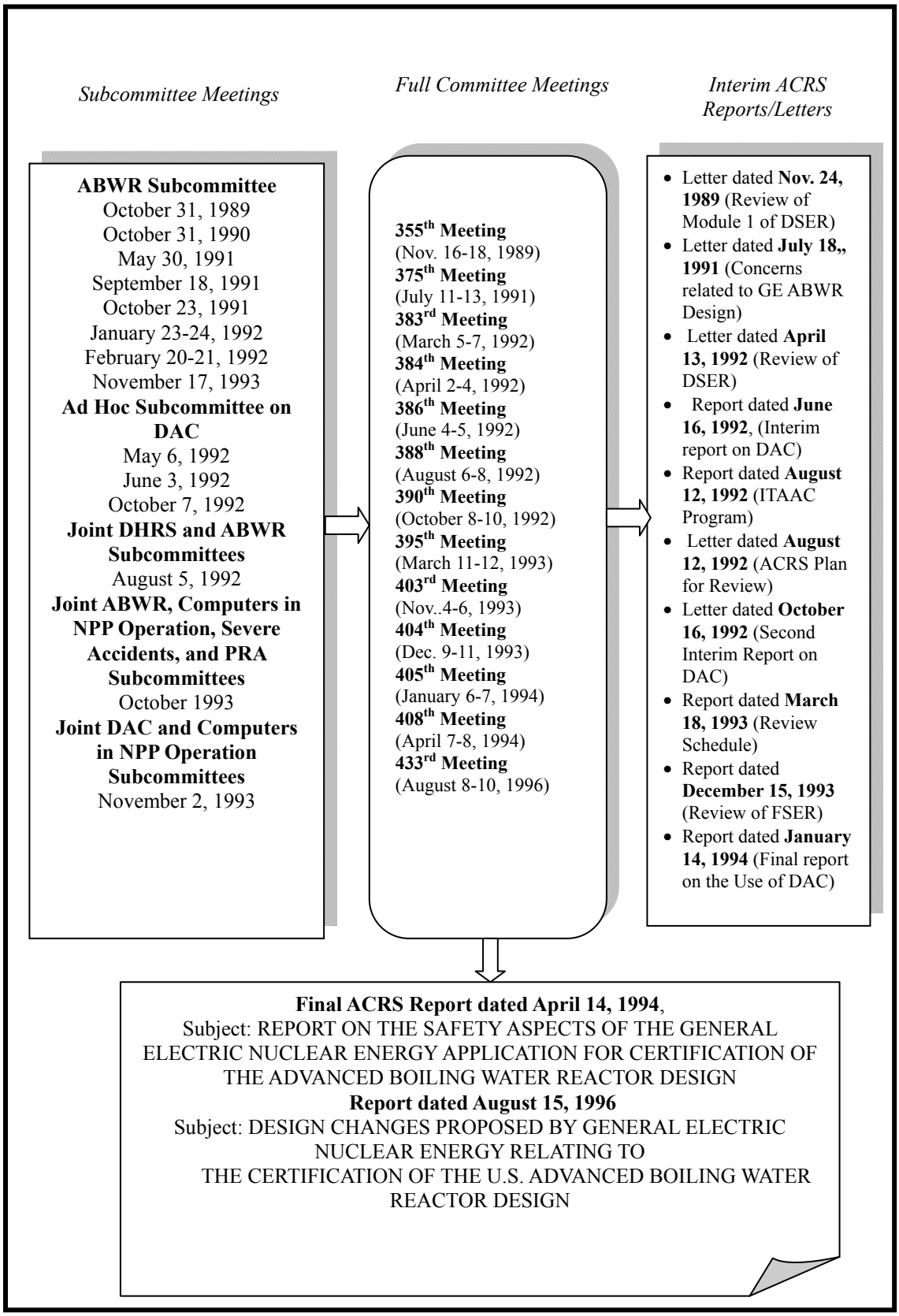

Figure 2 The ACRS review activities for the GENE ABWR design certification

the GENE application for certification of the ABWR design, the ACRS Subcommittee on ABWR held numerous meetings to review the ABWR Standard Safety Analysis Report (SSAR) and the NRC staff safety evaluations. During this same period, The ACRS subcommittee on Improved Light Water Reactors held several meetings to review the Electric Power Research Institute (EPRI) Utility Requirements Document (URD) and associated NRC staff safety evaluations for the Advanced Light-Water Reactor (ALWR) evolutionary plant. (The EPRI URD prescribed ALWR design requirements from the utility industry perspective.) Meetings were also held by the ACRS subcommittees on Computers in Nuclear Power Plant Operations, Human Factors, PRA, and Severe Accidents. These subcommittees reviewed a number of specialized aspects of the proposed ABWR design including those related to fire, digital control and protection systems, human factors, and severe accidents. The ACRS identified many technical issues during its review process which were resolved before the Committee provided its final recommendations on the design certification in April 1994.

The design certification application was based on the SSAR, which was submitted in modular form between September 1987 and March 1989. The NRC staff issued a Draft Safety Evaluation Report (DSER) on the first module of the SSAR in August 1989 [1]. The ACRS reviewed this draft and reported its findings in November 1989. In its November 24, 1989 letter on module 1 of the draft safety evaluation report for the advanced ABWR design review [2], the Committee questioned, in particular, the adequacy of the level of design detail available for review and recommended that the staff revisit the issue of what constitutes an "essentially complete" design.

3.1 ACRS Initial Concerns Related to the GE ABWR Design

Subsequent to November 1989, the ACRS review activities focused on several ABWR-related design concerns including control building flooding, physical separation, environmental protection of sensitive equipment, performance of essential chilled water systems, use of leak-before-break methodology, use of integral low-pressure turbine rotors, and the capability of the floor area beneath the reactor vessel to cope with severe accidents. These preliminary concerns were brought to the attention of the NRC staff in an ACRS letter dated July 18, 1991 [3] to ensure an early awareness and understanding. The ACRS believed that it was appropriate to document them in a letter for timely consideration and resolution in appropriate DSER sections.

