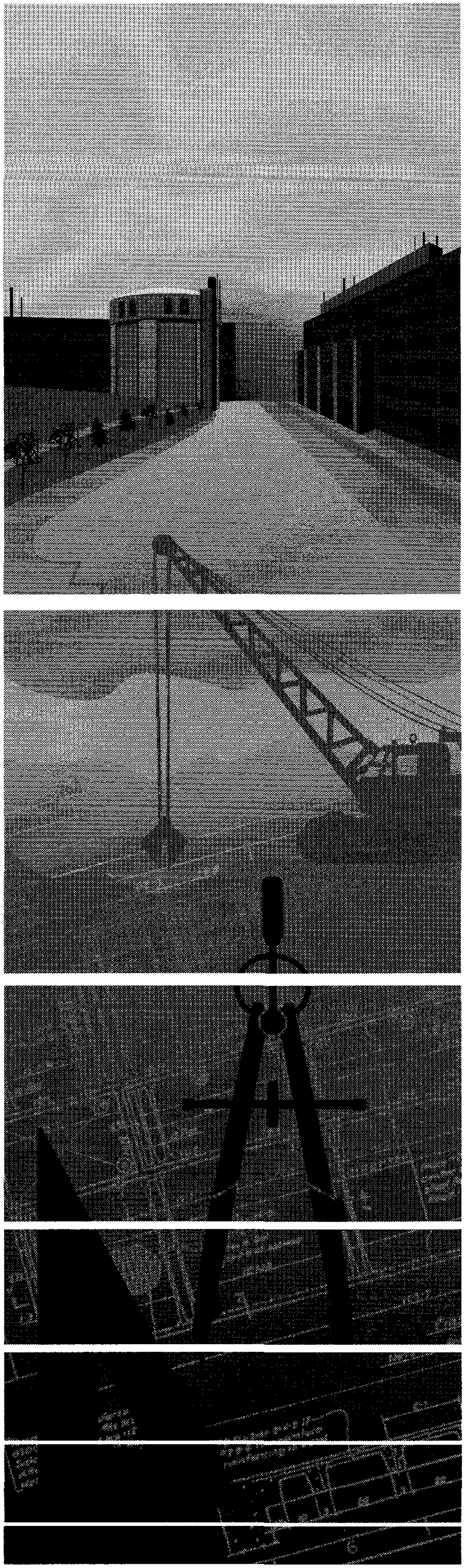

Final Safety Evaluation Report

Related to the Certification of the System 80+ Design

Supplement 1

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

May 1997

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Final Safety Evaluation Report Related to the Certification of the System 80 + Design

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ABSTRACT

This report supplements the final safety evaluation report (FSER) for the System 80+ standard design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) staff as NUREG-1462 in August 1994 to document the NRC staff's review of the System 80+ design. The System 80+ design was submitted by Asea Brown Boveri-Combustion Engineering (ABB-CE), in accordance with the procedures of Subpart B to Part 52 of Title 10 of the Code of Federal Regulations. This supplement documents the NRC staff's review of the changes to the System 80+ design documentation since the issuance of the FSER. ABB-CE made these changes as a result of its review of the System 80+ design details. The NRC staff concludes that the changes to the System 80+ design documentation are acceptable, and that ABB-CE's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the System 80+ design.

CONTENTS

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This report supplements the final safety evaluation report (FSER) for the System 80+ standard design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) staff as NUREG-1462 in August 1994 to document the staff's review of the System 80+ design. Asea Brown Boveri-Combustion Engineering (ABB-CE) made changes to the System 80+ design documentation, after issuance of the FSER, as a result of its review of the System 80+ design details. This supplement documents the NRC staff's evaluation of these changes to the System 80+ design and it also provides errata to the FSER.

Combustion Engineering, Inc. (ABB-CE, the applicant) submitted the System 80+ design documentation under Subpart B of Part 52 of Title 10 of the Code of Federal Regulations. The documentation and information pertaining to this supplement were submitted on Docket No. 52-002. The design documentation includes the standard safety analysis report (SSAR), certified design material (CDM), and design control document (DCD).

Each of the following sections or appendices of this supplement is numbered and titled the same as the section or appendix of the FSER that is being updated. The discussions are supplementary to and not in lieu of the discussion in the FSER unless otherwise noted. Accordingly, Appendix D is a list of the principal contributors to this supplement and Appendix F contains errata to the FSER. No changes were made to FSER Appendices A, B, C, and E by this supplement.

This supplement is issued by the Standardization Project Directorate in the Office of Nuclear Reactor Regulation. The licensing project manager for the System 80+ design is Jerry N. Wilson, PE. He may be reached by calling (301) 415-3145, or by writing to the Office of Nuclear Reactor Regulation, Mail Stop O-10-D-22, U.S. Nuclear Regulatory Commission, Washington, DC. 20555-0001. Copies of the System 80+ design documentation and all amendments and revisions are available for public inspection at the NRC's Public Document Room, 2120 L Street, NW. (Lower Level), Washington DC. Copies of the FSER (NUREG-1462) and this supplement are also available at the NRC's Public Document Room.

1.5 Summary Of Principal Review Matters

The NRC staff stated in the FSER that, subsequent to the completion of the staff's review of the SSAR and CDM for the System 80+ design, ABB-CE will submit a DCD for the staff's review. The DCD, which will be incorporated by reference into the final design certification rule, has two tiers of information that were derived from and include most of the information in the CDM and the SSAR.

ABB-CE submitted the DCD for the staff's review on December 16, 1994. In general, ABB-CE followed the NRC staff guidance in letters dated August 26, 1993, and August 3, 1994, regarding the format of the DCD. The staff provided comments on the DCD in a letter dated January 27, 1995. ABB-CE submitted a revision to the DCD on February 22, 1995, which addressed the staff's comments. Additional revisions to the DCD, based on additional discussions with the staff, were submitted by ABB-CE on March 24 and March 27, 1995. These revisions to the DCD are noted by a bar in the margin next to the change and a [2/95] footnote at the bottom of the page. The February 1995 revision was the last revision the NRC staff approved before issuing the notice of proposed rulemaking for the System 80+ design in the Federal Register on April 7, 1995. Subsequently, ABB-CE proposed additional changes to the System 80+ design documentation as a result of its review of the System 80+ design details. These changes were proposed in letters dated June 11 and July 17, 1996, and finalized in letters dated June 27 and July 25, 1996, respectively. The staff's review of this information is included in the appropriate sections of this FSER supplement.

