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

Dear Chairman Selin:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE ASEA BROWN BOVERI - COMBUSTION ENGINEERING APPLICATION FOR CERTIFICATION OF THE SYSTEM 80+ STANDARD PLANT DESIGN

During the 409th meeting of the Advisory Committee on Reactor Safeguards, May 5-7, 1994, we completed our review of the ASEA Brown Boveri - Combustion Engineering (ABB-CE) application for certification of the System 80+ standard plant design. This report is intended to fulfill the requirement of 10 CFR 52.53 that the ACRS "... report on those portions of the application which concern safety." During our review, we had the benefit of discussions with representatives of the NRC staff, ABB-CE and its contractors, Duke Engineering and Services, Inc., and Stone and Webster Engineering Corporation. We also had the benefit of the documents referenced.

System 80+ Application

The application for certification of the System 80+ design was filed on March 30, 1989, under the provisions of Appendix O to 10 CFR Part 50 and the NRC Policy Statement on Nuclear Power Plant Standardization (Ref. 1). In its letter of August 21, 1989, CE (which has been referred to as ABB-CE since May 26, 1992, as a result of CE becoming a subsidiary of ABB) stated that the application may be considered to have been submitted pursuant to 10 CFR 52.45 (Ref. 2). The application was docketed on May 1, 1991, and assigned Docket No. 52-002.

The application is based on the CE Standard Safety Analysis Report - Design Certification (CESSAR-DC), which describes the design of the facility and the site-specific interface requirements. The CESSAR-DC was originally submitted on March 30, 1989. Subsequently, ABB-CE supplemented the information in CESSAR-DC through a number of amendments. The last amendment that we received was Amendment V dated April 29, 1994. ABB-CE also submitted certified design material (CDM) (Ref. 3) on December 31, 1993, which contains Tier 1 design information which ABB-CE proposes to have certified under 10 CFR Part 52 by design certification rulemaking.

System 80+ Design Description

The ABB-CE System 80+ standard plant is designed for use at either single-unit or multiple-unit sites. In accordance with 10 CFR 52.47(b)(1), the design scope must provide an essentially complete nuclear power plant design except for site-specific elements of the design, such as the service water intake structure and the ultimate heat sink. The design evolved from the CE System 80 plant design. Three units of the System 80 design (Palo Verde Units 1, 2, and 3) have been licensed to operate in the United States.

The CESSAR-DC states that the Electric Power Research Institute (EPRI) Evolutionary Light Water Reactor Utility Requirements Document (URD) was used as a guide for the design of the System 80+ plant. Although there are some remaining differences between the System 80+ design and the EPRI URD, we do not view these differences to be significant from a nuclear safety perspective.

Four aspects of the plant design, i.e., piping design, radiation protection, instrumentation and control (I&C) design, and human factors engineering for the design of main control room and remote shutdown panel, will be completed by the Combined Operating License (COL) applicant/holder using a staff-approved design process described within the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). These ITAAC, which will be a part of the CDM, appear to be an appropriate use of the "Design Acceptance Criteria" process, which we discussed in our report of January 14, 1994 (Ref. 4).

The System 80+ nuclear steam supply system (NSSS) consists of a pressurized water reactor (PWR) with two primary coolant loops utilizing vertical U-tube steam generators. Each loop has two reactor coolant pumps. A pressurizer is connected to one of the loops. The NSSS also includes related auxiliary and engineered safety feature (ESF) systems.

The rated core thermal power is 3914 MWt. The design core thermal power, at which accidents are evaluated, is 3992 MWt. The reactor core consists of 241 16x16 Zircaloy-clad fuel assemblies and 93 control element assemblies.

The reactor containment is a 200 foot diameter spherical steel shell that is completely enclosed by a reinforced concrete Shield Building. The lower elevations of this building (the subsphere) house the four physically separated trains of shutdown cooling and ESF mechanical equipment.

The Shield Building is located within the Nuclear Island structure which also contains the fuel pool area, the maintenance outage area, the main steam valve enclosure, the two Class 1E emergency diesel generators and their dedicated batteries, and the control complex for the plant.

The Turbine Building and the Radwaste Building are located on opposite ends of the Nuclear Island. The Turbine Building, which contains no safety-related equipment, houses the 1800 rpm turbine generator and its auxiliary systems, and major components of the condensate and feedwater systems. The turbine generator is oriented so as to reduce the likelihood of damage to safety-related equipment in the event of turbine failure. The Radwaste Building houses equipment for the collection and processing of radioactive waste generated by the plant.

The component cooling water heat exchangers are located within structures in the yard which surrounds the Nuclear Island, thereby eliminating the potential for flooding within the Nuclear Island due to service water pipe breaks. The combustion turbine generator (the Alternate AC power source) and its fuel supply are also located within structures in the yard. Other yard structures include the fire pump house and associated tanks, diesel fuel oil and miscellaneous water storage tanks.