In1991, the DSER was completed by the NRC staff in the form of six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). The ACRS Subcommittee on ABWR 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. During the 383rd and 384th meetings of the ACRS, March 5-7 and April 2-4, 1992, the Committee discussed the DSERs and documented its findings in a letter dated April 13, 1992 [4]. In this letter, the Committee reconfirmed the preliminary concerns expressed in its July 18, 1991 letter and added several more including adequacy of the PRA, containment hydrodynamic loads, Reactor Water Cleanup System safety implications, plant design life and aging management, station grounding and surge protection, and corrosion control for structures. A summary of some of these concerns, extracted from the ACRS April 13, 1992 letter, are provided below:

<u>CONTROL BUILDING FLOODING</u>

"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 un-isolated 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."

<u>ADEQUACY OF PHYSICAL SEPARATION</u>

"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."

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."*

ADEQUACY OF SSAR TREATMENT OF THE REACTOR WATER CLEANUP SYSTEM

"*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."

ADEQUACY OF THE ABWR PRA

"*... 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)."*

Regarding the form, purpose and role that PRAs are to play in severe accident analysis and decision making, the staff responded that the criteria specified for resolution of severe accident issues in SECY-90-016 [5], and its successor, SECY-93-087 [6] will be incorporated into the ABWR design certification rulemaking as applicable regulations.

3.2 ACRS Review of the Use of the DAC Process in the Certification of the GE ABWR Design

In 1992, the NRC staff proposed the use of Design Acceptance Criteria (DAC) during 10 CFR Part 52 design certification reviews. This approach was being proposed because complete detailed design information had not been provided to the staff by an applicant for a design certification. In its February 14, 1992 report on the use of DAC during 10 CFR Part 52 design certification reviews [7], the ACRS supported the DAC approach for limited applications and encouraged the staff to continue development of the process with appropriate interchange with vendors and the industry. However, the ACRS believed that "carefully defined limits relating to scope and extent of design coverage should be placed on the use of DAC by the staff." The Committee also recommended that "*the use of DAC be limited to that portion of each given design feature where either the technology is still evolving (e.g., certain portions of the plant instrumentation and control or control room design) or the required information is unavailable for good reason. In any case, DAC should be used only when it is possible to specify practical and technically unambiguous criteria*."