ABB-CE submitted revised DCD pages, which incorporated the above design changes and corrected various editorial and typographical errors, for the staff's verification by letter dated December 13, 1996. The substantive changes to the DCD were identified by a margin bar adjacent to the change and a footer date of [11/96]; editorial or typographical changes also have a footer date of [11/96] but do not contain margin bars. ABB-CE submitted the final version of the DCD on April 30th and provided corrections on May 7, 1997. This revision of the DCD includes conforming changes to the DCD introduction, seismic site parameters, and inspections, tests, analyses, and acceptance criteria (ITAAC). These revisions are identified with a footer date of [1/97]. The final version of the System 80+ DCD is approved by this supplement to the FSER and is the version that will be incorporated by reference into the design certification rule for the System 80+ design.

1.6 Index of Applicable Regulations and Exemptions

In the FSER, the NRC staff identified new standards for selected technical and severe accident issues for the System 80+ design that were addressed and resolved during the design certification review. These new design standards were consequently included as additional applicable regulations in the proposed rule for the purposes of 10 CFR 52.48, 52.54, 52.59, and 52.63. The Commission decided not to codify the additional applicable regulations in the final rule, but the Commission did set forth its intent with regard to these new design standards in its SOC for the final design certification rule (See the SOC public comment summary and resolution section on the need for additional applicable regulations).

1.9 Index of Tier 2* Information

In the FSER, the NRC staff stated that any changes to certain SSAR commitments would require prior NRC approval before the change was implemented by a COL applicant or licensee who referenced the System 80+ design certification. The staff listed these SSAR commitments in the FSER, and required that they be identified in the DCD as "Tier 2*" information. ABB-CE identified the Tier 2* information in the appropriate sections of the DCD.

In various locations in the FSER, the NRC staff stated that any changes to Tier 2* information would involve an unreviewed safety question (USQ) and, therefore, require NRC review and approval prior to implementation. This statement regarding USQs was used simply to indicate that the change process for Tier 2* information would be the same as that for proposed changes to other Tier 2 information that is determined by an applicant or licensee to be a USQ. However, a determination of whether or not a proposed change to the Tier 2* information would constitute a USQ has not been made by the NRC, and the actual process for changing Tier 2* information is described in the final design certification rule. Therefore, the language in the FSER has been modified to conform with the language of the final rule and its SOC by the errata in Appendix F to this supplement. See the rule and the SOC section-by-section analysis regarding the process for changes and departures, and the SOC public comment summary and resolution section regarding the Tier 2 change process.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, & SYSTEMS

3.2 Classification of SSCs

ABB-CE proposed revisions to DCD Table 3.2-1, which changed the seismic and safety classifications of the Safety Depressurization System (SDS) spargers, vacuum breakers, and discharge piping so that all components in the discharge portion of the SDS are classified consistently. In addition, the spargers and vacuum breakers were deleted from the list of safety class 1, 2, and 3 valves in Table 3.2-2. As a result of these revisions, the spargers, vacuum breakers, and piping in the discharge portion of the SDS will be classified as non-nuclear safety (NNS), but Seismic Category II. The Seismic Category II classification assures that a failure or interaction of any of these NNS components will not degrade the functioning of a Seismic Category I structure, system, or component to an unacceptable safety level, which meets Position C.2 in RG 1.29. The NNS classification of the discharge piping downstream of the pressurizer safety valves is consistent with that of current operating PWRs and is acceptable. The staff concludes that these changes are for consistency and do not affect the findings in the FSER.

3.6.2.1 Pipe Break Criteria for High-Energy Piping Systems

ABB-CE proposed revisions to DCD Table 3.6-3, "High Energy Lines Within Containment." These revisions update the table to reflect the current design of the Pressurizer and Safety Depressurization System. All four pressurizer safety valves are now mounted directly on the pressurizer and the Rapid Depressurization Line extends from the pressurizer to relief valves RC-408 and 409. Therefore, since the discharge portion of the Safety Depressurization System is not classified as high energy piping, the high energy lines in Items 40, 41, 42, and 43 of Table 3.6-3 have been deleted. In addition, Items 58 and 59 have been revised to agree with the current design of the Rapid Depressurization Line. These changes result in criteria that are consistent with the guidelines in SRP Section 3.6.2, and are acceptable.

3.6.3.5 Review of ABB-CE Bounding LBB Analyses

ABB-CE proposed revisions to DCD Sections 3.6.3.7, 3.6.3.8, and Appendix 3.9A, which committed to combine the normal operating loads and the maximum design loads by the absolute summation method and change the factor on load for the leakage crack size from $\sqrt{2}$ to 1. As discussed in criteria #5 of FSER Section 3.6.3.5.2, this is an acceptable alternative criterion for leak-before-break (LBB) and does not change the findings in the FSER.

3.9.3.1 Loading Combinations, System Operating Transients, and Stress Limits

ABB-CE proposed revisions to DCD Table 3.9-2, "Loading Combinations ASME Code Class 1, 2, and 3 Components" to change the table title to "Loading Combinations for ASME Class 1, 2, and 3 Components and Component Supports." The NRC staff agrees that the loading combinations in Table 3.9-2 are applicable to component supports. Therefore, this change is acceptable and does not affect the findings in the FSER.

3.12.5.4 Damping Values

ABB-CE proposed revisions to DCD Section 3.7.1.3, Figure 3.7-32, Table 3.7-1, and Appendix 3.9A. These revisions changed the maximum allowable damping value for piping analyzed using the uniform envelope response spectrum method from the ASME Code Case N-411-1 values to a 5% value for all modes of vibration. The revised Table 3.7-1 contains a footnote stating that when the 5% value is used for such piping, the conditions in RG 1.84 for using CC N-411-1 will apply even though Code Case N-411-1 is not being used. Piping analyzed using either the time history or independent support method will use the appropriate values in Table 3.7-1.

In section 3.12.5.4 of the FSER, the NRC staff reported that as an alternative to the RG 1.61 damping values, which are in Table 3.7-1, variable damping values in accordance with the requirements and limitations of the ASME Code Case N-411-1 may be used, subject to the conditions given in RG 1.84 relative to the use of Code Case N-411-1. In its evaluation of the above changes, the NRC staff considered the following inherent conservatisms implicit in the overall DCD criteria:

1. Implementation of the conditions specified in RG 1.84 will generally result in a conservative design.

2. The use of the uniform 5% value could result in a small underprediction of support loads and piping deflection at higher frequencies. However, because the DCD (and other ALWR) seismic criteria are (1) based on ground response spectra as defined in RG 1.60 that are enhanced in the high frequency range (approximately 8-40 Hz), and (2) anchored at a relatively high peak ground acceleration value of 0.3g, the NRC staff finds that the use of the uniform 5% damping is acceptable only for use on ALWRs.