Safety Enhancement Features

The ABB-CE System 80+ design includes a number of features that we believe will enhance safety relative to past PWR designs. Some of these features resulted from the use of Probabilistic Risk Assessment (PRA) methodology by ABB-CE during the System 80+ design process. The more significant features include:

> The reactor vessel is fabricated using ring forgings that eliminate the need for beltline longitudinal welds. Combined with improved material specifications, this reduces concern over reactor vessel integrity.
>
> The pressurizer and the steam generators have larger water inventories (on a volume to MWt basis) than present PWRs. This improves plant response to most transients and reduces unnecessary challenges to safety systems. In addition, the steam generators use Inconel 690 tubing, which is expected to reduce susceptibility to tube failures.
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> The safety injection system (SIS) uses four half-capacity, physically separated mechanical trains that inject directly into the reactor vessel. The SIS is designed for full-flow testing during power operation. In addition to the SIS, four safety injection tanks are provided in the design. Under design basis loss of coolant accident (LOCA) conditions, these systems meet Appendix K to 10 CFR Part 50 over the spectrum of LOCA break sizes. The reactor core is expected to remain covered with water for breaks up to a 10 inch direct vessel injection line break.
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> An in-containment refueling water storage tank with external refill capability is provided as a source of borated water for both initial injection and long-term recirculation phases of the LOCA and for manually initiated cavity flooding under severe accident conditions. The tank also serves as the heat sink for the manually actuated safety depressurization system (SDS). The SDS provides the capability to rapidly depressurize the reactor coolant system, allowing the operator to initiate primary system feed and bleed during a total loss of feedwater event.
>
> The emergency feedwater system (EFWS) has two physically separated divisions, each consisting of an EFWS tank, a full-capacity motor-driven pump, and a full-capacity turbine-driven pump. Each EFWS division can feed both steam generators.
>
> The pressure boundary for the shutdown cooling system (SCS) is rated at 900 psig. This reduces concern for intersystem

LOCAs. The SCS can be interconnected with the containment spray system. The pumps from either system can serve as backup to the pumps in the other system.

The reliability of reactor coolant pump seal cooling has been improved by the inclusion of a seal cooling pump that can be powered from the combustion turbine generator under station-blackout conditions. This air-cooled pump can also provide seal cooling during loss of normal cooling water events. This pump is in addition to the charging pumps and component cooling water supplies that normally provide for reactor coolant pump seal cooling.

Safety-related systems and trains that perform redundant functions are physically separated by appropriate barriers that provide protection against fires, floods, and similar common-cause challenges.

The design provides for two independent offsite power connections from a main switchyard and a separate backup switchyard. The turbine generator is designed to run back and continue carrying plant auxiliary loads in the event of separation from the grid at maximum load. This feature should reduce the frequency of reactor trips following a loss of offsite power. A combustion turbine generator provides an alternate source of AC power in the event of station blackout.

The main control complex makes use of an evolutionary design referred to as Nuplex 80+. This complex includes the main control room, the remote shutdown room, the computer room, the technical support center, and the I&C and equipment rooms located throughout the plant. The increased use of digital control and protection systems in this design offers the potential for improving both the operator interface with the plant and the reliability of control and protection systems. The design also reduces the amount of electrical cabling, thereby reducing the potential for fire in safety-related areas.

The 3.4 million cubic feet free volume reactor containment is large and has a higher pressure capability under severe accident conditions (estimated median ultimate containment failure pressure of 172 psia at 290°F) than most operating PWRs. These features provide added protection against early severe accident containment challenges such as hydrogen combustion and direct containment heating. They also increase the time to late containment failure due to overpressure. Provision has been made for limited unfiltered containment venting, although venting is not expected to be needed for most severe accident conditions.

The containment design provides the capability for flooding a large (relative to current PWRs) lower reactor cavity debris spreading area prior to vessel breach. This flooding capability can be activated independently of AC power sources. In addition, a thick basemat made with ablation resistant concrete is used.

The design provides a massive reactor cavity/reactor vessel support structure. This structure is intended to withstand the pressure that could result from direct containment heating or ex-vessel fuel coolant interaction. A convoluted de-entrainment pathway is provided between the cavity and the upper containment to minimize the expulsion of corium out of the cavity during a core melt ejection event.

The design includes a hydrogen mitigating system employing manually activated glow plug igniters at 40 locations (two independently powered igniters per location) in the containment. Care was used in the design to vent those compartments where hydrogen could accumulate.

The containment spray system (CSS) uses two independent trains. A connection is provided to the CSS for an emergency containment spray backup system, consisting of a cooling pond water source, and a portable pump capable of being driven independently of AC power sources.

Design features that minimize shutdown and low power operation risk were analyzed with the result that no significant design vulnerabilities were found for accidents involving shutdown and low power operations.

Chronology of ACRS Review

Our review of the System 80+ application commenced after it was filed in March 1989. We held a series of Subcommittee meetings between April 1990 and February 1993. The staff issued a Draft Safety Evaluation Report (DSER) on October 1, 1992 (Ref. 5). In December 1993, the ACRS Subcommittee on ABB-CE Standard Plant Designs began a series of meetings dedicated to the final review of the CESSAR-DC and related material. This series of meetings built upon and continued the previous ACRS activities, and provided the basis for this report. The staff issued a Final Safety Evaluation Report (FSER) on March 3, 1994 (Ref. 6). Our activities related to System 80+ are described in the attachment.