In 1992, the ACRS established an Ad Hoc Subcommittee on DAC to review the DAC process as requested by the Commission. In its June 16, 1992 interim report on the use of DAC in the certification of the GENE ABWR design [8], the ACRS noted that, in general, they were satisfied with the progress that the staff and GE were making in the development of the DACs (each consisting of a set of DAC/ITAACs – Inspections, Tests, Analyses, and Acceptance Criteria) envisioned for use in the certification of the GE ABWR design.

In its October 16, 1992 second interim report on the use of the DAC process in the certification of the GENE ABWR design [9], the ACRS raised concerns about the significant number of post- design certification activities associated with two DACs: control room design, and instrumentation and controls. The Committee noted that "*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*."

In January 1994, the ACRS completed its review of the DAC to be included in the Certified Design Material (CDM) for the GENE ABWR. The four subject areas addressed by DAC were Human Factors Engineering, Radiation Protection, Piping Design, and Instrumentation and Control. In its January 14, 1994 final report on the use of the design acceptance criteria process in the certification of the GENE ABWR design [10], the ACRS noted that, with respect to CDM covering the four subject areas historically referred to as DAC, it was generally satisfied that it provides a reasonable basis for the staff final safety determination needed to support final design approval.

3.3 ACRS Review of Design-Specific Requirements Relating to the Various Evolutionary and Advanced Light Water Reactor Policy, Technical, and Licensing Issues

During the ACRS review of the ABWR SSAR, the Committee considered the design-specific requirements which related to the various evolutionary and advanced light water reactor policy, technical, and licensing issues included in SECY-90-016 [5] and its successor, SECY-93-087 [6]. These issues incorporated staff positions that deviated from or were not embodied in regulations at that time. Their resolutions would have become "applicable regulations" through incorporation into the design certification rule for the ABWR. The ACRS comments and recommendations concerning some of these issues pertaining to ABWR design are reported here.

FIRE PROTECTION

In its April 26, 1990 report on evolutionary light water reactor certification issues and their relationship to current regulatory requirements [11], the ACRS pointed out that redundant train separation is likely to be the most significant feature leading to reduced fire risk. The Committee 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.

In its April 26, 1993 report [12] on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," the ACRS noted that:

> "*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 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.*"

DEFENSE AGAINST COMMON-MODE FAILURE IN DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

> "*The staff's second recommendation is that the vendor or applicant analyzes 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."*"

CONTROL ROOM ANNUNCIATOR (ALARM) RELIABILITY

> "*The staff's basic recommendation is that the Commission approve the position that the alarm system for ALWRs meets 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.*"

3.4 ACRS Conclusion Concerning GENE ABWR Safety

Between April 1992 and April 1994, the ACRS ABWR subcommittee held additional meetings to review significant design changes in the SSAR and closure of all open items in the DFSER. It also included a review of written responses by

GENE to numerous questions and concerns raised by the subcommittee.

In its April 14, 1994 report on safety aspects of the GENE application for certification of the ABWR design [13], the ACRS made the following conclusion:

> *"Based on the results of our review of those portions of the GENE ABWR application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the U.S. version of the ABWR standard design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public."*

ACRS also reviewed design changes proposed in 1996 by GENE relating to the certification of the U.S. ABWR design. These "design changes" consisted of both actual modifications to the design and corrections to the documentation to remove inconsistencies and typographical errors. In an August 15, 1996 ACRS report [14], the Committee stated that those changes did not change the conclusion reached in the ACRS earlier report of April 14, 1994. The Committee noted that the ACRS *"continues to believe that acceptable bases and requirements have been established in the application to assure that the U.S. ABWR Standard Design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public*."

4. SUMMARY AND CONCLUSIONS

The U.S. NRC design certification formal review process and the interrelationships among various review activities were discussed. The ACRS played an important role in GENE ABWR design certification process by providing an independent review of the NRC staff's determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations.

The ACRS identified many technical issues during its review process which were resolved before the Committee provided its final recommendations on the design certification.

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