On the basis of the above evaluation, the staff has concluded that use of the uniform 5% damping value when implemented with the seismic and piping design criteria in the DCD will provide piping designs with margins which are consistent with those of designs using Code Case N-411-1, as limited by RG 1.84, and is therefore acceptable.

4 REACTOR

4.2 Fuel System Design

ABB-CE proposed changes to Section 4.2.2.4, "Control Element Assembly," (CEA) of the DCD and Figures 4.2.11, 4.3-46, and 4.3-47. These changes required conforming changes to the Tier 1 design descriptions in Section 2.2.2 and Figure 2.2.1-3. The objective of the changes is to allow: (1) for the possibility of having 4-element CEAs at twelve specific core locations; and (2) for the possibility of replacing 4-element CEAs with 12-element CEAs at specific core locations. These changes are covered and bounded by the acceptance criteria in Tables 4.1 and 4.2 of the FSER and are, therefore, acceptable.

ABB-CE also proposed an addition to Table 4.2-3 in the DCD. This change did not affect the findings in the FSER and, therefore, the change is acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2.1 Applicable Code Cases

ABB-CE proposed a change to Table 1.8-7, "ASME Code Cases Applicable to System 80+" of the DCD, which added ASME Code Case N-498, "Alternate Rules for 10-Year Hydrostatic Pressure Testing for Class 1 and 2 Systems, Section XI, Division 1." Code Case N-498 was endorsed by the staff in RG 1.147, Revision 9, dated May 13, 1991. Therefore, the staff finds the change of adding Code Case N-498 to be acceptable.

Implementation of ASME Code Case N-498 reduces the number of hydrostatic tests by five during the life of a plant that references this design. Therefore, changing the number of hydrostatic tests in Table 3.9-1 from 15 to 10 is acceptable. In addition, some conforming changes to page 5-4 of the FSER are provided in the errata in Appendix F to this supplement.

5.4.3 Shutdown Cooling System

ABB-CE proposed a change to Table 5.4.7-2, Item 3. The shutdown cooling pump discharge valve failure mode was changed from the "Fails Open" to the "Fails Closed" position, which accurately reflects the intended design purpose of the shutdown cooling system (SCS), as indicated in the FSER. The NRC staff concludes that the change will not alter the intended design of the SCS, which is used to provide cooling capability to the reactor during plant shutdown and transients. Therefore, the staff finds the proposed change to page 5.4-40 of the DCD acceptable.

6 ENGINEERED SAFETY FEATURES

6.5 Containment Spray System

ABB-CE proposed changes to Section 6.5.3.4 of the DCD for System 80+ concerning the minimum available net positive suction head (NPSH) for the shutdown cooling (SC) and containment spray (CS) pumps. The proposed changes are marked on page 6.5-23 and on Figures 6.3.2-1A and 1B.

Based on prospective pump vendors' data, ABB-CE found that the NPSH available to the SC pumps was insufficient when aligned for containment spray. However, there was adequate NPSH available to the SC pumps during shutdown cooling operation. As a result, ABB-CE proposed to increase the size of the crossover pipe between the shutdown cooling system (SCS) line and the containment spray system (CSS) line, and the CS pump suction line from 18-inch to 20-inch. The licensee has recalculated the available NPSH and the piping flow rates for the SC and CS pumps based on the new pipe size. The maximum allowable containment spray flow rate was reduced from 6500 gpm to 5500 gpm which was the value used in the safety analysis for containment spray. The minimum available NPSH to the SC pump was calculated to be 19.6 feet at 5500 gpm which exceeds the required NPSH of 18 feet for a typical CS or SC pump at runout flow. Therefore, ABB-CE concludes that the CSS would have adequate NPSH during all modes of operation.

The NRC staff reviewed ABB-CE's submittal and concludes that the proposed design change is acceptable on the basis that:

- The change is required to conform with the SC pump design.
- The revised NPSH available to the SC pumps will exceed the required NPSH when the SC pump is aligned for containment spray.
- The reduced containment spray flow rate is the value used in the previous safety analysis.

6.7 Safety Depressurization System

ABB-CE's preliminary design analysis for the safety depressurization system (SDS) used 6-inch piping and valves that would allow the reactor coolant system (RCS) pressure to be sufficiently reduced from 2500 psia to 250 psia following a total loss of feed water (TLOFW) event, in which the most limiting condition was assumed that steam generator feedwater was not recovered and feed and bleed for once-through cooling was not initiated. The SDS also provides other rapid RCS depressurization flows in TLOFW events with safety injection available to prevent core uncovery while maintaining a minimum required mixture level of 2 feet above the reactor core.

A recent detailed engineering study by ABB-CE concluded that the SDS using 4-inch piping and valves would provide enough RCS pressure relief capability to preserve the validity of the original TLOFW analysis. The resized rapid depressurization valves (4 inches) would open no less than 2 hours after the pressurizer safety valves first lift and allow the RCS pressure to be reduced from 2500 psia to 250 psia prior to the reactor vessel melt-through for a severe accident.

The NRC staff concludes that the SDS using 4-inch piping and valves would provide adequate RCS depressurization capability as required to mitigate TLOFW events. Testing requirements to validate the SDS valve flow capacity and other related test requirements as indicated in SDS inspection, tests, analyses, and acceptance criteria (ITAAC) will remain valid for the new piping and valve size. Therefore, the NRC staff finds the proposed changes on pages 5.4-45 and 6.7-17, and Figure 5.1.2-3 of the DCD acceptable.

9 AUXILIARY SYSTEMS

9.3.4 Chemical and Volume Control System

The currently approved System 80+ chemical and volume control system (CVCS) design includes an interlock in the charging pump controls so that both charging pumps cannot be operated at the same time during all modes of operation. The interlock was added as part of a protection feature that prevents an inadvertent boron dilution during Mode 5 operation, in which the lowered reactor coolant volume leads to a smaller dilution time constant and results in the fastest dilution rate and, therefore, yields the shortest time to a complete loss of shutdown margin.