ACRS Conclusion Concerning System 80+ Safety

Based on the results of our review of those portions of the ABB-CE System 80+ application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the System 80+ standard plant 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.

Sincerely,

T. S. Kress
Chairman

References:
1. U.S. Nuclear Regulatory Commission, Policy Statement, 10 CFR Part 50, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987
2. Letter dated August 21, 1989, from A.E. Scherer, CE, to T.E. Murley, NRC, Subject: Design Certification of the System 80+TM Standard Design
3. Letter dated December 31, 1993, from C.B. Brinkman, ABB-CE, to USNRC Document Control Desk, Subject: System 80+TM ITAAC Submittal
4. ACRS report dated January 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design
5. Letter dated October 1, 1992, from R.C. Pierson, NRC, to C.B. Brinkman, ABB-CE, Subject: Draft Safety Evaluation Report (DSER) of Nuclear Regulatory Commission (NRC) Staff Review of Combustion Engineering (ABB-CE) Standard Safety Analysis Report for Design Certification of System 80+ (NUREG-1462)
6. Letter dated March 3, 1994, from James M. Taylor, NRC Executive Director for Operations, to the NRC Commissioners, Subject: Advance Copy of the Final Safety Evaluation Report (FSER) on the ABB-Combustion Engineering System 80+ Standard Design Certification and Certified Design Material (CDM)

Attachment:
Chronology of ACRS Review

ATTACHMENT - CHRONOLOGY OF ACRS REVIEW

Discussions during the following ACRS Subcommittee and Full Committee meetings included the listed topics on ABB-CE System 80+:

April 3, 1990 - Advanced PWR Subcommittee

Licensing Review Basis (LRB) document, reactor coolant system, engineered safety feature systems, containment, Nuplex 80+, and probabilistic risk assessment (PRA)

September 21, 1990 - Advanced PWR Subcommittee

Use of operational experience at existing Combustion Engineering plants, including reactor coolant pump impellers, resistance temperature detectors, heated junction thermocouples, upper guide structure, safety injection nozzle thermal sleeves, steam generator geometry and operating parameters, fire protection, security, and flood design

November 1, 1990 - Advanced PWR Subcommittee

Licensing Review Basis Document. An ACRS report was issued on

November 14, 1990, regarding the LRB document for the Combustion Engineering, Inc. System 80+ Evolutionary Light Water Reactor.

February 6, 1991 - Joint meeting of the Subcommittees on Computers in Nuclear Power Plant Operations, and Instrumentation and Control (I&C) Systems on computer applications in advanced plant designs

Nuplex 80+ software reliability

March 6, 1991 - Advanced PWR Subcommittee

Design basis accident analysis, and seismic methodologies

September 4, 1991 - Advanced PWR Subcommittee

Piping layout, Nuplex 80+ advanced control room design, and PRA

December 3 and 4, 1991 - Joint meeting of the Subcommittees on Advanced PWR and Computers in Nuclear Power Plant Operations with Westinghouse and CE regarding digital computer experiences at nuclear power plants

Core Protection Calculator improvements and remote multiplexing

March 4, 1992 - Joint meeting of the Subcommittees on Computers in Nuclear Power Plant Operations, I&C Systems, and Human Factors with representatives of EPRI, CE, Westinghouse, and Software Engineering Institute

Nuplex 80+ control room design bases and features

September 10-12, 1992 - 389th ACRS meeting

Defense against common-mode failures in digital I&C systems

February 10, 1993 - Advanced PWR Subcommittee

Design overview, human factors engineering, protection for common-mode software failure of I&C systems, physically based radiological source term, and radiological equipment qualification

December 8, 1993 - ABB-CE Standard Plant Designs Subcommittee

Combustion Engineering Standard Safety Analysis Report-Design Certification (CESSAR-DC) and NRC staff Final Safety Evaluation Report (FSER) Chapters 7, 8, and 18

February 9, 1994 - ABB-CE Standard Plant Designs Subcommittee

CESSAR-DC and FSER Chapters 4, 10, 11, 12, 13, 14 (section 2), and 17

March 8 and 9, 1994 - ABB-CE Standard Plant Designs Subcommittee

CESSAR-DC and FSER Chapters 2, 3, 14 (section 3), and 19

March 17, 1994 - Palo Verde Nuclear Generating Station Site Visit

Several members of the ACRS attended a fact-finding visit which included familiarization with the plant, site arrangement, and operating history of the System 80 design

April 5 and 6, 1994 - ABB-CE Standard Plant Designs Subcommittee

CESSAR-DC and FSER Chapters 1, 5, 6, 9, 15, 16, and CESSAR-DC Appendix A (FSER Chapter 20). In addition, during this meeting the Subcommittee reviewed the applicant's evaluation that, for the worst credible accident, the dose at the site boundary (one-half mile from the reactor) will remain below the Environmental Protection Agency's lower Protective Action Guideline of 1 rem. This is expected to be the subject of a separate Committee report.

May 5-7, 1994 - 409th ACRS Meeting

ABB-CE and NRC staff responses to questions asked by ACRS members during previous Subcommittee meetings