A recent study performed by ABB-CE identified this feature as a potential operational problem because the interlock requires that one charging pump must be completely shut down in order to switch to the standby charging pump. In the process of shutting down the operating pump and switching to the standby pump, there will be periods in which reactor coolant pump (RCP) seal injection cannot be maintained. To eliminate this potential problem, ABB-CE proposes to delete the interlock signal and implement minor modifications to the CVCS, which would still validate the upper limit assumed in the boron dilution analysis. The boron dilution analysis for plant operations in Modes 2, 3, 4, and 5 indicates that, with a maximum charging flow rate of 160 gpm, the dilution time to reach the minimum margin is between 2.5 and 3.2 hours, as long as the RCPs are operating. In Mode 5, with no RCPs operating, the dilution time to reach the minimum margin is between 1.2 and 1.3 hours.

To preserve the maximum charging flow rate of no more than 160 gpm used in the design-basis accident analysis for inadvertent boron dilution, a design modification is proposed to include a flow indicator controller and isolation valves in the charging pump piping discharge, which would limit the maximum combined charging flow rate to 160 gpm when both pumps are on-line. The system flow will be controlled and monitored in the control room. This system modification will provide the flexibility for plant personnel to switch from the operating pump to the standby charging pump for maintenance purposes, by bringing both charging pumps momentarily on-line with the combined maximum flow of no more than 160 gpm. RCP seal injection will also be maintained.

The NRC staff concludes that a modification to the CVCS to allow both charging pumps on-line, momentarily, with a maximum allowed combined charging limit of 160 gpm, while maintaining the RCP seal injection, is acceptable in place of the currently approved interlock signal, whereby one pump must be completely shutdown before the standby pump is allowed to be on-line, resulting in a momentary loss of seal injection. The new design feature does not alter the results of the present safety evaluation. Therefore, the NRC staff finds the proposed changes to DCD pages 9.3-29 and 15.4-11 and pages 2.7-56 and 59 of Tier 1 acceptable.

ABB-CE also proposed three additional changes to Section 9.3.4 of the DCD. The first change corrects an inconsistency in Section 9.3.4.2.1. In Figure 9.3.4-1, Sheet 2, of the DCD, it is shown that the fluid leaving the purification ion exchanger is returned to the Reactor Coolant System by the charging pumps and not by the shutdown cooling pumps, as it was erroneously stated on page 9.3-30 of the DCD.

The second change corrects some errors in Section 9.3.4.3.1 of the DCD, which addresses the redundancy of components in the chemical and volume control system. The description of the redundancy for the seal injection and purification filters on page 9.3-37 referred to "pumps" instead of "filters." This was obviously incorrect.

The third change increases the normal operating pressure for the Volume Control Tank (VCT) from 20 psig to 20-50 psig. This higher operating pressure is needed to maintain sufficient hydrogen pressure in the VCT gas space, to keep dissolved hydrogen in the VCT water at between 15 and 50 cc H_2 (STP)/kg of water. This value is specified in Table 9.3.4-1A of the DCD. The increase in the normal operating pressure in the VCT to 50 psig does not pose any safety concern because there is still a 50% margin left to the design pressure.

The NRC staff finds these three additional changes to be acceptable because they do not change the findings in the FSER.

10 STEAM AND POWER CONVERSION SYSTEM

10.3 Main Steam Supply System

ABB-CE proposed changes to Section 10.3.2.3.2.1 and 10.3.4 of the DCD. The changes corrected the main steam isolation valve bypass valve closing time from 10 seconds to 5 seconds or less. The NRC staff reviewed the changes and found them acceptable because the 5 second closure time was used in the safety analysis.

16 TECHNICAL SPECIFICATIONS

ABB-CE proposed changes to Technical Specification 3.5.4 In-containment Refueling Water Storage Tank (IRWST). Specifically Figure 3.5.4-1, which provides a curve of containment atmosphere temperature vs. the IRWST water temperature. The current figure has an IRWST temperature range from 40°F to 110°F. The proposed change is to revise the scale for the IRWST temperature range from 60°F to 110°F. There is no change to the curve itself. The basis for the change is to achieve consistency between the Technical Specifications and the assumptions in the safety analysis. A minimum allowable IRWST temperature of 60°F was assumed in the containment pressure analysis in Section 6.2.1.5.3.4, "Active Heat Sinks," in the DCD.

The other affected DCD sections are Appendix 16A, Section B 3.5.4, and Chapter 6, Table 6.2.1-22. An example of IRWST temperature of 53°F corresponding to a containment temperature of 90°F is given in the Technical Specification bases on page B 3.5-25; it will be revised to conform with the change on Figure 3.5.4-1 of Chapter 16. The revised example shows an IRWST temperature of 81°F and a corresponding containment temperature of 110°F. For consistency, Table 6.2.1-22 of Chapter 6 will also be corrected to change the refueling water temperature from 80°F to 81°F.

The NRC staff has reviewed the above changes and finds them acceptable because the original curve has not been changed and the changes were made to achieve consistency in the System 80+ documentation.

19 SEVERE ACCIDENTS

19.1 Probabilistic Safety Assessment

ABB-CE proposed changes to Section 19.7, "External Events Analysis," of the DCD. ABB-CE deleted the component and human error failure probabilities from Tables 19.7.5.1-1 and 19.7.5.3-1 but retained the high confidence of low probability of failure (HCLPF) values. The deletion of the quantitative portions of the design-specific probabilistic safety assessments is consistent with the NRC's guidance for preparation of a DCD, as discussed in the statements of consideration for the final design certification rules. The retention of the HCLPF values is necessary to meet commitment #2 in Table 19.15-1 of the DCD. ABB-CE also deleted Tables 19.7.5.3-2 and Tables 19.7.5.4-1 through 19.7.5.4-7. These Tables did not provide significant insights and would not be needed by an applicant referencing the System 80+ design. The NRC staff has reviewed these proposed changes to the DCD and found them to be acceptable.

19.2 Severe Accident Performance

ABB-CE proposed a revision to Section 19.11.5.4.6.1 of the DCD in order to achieve consistency between the description of the reactor coolant system (RCS) response characteristics and Table 19.11.5.4.6-1 and Figure 19.11.5.4.6.1-2 that are referenced in this section of the DCD. The analysis of the RCS response is not affected by these changes and, therefore, the NRC staff finds the change to page 19.11-145 of the DCD acceptable.

19.3 Shutdown Risk Evaluation

ABB-CE proposed changes to the system-level monitoring of the reactor vessel coolant level. One of the reactor water-level monitoring capabilities is provided by the currently approved refueling heated junction thermocouples (HJTCs). This HJTC system provides narrow range indications, with an accuracy to within plus-or-minus 1 inch, of the reactor vessel water level during mid-loop operations via measurement of the reactor water level in the hot-leg region. Operation of this refueling HJTC system is limited to those periods when the reactor vessel head is installed.

A study by ABB-CE concluded that a permanently installed mid-loop reactor water level measurement system using submerged HJTCs in a tank connected to the reactor coolant system (RCS) hot-leg piping will continuously measure the RCS water level during refueling operations when the RCS is in a reduced-inventory or mid-loop condition. This system will also provide an RCS hot-leg water level indication when the reactor vessel head is detensioned and removed, as compared with the currently approved system in which operation of the HJTCs connected to the reactor vessel head is limited only to those periods when the reactor vessel head is installed.

This new mid-loop HJTC system consists of an instrument installed in a tank connected to the RCS hot-leg piping near the shutdown cooling suction connection. The piping of the system tank is connected directly to the top and bottom of the RCS hot-leg and is isolated by a series of isolation valves, with appropriate valve position controls, indications, and displays in the control room. Each RCS hot-leg will have a permanently installed and separate mid-loop HJTC system.

The connecting pipe, up to and including the second system isolation valve from the RCS hot-leg is designed in accordance with ASME Section III, Class 1 requirements. The system's tank and piping downstream of the second isolation valve are designed for RCS operating pressure and temperature in accordance with ASME Section VIII, including the system drain valve. This RCS mid-loop water level system is available only during reduced inventory and mid-loop conditions (Mode 5) and is usually isolated during normal operating conditions (Modes 1 through 4). The mid-loop HJTC instrument consists of a vertical array of the heated and unheated junction thermocouples that provide alarm setpoints for high water levels (water level approaching the steam generator nozzles) and low water levels (water level approaching loss of shutdown cooling suction). The HJTC design will retain the same level of accuracy to within plus-or-minus 1 inch of the RCS hot-leg water level indication and is displayed in the control room. Each HJTC system will have a separate power supply and heater controller to prevent common-mode failure.

The new permanently installed mid-loop level monitoring system will not be affected by refueling activities in which the reactor vessel head is detensioned and removed. The new design should result in fewer water level reading errors and higher instrument reliability by relocating instruments to a more benign area.

The NRC staff concludes that the proposed mid-loop HJTC water-level monitoring system provides a better alternative for measuring the reactor coolant level during reduced inventory and mid-loop conditions as compared to the currently approved HJTC system and, therefore, approves the proposed changes to pages 5.1-1, 7.7-19, 19.8A-47, 19.8A-156, and 19.8A-193 of the DCD.

19.4 Consideration of Potential Design Improvements Under Requirements of 10 CFR 50.34(f)

19.4.6 Cost-Benefit Comparison

In the FSER, the NRC staff utilized a value of $1,000/person-cSv ($1,000/person-rem) averted to estimate that a design improvement that cost more than $17,000 would not be cost-beneficial. This figure conservatively assumed that the total 60-year lifetime risk for the System 80+ design was eliminated by the design improvement (17 person-cSv averted risk x $1,000/person-cSv = $17,000). Since the FSER was issued, the NRC issued "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (NUREG/BR-0058, Revision 2, November 1995). This guidance document adopted a $2,000/person-cSv ($2000/person-rem) conversion factor, subject to present worth considerations, and is limited in scope to health effects. Limiting the conversion factor solely to health effects required that the regulatory analysis include an additional dollar allowance for averted offsite property damage.

The NRC staff reviewed the design alternatives identified in the SSAR using a $2,000/person-cSv averted for health effects and adopting a $3,000/person-cSv supplemental allowance for offsite property (See NUREG/CR-6349, "Cost benefit Considerations in Regulatory Analysis"). Assuming a base case 7% real discount rate as prescribed in NUREG/BR-0058, Revision 2, the present value of the health and safety benefits attributable to a cost-beneficial design improvement would approximate $20,000. A comparable estimate for the health and safety benefits of a cost-beneficial design modification based on a 3% real discount rate, which is recommended for sensitivity analysis purposes, is $40,000.

Most of the candidate design alternatives were estimated to cost more than $20,000 and, therefore, were not cost-beneficial. The only design alternative that cost less than $20,000 is the hookup for portable generators. The estimated cost for this design alternative is $10,000 as shown in Table 19.6 of the FSER. However, given that the hookup for portable generators was estimated to cost on the order of $10,000, under either the 7% or 3% discount rate scenario, this design alternative would have to eliminate at least 50% or 25%, respectively, of the total lifetime risk. Since the hookup for portable generators was estimated to only account for less than 1% of the total risk, even for this most cost-beneficial design modification, the total costs continued to be well in excess of the total benefits.

In summary, the NRC staff concludes that with the significant margins in the results of the cost-benefit analysis, consideration of severe accident design alternatives using the new values provided in NUREG/BR-0058 do not change the findings in the previous analysis in the FSER.

21 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards considered the information discussed in this supplement to the System 80+ FSER during their 433rd meeting on August 8, 1996, and subsequently issued its letter on August 14, 1996. The letter, which follows, reflects approval of the application for design certification and includes no recommended actions for either the NRC staff or ABB-CE.

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

August 14, 1996

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: DESIGN CHANGES PROPOSED BY ASEA BROWN BOVERI - COMBUSTION ENGINEERING RELATING TO THE CERTIFICATION OF THE SYSTEM 80+ DESIGN

During the 433rd meeting of the Advisory Committee on Reactor Safeguards, August 8-10, 1996, we reviewed recent design changes proposed by ASEA Brown Boveri - Combustion Engineering (ABB-CE) relating to the certification of the System 80+ design. These "design changes" consist of both actual modifications to the design and corrections to the documentation to remove inconsistencies and typographical errors. We had the benefit of discussions with representatives of the NRC staff and of ABB-CE. We also had the benefit of the documents referenced.

Conclusions

Our review of Supplement 1 to NUREG-1462, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," did not change the conclusion reached in our earlier report of May 11, 1994. We continue to believe that acceptable bases and requirements have been established in the application to assure that the System 80+ 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.

Background and Discussion

We have been involved in the review of the System 80+ design since ABB-CE applied for certification. This review was carried out in accordance with 10 CFR Part 52, which requires ACRS to report on those portions of 10 CFR Part 52 applications that concern safety. In our May 11, 1994 report to the Commission, we supported the certification of the System 80+ design. This report was included in the staff Safety Evaluation Report (NUREG-1462). The present review is intended to supplement our earlier review of this ABB-CE application.

Sincerely,

T. S. Kress

T. S. Kress
Chairman

References:

1. U. S. Nuclear Regulatory Commission, NUREG-1462, Supplement No. 1, "Final Safety Evaluation Report Related to the Certification of the System 80+ Design," dated July 1, 1996
2. ACRS Report dated May 11, 1994, from T. S. Kress, Chairman, ACRS, to Ivan Selin, Chairman, NRC, Subject: Report on the Safety Aspects of the ASEA Brown Boveri-Combustion Engineering Application for Certification of the System 80+ Standard Plant Design
3. Letter dated June 27, 1996, from C. B. Brinkman, ABB-Combustion Engineering Nuclear Systems, to U.S. Nuclear Regulatory Commission, regarding System 80+ Standard Plant Design Changes
4. Letter dated July 17, 1996, from C. B. Brinkman, ABB-Combustion Engineering Nuclear Systems, to U.S. Nuclear Regulatory Commission, regarding six additional design changes for System 80+ Standard Plant Design

22 CONCLUSION

The NRC staff performed its review of changes made to the System 80+ design documentation by ABB-CE in its letters dated June 27, July 25, and December 13, 1996 and other changes made to conform the System 80+ Design Control Document to the Commission's guidance for the final design certification rules. The design changes were reviewed by the Advisory Committee on Reactor Safeguards as described in Chapter 21 of this report. On the basis of the evaluation described in NUREG-1462 and this report, the NRC staff concludes that the changes to the System 80+ design documentation are acceptable, and ABB-CE's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the System 80+ design

Appendix A

CONTINUATION OF CHRONOLOGY OF CORRESPONDENCE

This appendix contains an update of the chronological list of routine licensing correspondence in Appendix A of NUREG-1462. The correspondence is between the NRC staff and ABB-CE regarding the review of the System 80+ design under Project 675 and Docket Number 52-002. Correspondence regarding the system 80+ design certification rulemaking is not included here, but may be found in the rulemaking records.

July 26, 1994	W. T. Russell, NRC, Letter transmitting FDA for System 80+ in accordance with Appendix O of 10 CFR 52 Fiche: 80366-001/80370-217 acn: 9407280072
July 26, 1994	R. W. Borchardt, NRC, Letter responding to a letter requesting a detailed thermal-hydraulic summary analysis Fiche: 80423-310/80423-314 acn: 9408020147
August 5, 1994	D. M. Crutchfield, NRC, Letter updating guidance on preparation of design control document. Fiche: 80498-192/80498-195 acn. 9408100207
September 22, 1994	R. W. Borchardt, NRC, Letter forwarding three documents used by NRC staff in evaluating severe accident phenomenon Fiche: 81064-001/81064-174 acn: 9409280218
October 12, 1994	C. B. Brinkman, ABB-CE, Transmits input to System 80+ design control document Fiche: 81430-001/81432-207 acn. 9410210016
October 19, 1994	R. W. Borchardt, NRC, Letter providing ABB-CE with necessary corrections to design control document introduction Fiche: none acn: 9410240089
October 28, 1994	C. B. Brinkman, ABB-CE, Transmits language related to System 80+ design control document 2 Fiche: 81587-075/81587-076 acn. 9411010253
November 23, 1994	W. T. Russell, NRC, Letter transmitting revised FDA System 80+ in accordance with Appendix O of 10 CFR 52 Fiche: 81866-088/81866-093 acn: 9411290113
December 8, 1994	C. B. Brinkman, ABB-CE, Forwards marked up CESSAR-DC Figure 5.1.2-2 Fiche: 82110-286/82110-313 acn. 9412230043
December 14, 1994	R. W. Borchardt, NRC, Letter forwarding documents that form the basis for NRC staff severe accident safety evaluation Fiche: 82029-001/82029-317 acn: 9412140052

December 15, 1994	C. B. Brinkman, ABB-CE, Transmits input to System 80+ DCD Fiche: 82164-001/82165-140 acn. 9412280001
December 16, 1994	C. B. Brinkman, ABB-CE, Forwards System 80+ design control document Fiche: 82190-001/82216-048 acn. 9412280070
December 16, 1994	C. B. Brinkman, ABB-CE, Transmits justification for Tier 2 seismic & valve testing expiration Fiche: 82168-007/82168-009 acn. 9412280327
January 6, 1995	C. B. Brinkman, ABB-CE, Transmits Rev. 2 to "Technical Support Document" Fiche: 82334-001/82334-099 acn. 9501120197
January 27, 1995	R. W. Borchardt, NRC, Letter forwarding comments on System 80+ DCD Fiche: 82591-065/82591-1787 acn: 9502010303
February 10, 1995	C. B. Brinkman, ABB-CE, Forwards proprietary parameters list Fiche: 82779-021/82779-026 acn. 9502160100
February 22, 1995	C. B. Brinkman, ABB-CE, Forwards revised design control document Fiche: 82907-001/82911-239 acn. 9502280272
March 3, 1995	C. B. Brinkman, ABB-CE, Forwards additional information on removal of auxiliary throttle coolers Fiche: 83045-290/83045-291 acn. 9503100148
March 14, 1995	D. M. Crutchfield, NRC, Letter discussing ABB-CE submitted views and positions Fiche: 83153-322/83153-323 acn. 9503170207
March 16, 1995	R. W. Borchardt, NRC, Letter forwarding environmental appraisal of the severe accident design alternatives Fiche: 80252-302/80252-322 acn: 9503240046
March 17, 1995	C. B. Brinkman, ABB-CE, Forwards revision pages to the DCD Fiche: 83178-254/83178-314 acn. 9503200178
March 24, 1995	C. B. Brinkman, ABB-CE, Forwards revision to the design control document Fiche: 83329-001/81329-142 acn. 9503290050
March 24, 1995	T. R. Quay, NRC, Letter discussing status of a ABB-CE request to withhold proprietary information from public disclosure Fiche: 83681-304/83681-309 acn. 9504280181

March 27, 1995	C. B. Brinkman, ABB-CE, Forwards correction pages to the design control document revisions Fiche: 83393-326/83393-329 acn. 9504040329
January 24, 1996	T. R. Quay, NRC, Letter discussing clarification of System 80+ PRA results Fiche: 86893-007/86893-009 acn. 9601290077
May 15, 1996	B.K. Grimes, NRC, Letter discussing final DCD for System 80+ design Fiche: 88297-307/88297-308 acn. 9605170020
June 27, 1996	C. B. Brinkman, ABB-CE, Forwards finalized changes to the System 80+ design control document Fiche: 88842-249/88842-351 acn. 9607010016
July 17, 1996	C. B. Brinkman, ABB-CE, Forwards draft changes for staff review and approval Fiche: 89067-009/89067-025 acn. 9607180288
July 25, 1996	C. B. Brinkman, ABB-CE, Forwards finalized change package of six draft changes to the System 80+ design control document Fiche: 89227-220/89227-236 acn. 9607310161
October 1, 1996	S. L. Magruder, NRC, Letter forwarding RAI on small break LOCA Fiche: 89911-001/89911-061 acn. 9610030258
October 22, 1996	S. L. Magruder, NRC, Letter forwarding status of a ABB-CE request to withhold company proprietary information Fiche: 90546-191/90546-195 acn. 9610250172
December 13, 1996	C. B. Brinkman, ABB-CE, Forwards revisions to the System 80+ DCD Fiche: 91156-001/91156-324 acn. 9612180372
December 16, 1996	F. J. Miraglia, NRC, Letter responding to a requested status of CESSAR-DC relative to FDA for System 80+ Fiche: 91101-194/91101-195 acn. 9612160082

Appendix D

CONTRIBUTORS TO THIS FSER SUPPLEMENT

NAME	RESPONSIBILITY
Anthony Attard	Fuel System Design
Bernard Bordenick	Legal Review
Herbert Brammer	Mechanical Engineering
David Diec	Reactor Systems
Jin-Sien Guo	Plant Systems
John Huang	Mechanical Engineering
Shou-Nien Hou	Mechanical Engineering
Larry Kopp	Nuclear Physics
Jay Lee	Radiological Analyses
Chang-Yang Li	Plant Systems
James Lyons	Section Chief, Plant Systems
Stewart Magruder	Project Manager
Janice Moore	Legal Review
Son Ninh	Project Manager
Robert Palla	Severe Accidents
Krzysztof Parczewski	Chemical Engineering
Janak Raval	Plant Systems
Nicholas Saltos	Probabilistic Risk Assessment
Dino Scaletti	Senior Project Management, Generic Issues
Michael Snodderly	Severe Accidents
Jerry N. Wilson	Senior Policy Analyst and Project Manager

Appendix F

ERRATA TO THE SYSTEM 80+ FSER

Page, Column, Paragraph	Change
Page 1-12, 1st column, 1st paragraph	In the 3rd entry, change "Applicable regulation for electric power system" to "Applicable regulation for offsite power source to safety division". Also, add the 4th entry with "8.3.1.1 Applicable regulation for alternate power source to non-safety equipment".
Page 1-12, 2nd column, 4th paragraph	In the 2nd entry, change "ACI-349 (1985 Edition for design and construction of internal structures" to "ACI-349 (1985 Edition) for design and construction of seismic category I structures". In the 3rd entry, change "N690 (1984 Edition) for structural design and construction" to "N690 (1984 Edition) for design and construction of steel structures".
Page 2-9, 1st column, 1st paragraph	Replace the 1st paragraph in Section 2.5.2.6 with "The COL applicant will compare site-specific earthquake free-field surface ground motions, assuming a rock outcrop, to the ground motions used as input for the design certification. The COL applicant must verify that these site-specific design response spectra are enveloped by the control motion spectra shown in Figure 2.1 of the System 80+ FSER. This action is identified as COL Action Item 2.5-1."
Page 3-43, 2nd column, 1st paragraph	Replace the 1st paragraph with "ABB-CE has presented the site acceptance criteria in Section 2.5.2.5.3 of the DCD. For a rock site, site-specific free-field ground surface response spectra at 5 percent of critical damping in the horizontal and vertical directions will be developed and compared to Figure 2.5-38 of the DCD. If the site-specific response spectra are enveloped by the spectra in Figure 2.5-38, then the site is acceptable. If the site-specific response spectra exceed either spectrum in Figure 2.5-38 at any frequency, a site-specific evaluation can be performed. In this evaluation, a site-specific structural dynamic analysis will be performed and the resulting in-structure response spectra at six critical elevations [foundation basemat elevation (El) 15.24 m (50 ft), interior structure El 27.97 m (91.75 ft), control room El 35.2 m (115.5 ft), top of steel containment vessel El 76.5 m (251 ft), interior structure El 44.5 m (146 ft), and shield building El 80.31 m (263.5 ft)] will be compared to the respective design response spectra in Figures 3.7D-1 through 3.7D-21 of the DCD. If the in-structure response spectra from the site-specific evaluation are enveloped by the in-structure design response spectra, for each of the six elevations, the site is acceptable. If the in-structure response spectra from the site-specific evaluation exceed the in-structure design response spectra, for any of the six elevations at any frequency, the design might still meet the design and licensing commitments due to the substantial design margin between the design commitments and the actual bases upon which the plant was designed. To demonstrate that the plant design meets the design and licensing commitments, a confirmatory site-specific evaluation can be performed to demonstrate that the System 80+ design meets the applicable design criteria for structures, systems, and components when subjected to the site-specific response spectra. The results of the confirmatory site-specific evaluation will be reviewed by the NRC staff."

Page, Column, Paragraph	Change
Page 3-43, 2nd column, 2nd paragraph	Replace the 2nd paragraph with "For a soil site, site-specific response spectra at 5 percent of critical damping in the horizontal and vertical directions at the free-field ground surface will be developed and compared to Figures 2.5-39 and 2.5-40 of the DCD. If the site-specific ground surface response spectra are enveloped by the spectra in Figures 2.5-39 and 2.5-40, then the site is acceptable. If the site-specific response spectra exceed either spectrum at any frequency, a site-specific evaluation can be performed. In this evaluation, in-structure response spectra, at six critical elevations defined above, obtained from the site-specific evaluation will be compared to the respective design response spectra in Figures 3.7D-1 through 3.7D-21 of the DCD. If the in-structure response spectra from the site-specific evaluation are enveloped by the in-structure design response spectra, for each of the six elevations, the site is acceptable. If the in-structure response spectra from the site-specific evaluation exceed the in-structure design response spectra, for any of the six elevations at any frequency, the design might still meet the design and licensing commitments due to the substantial design margin between the design commitments and the actual bases upon which the plant was designed. To demonstrate that the plant design meets the design and licensing commitments, a confirmatory site-specific evaluation can be performed to demonstrate that the System 80+ design meets the applicable design criteria for structures, systems, and components when subjected to the site-specific response spectra. The results of the confirmatory site-specific evaluation will be reviewed by the NRC staff."
Page 3-61, 1st column, 1st paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 3-62, 2nd column, 1st paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 3-65, 1st column, 1st paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 3-99, 2nd column, last paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."

Page, Column, Paragraph	Change
Page 3-104, 2nd column, 5th paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Furthermore, any requested change ..."
Page 3-136, 2nd column, 1st paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 4-3, 2nd column, 2nd and 4th paragraphs	Change "... would involve an unreviewed safety question and require prior NRC review and approval prior to implementation," to "... will require NRC approval prior to implementation," Also, delete the first sentence in paragraph 4 beginning with "Any requested change ..."
Page 5-4, 1st column, 2nd paragraph	Change "... RGs 1.84 and 1.85, ..." to "... RGs 1.84, 1.85, and 1.147, ..." and delete the next sentence beginning with "None of the specified ...".
Page 5-4, 1st column, 4th paragraph	Change "... RGs 1.84 and 1.85, ..." to "... RGs 1.84, 1.85, and 1.147, ...".
Page 6-4, 2nd column, 4th paragraph	In line 5, change "RG1.5" to "RG 1.50".
Page 6-17, 2nd column, 4th paragraph	In line 4, delete "6.3.4".
Page 7-3, 1st column, 2nd paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and acceptance prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 7-8, 1st column, 1st paragraph	Delete "... would involve an unreviewed safety question ..." and the next sentence beginning with "Any requested change ..."
Page 7-9, 1st column, 1st paragraph	Change "... would involve an unreviewed safety question and, therefore, will require NRC review and acceptance prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 7-9, 2nd column, 6th paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and acceptance before implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 7-21, 1st column, 5th paragraph	Change "... would involve an unreviewed safety question and, therefore, require NRC review and acceptance before being implemented." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."

Page, Column, Paragraph	Change
Page 7-32, 2nd column, 4th paragraph	Delete "... an unreviewed safety question would result from ..." the 2nd sentence and replace the period at the end of the 2nd sentence with a comma. Delete "Therefore, any change to these issues ..." from the beginning of the 3rd sentence and delete the 4th sentence beginning with "Any requested changes ..."
Page 7-32, 2nd column, last paragraph	Change "... would involve an unreviewed safety question and, therefore, will require NRC review and acceptance before being implemented." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested changes ..."
Page 9-32, 1st column, 5th paragraph	Delete the 3rd sentence and add "In Amendment Q to the CESSAR-DC, ABB-CE revised Section 9.2.6.2 to state that the CSS consists of a condensate storage tank (CST), piping and two recycle pumps. The minimum capacity of the CSS is based on the maximum usage during startup (e.g., maximum steam generator blowdown vs. startup duration) plus 100-percent margin. The CSS is constructed of stainless steel and has a stainless floating cover minimize air ingress".
Page 9-32, 2nd column, 1st paragraph	In line 3, change "CSTs" to "CST". In line 5, change "CSTs are" to "CST is" and change "are" to "is".
Page 9-38, 2nd column, 4th paragraph	In line 5, change "10 CFR 50.34(f)(s)(viii)" to "10 CFR 50.34(f)(2)(viii).
Page 9-56, 2nd column, 3rd paragraph	In line 4, change "Section III.G, III.J and Appendix R" to "Sections III.G, III.J and III.O of Appendix R".
Page 18-79, 2nd column, 3rd paragraph	In line 14, change "18.2(6), 18.4.2.1(14), 18.4.2.8 and 18.4.2.11" to "18.7.1.8.1 and 18.7.1.8.2".
Page 18-127, 2nd column, last paragraph	Change "... would involve an unreviewed safety question and, therefore, would require NRC review and approval prior to implementation." to "... will require NRC approval prior to implementation." Also, delete the next sentence beginning with "Any requested change ..."
Page 19-60, 1st column, 5th paragraph	In line 2, change "2000" to "200".
Page 19-67, 2nd column, 2nd paragraph	In line 9, change "662.9 m^2" to "62.9 m^2".
Page 19-77, 2nd column, 3rd paragraph	In line 15, change "19.11.4-1" to "19.11.4.4-1".
Page 19-108, Table 19.6, 4th column	Change 8.3×10^{-5} for the Third Diesel Generator to $8.3\times10^{+5}$ and 3.3×10^{-5} for the Filtered Containment Vent to $3.3\times10^{+5}$.
Page 19-109, Table 19.6, 4th column	Change 3.6×10^{-5} for the Refractory Lined Crucible to $3.6\times10^{+5}$.
Page 20-1, 2nd column, 1st paragraph	In line 18, delete "dated December 21, 1992". In Line 19, change "Supplement 15" to "Supplement 15, dated April 1993".

Page, Column, Paragraph	Change
Page A-60, 6th entry	Change "C.B. Brinkman, CE, letter forwarding revised Combustion Engineering Nuclear Fuel. FICHE: 80016 315" to "C.B. Brinkman, CE, letter forwarding copies of Amendment W to CESSAR-DC. FICHE: 80196-001".

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION — **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG-1462 Supplement 1
2. TITLE AND SUBTITLE *Final Safety Evaluation Report related to the Certification of the System 80+ Design* Docket No. 52-002	3. DATE REPORT PUBLISHED MONTH: May — YEAR: 1997 4. FIN OR GRANT NUMBER
5. AUTHOR(S)	6. TYPE OF REPORT Safety Evaluation Report 7. PERIOD COVERED *(Inclusive Dates)* Sept. 1994 - May 1997

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as above

10. SUPPLEMENTARY NOTES

Docket Nos. 52-002 and 50-470; Project Number 675

11. ABSTRACT *(200 words or less)*

This report supplements the final safety evaluation report (FSER) for the System 80+ standard design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) staff as NUREG-1462 in August 1994 to document the NRC staff's technical review of the System 80+ design. The application for the System 80+ design was submitted by Combustion Engineering, Inc., now Asea Brown Boveri - Combustion Engineering (ABB-CE) pursuant to Subpart B of 10 CFR Part 52. This supplement documents the NRC staff's review of the changes to the System 80+ design documentation since the issuance of the FSER. ABB-CE made these changes as a result of its review of the System 80+ design details. The NRC staff concludes that the changes to the System 80+ design documentation are acceptable, and that ABB-CE's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the System 80+ design.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT: unlimited
Asea Brown Boveri - Combustion Engineering, Inc. (ABB-CE) Design Certification Evolutionary Design Final Safety Evaluation Report (FSER) System 80+ Design Subpart B to 10 CFR Part 52	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE